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HANFORD LABORATORIES OPERATION MONTHLY ACTIVITIES REPORT

OCTOBER, 1962

NOVEMBER 15, 1962

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HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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HANFORD LABORATORIES OPERATION MONTHLY ACTIVITIES REPORT OCTOBER, 1962

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Compiled by
Operation Managers

November 15, 1962

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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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: October 31, 1962

	At Beginning of Month		At Close of Month		Total
	Exempt	Salaried	Exempt	Salaried	
Chemical R & D	131	133	131	136	267
Reactor & Fuels R & D	175	163	177	164	341
Physics & Instrument R & D	91	61	92	61	153
Biology	39	59	40	59	99
Operations Res. & Syn.	18	5	18	4	22
Radiation Protection	41	91	41	92	133
Finance and Administration	120	111	122	110	232
Programming	15	3	15	3	18
General	3	4	3	4	7
Test Reactor & Auxiliaries	58	299	58	300	358
TOTAL	691	928	697	933	1,630

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BUDGETS AND COSTS

October operating costs totaled $2, 293, 000, a decrease of $129, 000 from the previous month; fiscal year-to-date costs are $9, 087, 000 or 31% of the $29, 184, 000 tentative control budget. A firm financial plan for the research and development programs has not been released yet by Washington AEC. Hanford Laboratories' research and development costs for October, compared with last month and the tentative control budget, are shown below:

(Dollars in thousands)	COST Current Month	Previous Month	FY To Date	Budget	% Spent
HL Programs					
02 Program	$ 70	$ 78	$ 284	$ 1 069	27%
03 Program	16	1	26	175	15
04 Program	1 019	1 060	3 928	11 418	34
05 Program	87	94	366	1 293	28
06 Program	244	270	987	3 154	31
	1 436	1 503	5 591	17 109	33
FPD Sponsored	104	106	383	1 370	28
IPD Sponsored	93	105	379	1 325	29
CPD Sponsored	100	146	517	1 369	38
Total	$1 733	$1 860	$6 870	$21 173	32%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

A high density (96% TD) UO_2-tungsten cermet capsule was fabricated for irradiation in the MTR. The vacuum-insulated, tungsten-clad capsule will operate at a calculated cladding temperature of 2450 C. The cermet was fabricated by high-rate densification in the Dynapak machine.

The first direct evidence of low temperature (as low as 300-400 C) in-reactor sintering of UO_2 was obtained. Photomicrographs of swaged UO_2 from irradiated PRTR fuel rods revealed sintering at stressed point contacts between fuel particles. This sintering was presumably the result of localized, transient high temperatures in the regions of

[redacted]

individual fission events. Since much of the volume of the fuel element core material operates at relatively low temperatures, this discovery has important engineering implications.

Transverse excitation of cladding was investigated for application to remote fabrication of irradiated or plutonium-bearing fuels by vibrational compaction. Compaction efficiencies of 90-91% were achieved in compacting fused UO_2 in eight-foot-long, 0.505-inch-ID Zircaloy tubes. These results are comparable to those obtained using vertical excitation.

On the basis of destructive examinations and other evidence, a tentative explanation of the cause of failure of an MgO-PuO_2 element in the PRTR last month is as follows: A high local temperature caused by plutonium segregation during core fabrication and/or the presence of impurities, e.g., fluoride, in an internal cladding defect, coupled with release of water by the MgO core material at operating temperature, resulted in locally accelerated corrosion and hydriding. The hydride precipitated as a massive layer on the cooler outside surface of the cladding. Brittle failure at this point resulted in a small crack and a "leaker." Later, waterlogging and/or hydration of the MgO caused a ductile-type enlargement of the crack; core washout followed.

PuC samples aged 14 months show a continued lattice expansion attributed to self-induced alpha damage. There seems to be an increase in the growth rate with time.

The melting point of Pu_2S_3 was measured in vacuo (approximately 10 microns) on a tungsten ribbon and found to be 1725 ± 5 C. In argon, Pu_2C_3 was stable to the temperature limit of the furnace - 2300 C. The material was stable in boiling demineralized water.

An experiment to investigate the irradiation characteristics of ThO_2-PuO_2 solid solution fuel materials has been started. Plutonium acts as an excellent sintering additive in these materials.

Four Zircaloy-clad MgO-PuO_2 capsules irradiated to high exposure (about 10^{20} fissions/cc) are being examined.

A small diameter, fluted tubular fuel element of metallic uranium has successfully completed one cycle of irradiation in the ETR P-7 loop.

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Another experimental element, a tubular dual enriched fuel element of metallic uranium has completed four cycles of irradiation in the ETR M-3 loop.

A total of 500 N-Reactor inner fuel tube supports with superior bend ductility were made from Zircaloy-2 strip produced experimentally from three types of source material. None were rejected because of cracking.

Pressure drop measurements of NPR fuel elements revealed that "buggy spring" type supports between the inner and outer fuel pieces have a lower drag coefficient than "suitcase handle" supports. This lower drag coefficient would cause a slightly higher flow rate in the middle annulus of those fuel elements using the "buggy spring" supports.

Visual studies performed with an electrically heated test section in a glass tube revealed that vigorous boiling can take place in the vicinity of fuel support devices if the fuel elements depart excessively from a concentric position within the process tubes. Boiling was less severe with elliptically shaped than with square-end type support devices.

Corrosion of Zircaloy-2 in gases such as CO_2 has been found to be highly dependent upon the amounts of trace impurities in the gas. Corrosion rates of Zircaloy-2 at 550 C in CO_2, from which all traces of O_2 and H_2O were removed, were much lower than published data and confirm that CO_2 will not act as an inhibitor of accelerated hydriding of Zircaloy-2 in a CO_2 plus H_2 atmosphere.

The prototype device for burst testing irradiated pressure tubes has been operated successfully. Burst tests at 550 F on sections of Zircaloy-2 pressure tubes that had been exposed to 530 F, 1050 psi water, and 3×10^{20} nvt ($E > 1$ Mev) indicated that the increase in ultimate strength caused by neutron irradiation is twice as great when measured at 550 F as when measured at room temperature.

The effect of spring loaded fuel supports on the corrosion of Zircaloy-2 at points of contact with the pressure tube was tested at 300 C in flowing water. Lightly loaded supports penetrated the tube wall at an accelerated rate; however, heavily loaded supports did not penetrate. Apparently, the heavier load eliminated any relative motion between the supports and the tube wall.

The Inconel and stainless steel pressure tubes are being removed from the DR-1 gas loop for tests to measure the effects of neutron irradiation and gas environment on the properties of the metals. In pulling the two tubes from the reactor, the Inconel tube broke at a point about two feet into the flux region leaving the balance of the tube in the reactor. Underwater examination revealed a transverse break that appeared to be smooth and brittle. Tests will be performed to determine whether the apparent embrittlement of the Inconel was the result of neutron irradiation, the exposure to CO_2 at 1000 F, or combination of the two.

A shift from failure by yielding to failure by fracture for cold-worked Zircaloy-2 was observed after irradiation at 540 F to 6 x 10^{14} nvt. A notched tensile specimen containing 40% cold work failed in a brittle manner with relatively little plastic strain.

During the month, 47 bend test specimens were tested at room temperature after having been irradiated in the G-7 ETR Loop. The Zircaloy-2 specimens were irradiated between 3.5 x 10^{19} and 1 x 10^{20} nvt (fast). Weight gain and hardness measurements were made on each specimen before testing.

A creep activation energy of 58,000 cal/gm mole was measured for 20% C.W. Zr-2 in-reactor at 350 C during a reactor outage. A value of 85,000 cal/gm mole had been observed at this same temperature during reactor operation. This difference is apparently related to radiation-induced defects which anneal rapidly after cessation of neutron irradiation at 350 C. This explanation is consistent with observed increases in creep rate shortly after reactor shutdown.

Density measurements on uranium metal specimens irradiated to 0.16 a/o B.U. indicate about a 50% volume change on specimens irradiated at 400 C and a 10-20% volume change on specimens irradiated at 625 C. Irradiation growth leading to severe surface roughening (and possible erroneous density values) and internal tearing are believed responsible for the anomalous temperature dependence of the density changes.

X-ray diffraction observations on irradiated molybdenum foils revealed that both line breadth and lattice parameter have increased after an irradiation dose of 10^{18} nvt (fast). The increases are proportional to carbon content, suggesting that irradiation results in a non-equilibrium solution

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of carbon in molybdenum. Post-irradiation annealing decreases line width and lattice parameter, again in proportion to carbon content.

Molybdenum foils 0.003-inch thick containing various levels of carbon as intentional interstitial impurity were annealed at 1850 C for one-half hour and then irradiated to 10^{19} nvt (fast) at 50 C. After thinning for transmission electron microscopy, all foils show defects in the form of loops and black spots. These defects are very similar in appearance to those which form in irradiated cold worked high purity molybdenum after post-irradiation annealing at 600 C.

Iron foils have been successfully thinned and examined by electron transmission microscopy. Dislocations and a few foreign particles were observed in a foil previously annealed at 600 C.

Three square feet of four-inch A212B, four square feet of six-inch A302B, and 20 square feet of HY 80 pressure vessel plate have been received from the United States Steel Corporation. This material is a part of standardized documented heats of material furnished by USS in 1959 for the purpose of testing in nuclear environments. The material at Hanford Laboratories will be used in the Irradiation Effects on Reactor Metals Program to correlate with data obtained from new heats of these alloys.

A ductile-to-brittle failure transition temperature has been determined with controlled temperature load-deflection tests on precracked samples from NPR carbon steel pipe. Cracks, introduced in the samples by fatigue, propagated in a ductile manner above the temperature range zero to twenty degrees Fahrenheit and in a brittle manner below this temperature range. Depending upon the manufacturing methods, the transition temperature for this steel may range from -50 F to +80 F.

Relatively low exposures at 500 to 700 C of graphites containing additives of Fe_2O_3 and Cr_2O_3 were completed. In most cases the additives decreased the contraction rate and in a few instances caused the samples to grow instead of contract.

An apparent tendency toward saturation of radiation induced contraction of graphite was observed. A lampblack-based graphite sample exposed at 500 to 700 C to 0.9×10^{21} nvt ($E > 0.18$ Mev) contracted 2.24%. An additional exposure of 0.7×10^{21} nvt caused only 0.8% additional contraction.

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The oxalic acid solutions that successfully decontaminated PRTR components in the laboratory were found to become unstable in the PRTR primary system because of the long times required for filling, heating, and draining. A highly contaminated oxalate film formed on the inner surfaces of the piping, but this was successfully removed by an alkaline permanganate-ammonium citrate solution.

Investigations of PRTR cooling during a total power outage were continued with emphasis on cooling by boiling convection (where water would boil in the reactor and be condensed in the steam generator tubes). Calculations indicate that cooling by this method would be satisfactory, provided that leakage from the primary system is not allowed to reduce the liquid level below the top of the fuel.

Reactor kinetics studies of five accidents revealed that the peak power levels reached before the excursion is terminated by a reactor scram are not significantly different for PRTR spike enriched cores and uniformly enriched cores (cores consisting entirely of UO_2-PuO_2 fuel elements).

In a conceptual design study of a Fast Supercritical Pressure Power Reactor, the reference core was found to have a very large negative flooding coefficient, but the void effect was positive and large ($\sim 18\% \, \Delta k/k$), as expected. Methods for reducing the void effect are being reviewed for selection of the most promising approaches to reducing the effect to less than $1 (~ 3 mk).

A report on studies of fuel re-use (regimes involving direct interchange of fuel between fast and thermal reactors) is about to be published.

Based on critical mass calculations for the ORNL and NMPO spacecraft reactors, preliminary estimates were made for the potential size and weight reductions with the use of plutonium fuel. In both cases, significant reductions (about 25%) can be made in the weight of the shielded reactor.

Thirty-five experimental heat transfer runs were made with a full scale, electrically heated model of a 19-rod CANDU Reactor fuel bundle. Coolant mixing, the possibility of steam stratification in the horizontal position, and boiling burnout limits were of interest.

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A hydraulic test of the new nozzle assemblies designed for use on zirconium tubes in Hanford's K-Reactor showed that these nozzles will not cause higher pressure drop losses than those encountered in the nozzles now being used.

Experiments to determine flow rates through reactor process tubes following failure of a Panellit pressure tap connection showed that the decreased flow would not be a problem at any reactor except C-Reactor. At C-Reactor the flow decrease could be as much as 50% in a fringe tube, requiring an immediate shutdown to prevent fuel damage.

2. Physics and Instruments

Studies on plutonium uses have shown that for compact fast reactors a core size reduction of approximately 50% is achieved when plutonium is used in place of highly enriched uranium. These reactor types are of interest for auxiliary power sources in space applications. The study continues. Phoenix-fueled reactor studies are being performed using more sophisticated methods (TEMPEST-GAM-HFN) than previous calculations (MELEAGER).

In preparation for the startup of the Plutonium Recycle Critical Facility, the writing of detailed procedures for the physics tests has been completed, and they have been distributed for review and comment. Also, an improved preamplifier was developed to increase the signal-to-noise ratio of fission chambers for use in the Facility.

PuC-UC fuels will have physics properties so nearly like those of PuO_2-UO_2 fuels that experimental data obtained with one type may be readily applied to the other type according to results of recently completed theoretical studies.

In the development of calculational methods, a computer code is now available which treats the problem of the rate of power level changes in breeder reactors which have several fissile isotopes present, each with its own fraction of delayed neutrons and these varying strongly with location. Improvements were also made in updating data libraries for several codes and in methods of interpreting measurements directed toward determining the neutron energy spectrum in fuel elements.

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Control rod strengths in the NPR under varying conditions have been studied experimentally using an exponential pile with the following results: (1) The rod strength is decreased by 6% when the control rod coolant is lost; (2) the rod strength is increased by 7 to 11% when the pile is flooded to varying degrees; and (3) the strength of the samarium ball channels is less than the strength of the control rod by about 14%.

Continued technical assistance was provided the NPR project section in evaluation of NPR instruments. Studies continued on the simplification of the NPR plant simulation.

At the Critical Mass Laboratory, additional measurements were made to compare the effectiveness of thick-concrete and light-water reflectors on the criticality of a 14" spherical vessel filled with plutonium nitrate solution. Acid molarities of about 4 and 6 were used. It was found that the required plutonium concentration is about 8% less for the concrete reflector than for the light water reflector.

An experiment to throw light on the effect of nearby reflecting materials on the nuclear safety of plant equipment was performed by introducing an air gap between the criticality vessel and a surrounding concrete reflector. Introduction of a four-inch gap increased, by 80%, the mass required to achieve criticality. It is concluded that small vessels with nearby reflectors may be conservatively assigned to the category of "nominally reflected" for determining critical mass limits.

In performing critical mass experiments, many situations occur in which criticality is achieved with the sphere partially filled. In other cases, the sphere will be subcritical when completely full. Usefulness of these data could be increased if it were possible to interpret them in terms of equivalent full spheres. A method for doing this is now under development.

Development of new and improved instruments for the Critical Mass Lab will be aided by the use of a recently developed method for simulating, on the analog computer, the behavior of critical assemblies during experimental runs.

The ability to nondestructively detect, with good accuracy, small discontinuities in the inner and outer surfaces of 3/16-inch OD instrument tubing was verified on the NPR instrument line tubing. Only one 40-foot section

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out of 8, 300 feet was found to be free of discontinuities.

Trial application of the recently developed Laundry Monitor over several eight-hour shifts has indicated that one person using the system can obtain the same results as six persons using previous methods.

Radioactivity of Alaskan Eskimos continued under study with further analysis of data obtained during the past summer. Detectable burdens of Cs^{134} were found in those having high Cs^{137} burdens, paralleling the situation previously noted in Laplanders by other investigators. This observation is of some interest since the production of significant quantities of Cs^{134} was unexpected in weapons testing.

Atmospheric diffusion experiments have been conducted here for several years using sources at ground level. During the month, five successful experiments were performed using for the first time the same technique with the source at a height of 185 feet on the weather tower. Meanwhile good progress was made on the circuitry for the new experimental portable mast instrumentation system.

In other instrument work, development was completed on a river monitor for use by the Radiation Protection Operation.

Initial investigation indicates that a suitable method of telemetering radiological and other information from animals to central receiving and measuring instrumentation in the Biology Laboratory is feasible.

3. Chemistry

Preliminary results obtained from analysis of the effluent of two experimental reactor tubes which were being fed with deionized water show that the concentrations of the Na^{24} and Ga^{72} radioisotopes may be a useful measure of the aluminum tube corrosion rate.

Progress has been made toward quantitative measurement of parameters for nickel electroplates which would yield satisfactory diffusion-bonded slugs. Capacitance of a uranium electrode electrical cell correlates systematically with prior uranium surface treatment, with minimum capacitance being observed for a treatment which is concluded optimum from empirical considerations.

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Tests have shown that the thickness of nickel plating on uranium metal can be determined to a fraction of a micron by measurement of the fluorescent x-ray intensities of four uranium lines after their attenuation by the nickel plating. Preliminary studies show that these techniques may also be used to measure the degree to which plated nickel has diffused into the base uranium metal as a result of heat treatment.

Preliminary to work with plutonium, initial solid state electrorefining experiments with cerium metal traced with Fe^{59} have shown that the iron content of 25% of the rod can be reduced by a factor of 30 and 50% of the rod by a factor of 10.

Analyses of water samples from well 699-17-5 shows the presence of beta-gamma emitters slightly above the detection limit of 8×10^{-8} μc/cc. The inclusion of this well within the contamination pattern extending from the 200 East area places the detectable limit of beta-gamma contamination to within three miles of the Columbia River.

During cold engineering scale investigations to resolve foaming problems resulting from the formaldehyde denitration of Purex 1WW, tributyl phosphate was found to be an effective antifoam agent.

Linde AW-400 synthetic zeolite in the 14-30 mesh size range has been found to be fully satisfactory for the extraction of Cs^{137} from Purex alkaline supernate waste.

Cs^{137} product solutions resulting from the cesium nickel ferrocyanide recovery process is contaminated with silver ion. Laboratory studies show that the presence of silver ion adversely affects the storage capacity of Linde AW-400 for Cs^{137}.

Removal of the interstitial gas in Linde 4A zeolite ion exchange material (30-35 mesh range) is found to have little beneficial effect on strontium loading kinetics.

A compound, 4-tert-butyl-2-α methyl benzylphenol (ter BAMBP), was synthesized and compared with 4-sec-butyl-2-α methyl benzylphenol (BAMBP) for its ability to extract Cs^{137} from Purex wastes. The former compound was found to have cesium extraction characteristics identical to BAMBP and should be only about one-fourth as expensive to manufacture.

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Difficulty has been encountered in engineering-scale, solvent extraction experiments seeking the selective and sequential stripping of first strontium then the cerium-rich rare earths from a strontium-rare earth laden D2EHPA-TBP-Soltrol organic solvent. Laboratory findings suggest that this difficulty stems from insufficient residence time of the stripping agent in the solvent extraction contactor.

The partition of strontium from cerium in the peroxide-tartrate strontium recovery process is significantly improved in the presence of ethylene glycol, a free radical scavenger.

The Dynapak (a high-energy impact press) was successfully used to produce strontium titanate of high density, i.e., 4.79 g/cc.

Engineering studies show that the use of packed columns with air sparge agitation is acceptable for the dipicrylamine extraction of cesium from Purex wastes.

Partial analytical results of a hot cell Salt Cycle run, which used segments of three-month cooled PRTR element as feed, showed that 98% of the plutonium in the melt was precipitated by a Cl_2-O_2 sparge, and the decontamination factors measured for the electrolytically prepared UO_2 product were: Pu, 75; Zr-Nb, 1; Ce-Pr, 85; Ru-Rh, 20; Pm, $>$ 250; and total rare earths, 60.

A second radiant heat spray calcination run was successfully carried out using full level Purex 1WW as feed. The calcines from the two runs were combined. Nearly 5 kilograms of solids, equivalent to the waste obtained from processing 1/3 ton of uranium, were obtained in a volume of 1.7 liters. Fission product heat maintained the centerline temperature of the pot at about 240 F.

Fallout studies show that of the radioisotopes investigated, all are in the particulate form except I^{131} which appears to be largely gaseous immediately following a nuclear test but increasingly particulate in the following period.

A radioiodine monitor, operating on pile effluent and capable of detecting slug ruptures, was operated successfully over a ten-day period. During the operating period a slug rupture was detected.

4. Biology

Since quachrome glucosate, a promising corrosion inhibitor, may become a constituent of reactor influent water, young chinook salmon will be exposed to various dilutions of this material for five or six months.

The annual salmon nesting survey indicates a normal number of nests for this period of the spawning season.

An interesting incidental finding in our Sr^{90} toxicity test is that dental tartar removed from the teeth of pigs contains as high a concentration of Sr^{90} as the pig's bone. (This high concentration may be due entirely to local adsorption of the material into the tartar during the daily feeding of the Sr^{90}. If this is true in humans, we may have a simple bioassay procedure for people continuously exposed to Sr^{90}.)

About 5 to 10% of radioiodine applied to skin is absorbed and translocated to the thyroid. When the radioiodine is administered subcutaneously, orally, or intravenously, 30 to 50% of the administered dose goes to the thyroid. This work was done in sheep.

Old sheep excreted administered Cs^{137} about one-half as fast as young sheep. This may be another example of the difference age has on permissible limit parameters.

Since elevated glucose in humans treated with DTPA was reported, it has been noticed that blood glucose levels in rats administered DTPA also rises one or two hours after the administration. Twenty-four hours post treatment, the blood glucose levels returned to normal. Insulin was effective in preventing the rise.

After the subcutaneous injection of plutonium nitrate into the legs of pigs, the effectiveness of DTPA was tested. Locally applied solutions of DTPA did not appear to affect movement of plutonium from the site. However, animals that were I. V. injected showed less plutonium in the bone and liver than those whose wounds were treated with DTPA.

Eight dogs are now smoking three to five cigarettes daily in preparation for a study of the effect of cigarette smoking on clearance of radioactive particles.

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The dog breeding program was discontinued because of congested quarters. This will seriously affect the availability of dogs for experiments we are committed to perform beginning one year from now. Early completion of the new runs will permit continuation of our breeding program.

Some success is being achieved in modifying the fatal immune-reaction (secondary disease) which results when irradiated animals are protected by administration of bone marrow from a different species. Thus LAF mice are protected from lethal radiation by injection of rat bone marrow, but most of these animals die within six weeks from the "secondary disease." By pretreating the donor rats at birth with LAF mouse cells, the six-week survival of "protected" mice was increased. There was evidence that in mice treated with the "pre-sensitized" rat bone marrow, the survivors show predominantly rat-type red blood cells; whereas, when treated with ordinary rat bone marrow, there is a reversion to mouse-type cells.

Deuterium oxide produces some degree of toxicity in all living forms tested. The physiological basis for toxicity has not been established. Effects of a D_2O environment are being examined with yeast to find the biochemical basis for differences between D_2O and H_2O.

Deuterated glucose produces a reduced permeability of the cell. Utilization of this deuterated glucose was significantly less in cells grown on H_20 than on cells grown on D_2O when anaerobic conditions were maintained. The opposite effect was noted under aerobic conditions.

5. Programming

In a study of plutonium values in terms of generalized nuclear parameters, the total neutron absorption cross section of the fissionable plutonium isotopes was held constant while the ratio of the fission cross section to the capture cross section was varied. In the case of the Advanced Pressurized Water Reactor concept, which is the only case studied thus far, the most striking result was the observation that there is an optimum value of this ratio. This result indicates that if Pu^{239} had no capture cross section, the reactor would be denied the benefits of the fertility of Pu^{240}, with an associated decrease in the computed plutonium value.

TECHNICAL AND OTHER SERVICES

Eleven new cases of plutonium deposition were confirmed by bioassay analyses during October. Internal deposition of plutonium, estimated to be less than one percent of the permissible body burden for each of ten CPD employees, occurred during the transfer in late August and early September of recycle material from a 234-5 Building hood to PR cans. Exposure to air-borne plutonium contamination without respiratory protection caused the intake. The other new deposition case, also estimated to be less than one percent of the MPBB, resulted from a plutonium contaminated minor injury. The total number of plutonium deposition cases that have occurred at Hanford is 310, of which 227 are currently employed.

Three IPD employees were exposed to high dose rates on the rear face of the 105-DR reactor during charging operations. A large number of tubes had been flush discharged for maintenance and tube inspection. A mechanical procedure used to verify that the tubes were empty had been conducted on all except 12 of these tubes. One of these tubes was not completely discharged. When a new charge of metal was introduced at the front face, irradiated metal was forced into the capped rear nozzle. Activation of the high dose rate alarm system on the rear face of this reactor resulted in immediate evacuation by the three employees from the rear face. As determined by the HM Chamber, the dose rate at 20 feet from the nozzle was about 30 r/hour. Prompt response to the alarm minimized the dose received by the employees to about 0.2 r.

The average concentrations of fallout materials in air at the ten Pacific Northwest locations was 5 $\mu\mu c$ Beta/m^3 during the four-week period of September 22 to October 19, 1962. The average for the week ending October 26, 1962 was 15 $\mu\mu c/m^3$, the highest weekly average obtained since October 1958. The maximum weekly result was 23 $\mu\mu c$ Beta/m^3 noted at both Walla Walla, Washington, and Lewiston, Idaho. A milk sample representing the Pomeroy area and collected on October 25, 1962, was found to contain 190 $\mu\mu c$ I^{131}/liter. A sample collected southwest of Walla Walla on the same date contained 45 $\mu\mu c$ I^{131}/liter of milk.

Analyses of pilot plant test results from the hot die sizing process have provided much useful information to the engineers responsible for developing the process. The pilot plant experiment was designed to evaluate the effects of four process variables on fuel quality, and involved running 54 combinations of a 3^4 experiment in six groups of nine experiments each - the group corresponding to one day's production.

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Work was started to update and complete the draft report of a HAPO production model.

A program has been written to solve for probabilities of weld cracks in primary piping. From these probabilities, distributions of crack length and frequency will be estimated. A separate program is being written to enable analysis of individual sections.

The General Managers' report on the 1962 HAPO Attitude Survey was prepared and a cover letter was drafted explaining the IBM listings of section reports.

Trial runs are in progress for shear-spinning theoretically designed metal blanks on a Floturn machine into certain preselected shapes having near uniform shear characteristics. Preliminary investigations show that the metal appears to be deforming as planned, but more specific results will be known when dimensional and radiographic tests have been completed.

A number of Monte Carlo test cases were run on the GEM Program to determine the precision with which a particular nuclide can be estimated as a function of its half life and the number and half lives of other interfering nuclides. The zero degrees of freedom case is also being programmed into GEM as a degenerate option.

Authorized funds for 11 active projects total \$2, 191, 100. The total estimated cost of these projects is \$8, 595, 000 of which \$1, 352, 000 had been spent through September 30, 1962.

SUPPORTING FUNCTIONS

The Plutonium Recycle Test Reactor remained shut down the entire month for primary system decontamination. Work included system preparation for chemical cleaning, the chemical decontamination process, system inspection and return to normal activities. Chemical flushes with appropriate intermediate demineralized water rinse flushes were conducted in the following sequence:

a. 10% oxalic acid
b. Caustic permanganate
c. 1% oxalic acid
d. Proprietary oxalic acid compound

[redacted]

e. Caustic permanganate
f. 1% oxalic acid
g. Proprietary oxalic acid compound
h. Caustic permanganate
i. Ammonium citrate

Over-all decontamination factor was about 4 as measured by field and external system readings. Internal surfaces were cleaned very efficiently, contact maintenance readings were actually less than those experienced prior to the rupture.

Fuel element decontamination using ultrasonic vibration continued. By the end of the month, a total of 32 elements had been cleaned.

A planned wiring improvement program in the control room has been completed. The program involved installing a better quality wire marker on about 1500 wires, removing unused wires, and rerouting others for easier and quicker trouble-shooting.

Maintenance effort following chemical decontamination principally involved activities directed at reactivation of isolated components and systems, regasketing of the primary system, overhaul of injection pumps, replacement of primary pump seals, and miscellaneous valve repairs.

Final preparation of process specifications for the Plutonium Recycle Critical Facility is underway. Detailed descriptions of startup physics tests have been completed. Startup work included installation of cover blocks, partial completion of safety rod thimble modification, and preparation for the cell pressure test. Seventy manhours were devoted to training, and the training of personnel from PIRDO to supplement operating personnel in the future has been initiated.

Activities in the Fuel Element Rupture Test Facility were directed at project close-out in conformance with the directive completion date of 10-31-62. Accrual items include the following:

1. B Cell shielding
2. Installation of ΔP scram and automatic cooldown instrumentation
3. Emergency depressurizing valve installation
4. Flow limiting orifice installation
5. In-reactor test section installation
6. Fueling vehicle modifications

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During PRTR decontamination activities, the rupture loop makeup ion exchange system was used to supply demineralized water. Capacity was found to be 140,000 gallons compared to a design rating of 115,000 gallons.

Construction of the Gas Cooled Loop for the Gas Cooled Power Reactor Program is 93% complete. The replacement heater was installed, and several punch list items were completed. Bristol-Siddeley has advised that they are changing the journal bearing facings to Teflon impregnated bronze which has been used successfully in other designs. In-loop testing by the vendor is scheduled to begin 10-30-62.

Total productive time in the Technical Shops Operation for the period was 22,885 hours. This includes 14,879 hours performed in the Technical Shops, 3,037 hours assigned to Minor Construction, 4,858 hours assigned to off-site vendors, and 111 hours to other project shops. Total shop backlog is 20,776 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 were 4.3% (846) of the total available hours.

Total productive time of the Laboratory Maintenance Operation was 16,800 hours of a possible 18,720 hours theoretically available. Of the total productive time realized, 89% was expended for HLO components with the remaining 11% of effort directed toward providing service for other HAPO organizations. Overtime worked during the month was 2.8% of total available hours.

There were no Ph.D. applicant visits to HAPO during October, however, four offers were extended. No acceptance or rejection activity occurred. Five offers are currently open.

Five program offers and five direct placement offers were extended. Offers accepted: one program and three direct placement. Offers rejected: three program and two direct placement. Current open offers: five program and two direct placement.

Two Technical Graduates were placed on permanent assignment. Three new members were added to the rolls; there were no terminations. Current program members total 60.

Responsibility for taking Hanford Laboratories' plant and equipment inventories was transferred to Contract and Accounting Operation effective

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October 15, 1962. The clerk assigned to this work was transferred to Contract and Accounting Operation on the same date.

Eighty pages of the Hanford Classification Guide (HW-37965) were revised to reflect changes which have been authorized by the AEC.

HM Parker

Manager
Hanford Laboratories

HM Parker:JEB:mlk

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REACTOR AND FUELS RESEARCH AND DEVELOPMENT OPERATION

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 02 PROGRAM

1. METALLURGY PROGRAM

Corrosion Studies

Effect of Heat Treatment on the Corrosion and Hydriding of Zircaloy-2. The effect of various heat treatments on the corrosion and hydriding of Zircaloy-2 is presently being investigated. Samples cut from Zircaloy-2 sheet stock were heat treated at temperatures ranging from 700 C to 1010 C and times from 10 minutes to 4 hours. Following heat treatment, the samples were either slow cooled (< 30 C/min) or fast cooled (> 300 C/min) through the alpha plus beta region (900 C-1000 C). All samples were then exposed to 400 C steam at 1500 psi. Periodically, duplicate samples representing each heat treatment condition are removed from the test for corrosion and hydriding measurements.

After 56 days of exposure, some effect from heat treating on corrosion can be seen. A general increase in corrosion with increasing heat treatment temperature up to 950 C is evident. This is followed by a slight decrease in corrosion rate with higher heat-treatment temperatures up to 1010 C. The corrosion weight gain for the 950 C samples was 66 mg/dm^2 compared to 60 mg/dm^2 and 43 mg/dm^2 for samples heat-treated at 1010 C and 700 C, respectively. To date, there has been no effect from cooling rates or extended time at temperature.

Heat treatments result in significant difference in corrosion produce hydrogen pickup following 56 days of exposure in 400 C steam at 1500 psi. The percent hydrogen pickup of Zircaloy-2 samples heat treated in the range of 700 C to 800 C characteristically increases from 15 to over 30 percent following the corrosion transition. The Zircaloy-2 samples heat treated at 850 C (high alpha region) and higher have not shown any increase in hydrogen absorption after transition.

Corrosion of Aluminum in Reactor Process Water. Corrosion rates of X-8001 aluminum in pH 6.7 reactor process water have been measured at 100 C and 125 C. Aluminum samples (2" x 1/2" x 0.062", center mounted) are rotated at 1250 rpm to give flow rates of 10 fps at the sample tip. At 100 C a linear corrosion rate was established after 24 hours; the rate was 31 mils/yr. At 125 C a linear rate of 37 mils/yr was established after about 50 hours. The rates were determined at a refreshment rate of 1.5 $1/cm^2/hr$.

Runs have also been made at 150 C. However, the shape of the corrosion curve suggests possible interference from scale formation. This will be checked by analysis of the corrosion films. If scale formation is indicated, the test temperature will be lowered to 140 C.

Water contents of films formed at 100, 125, and 150 C were determined by dehydration at 1300 C. Average water contents are as follows: 100 C, 31.5%; 125 C, 29.2%; 150 C, 27.0%. Theoretical water contents for mono- and trihydrates are 15 and 34%, respectively. While there is a gradual decrease in degree of hydration with temperature, there appears to be no abrupt change of phase of the corrosion film between 100 and 150 C.

Basic Metallurgy Studies

Notch Sensitivity of Zircaloy-2. Zircaloy-2 cladding has on occasion shown unpredicted localized failure under stress. Metallographic examination through the failed section reveals little uniform straining in the remainder of the clad. Irregularities in the cladding such as a notch or thinning during formation might logically contribute to such localized failure.

Flat Zircaloy-2 tensile specimens cut transverse to the rolling direction with lateral face notches were irradiated in a fast flux region (1.6×10^{14} nv) of the ETR. Total exposure was 3.2×10^{20} nvt (>1 Mev). Specimen holders were designed to permit low (127 F) specimen irradiation temperatures. Tensile data were obtained for the irradiated material at 280 C with three strain rates employed. The nominal strength at the base of the notch increased as a function of notch depth with all three strain rates. Curves of strength versus notch fraction were similar to those of unirradiated material with irradiated strength values being higher. As in the unirradiated condition a definite strain rate effect was observed for the irradiated samples. The exposure of 3.2×10^{20} nvt (fast) eliminated any dependence of uniform strain on notch depth. Uniform strains were less than one percent although fracture patterns in all but the deepest notches were ductile. The critical notch fraction (0.3) below which failure occurred outside of the notch for the unirradiated material did not change significantly with irradiation. Sharp defects (0.3 fraction, 0.005-inch root radius) used on these samples greatly exceed stria introduced during the coextrusion process.

Metallic Fuel Development

Fuel Irradiations. The first charge of NPR fuel elements (NAE's) to be successfully irradiated at near prototypic N-Reactor conditions has been discharged from KER Loop 4 after attaining an average exposure of 1250 MWD/T. The elements were irradiated and discharged without incident. Post-irradiation examination in the K-Reactor basin revealed that the first few upstream elements had a very slight film deposit and that the Zr-Be eutectic brazed closures were discolored (grey to white). The extent of discoloration increased in the downstream direction. Outer fuel tube supports were in reasonably good condition, but various degress of damage had occurred to the inner tube supports (buggy spring type). Part of the inner tube support damage is believed to have occurred during post-irradiation basin handling. All clad surfaces were free of bumps, ripples, or striations with no visible evidence of warp. Swelling determinations were made by fuel element displacement measurements. The maximum swelling measured on the inner tubes was about 0.8 v/o, which compares to 1.2 v/o predicted using the empirically developed swelling model derived from KSE-3 data. The maximum swelling measured on the outer tubes was about 1.1 v/o which compares to 0.7 v/o predicted using the swelling model. It is believed that the outer tube may have swelled more than predicted and the inner tube less than predicted because of the differences in geometries. With data now available from three geometries, the influence of geometry dependent clad restraint on swelling can be factored into the empirical swelling model.

An experimental single tube N-Reactor element having two coaxial zones of enrichment in the uranium, 1.6 percent enrichment in the outer and natural uranium in the inner portion, has completed five cycles of irradiation in the M-3 loop of the ETR. The element has accumulated a maximum exposure of approximately 900 MWD/T and is to be measured and examined prior to re-insertion into the reactor. A flux run is scheduled for the next reactor startup which will provide a measurement of the radial flux distribution in the M-3 position. This information will make possible a more accurate estimate of the average and maximum burnup incurred by the test element.

Radiometallurgical examination is complete on the variable braze thickness irradiation test, GEH-468, 69, and 70. GEH-4-70 has been sectioned longitudinally through its entire length and it has been confirmed that the cracking observed in the uranium is confined to the braze heat affected zones of the uranium and that the cracks extend only from the ends of the element to about one inch from the ends. One small, hardly visible crack was found near the center of the element but may have been caused by cutting.

Self-Brazed Closures. Efforts to seam-weld closures having unplated interfaces, as described in the August 1962 report, resulted in gradual buildup of Zircaloy on the small, inside electrode. During the welding run on the fourth closure, the electrical resistance became so high, due to the accretion of Zr-bearing alloy on the electrode, that local liquation of the tube wall took place, and a hole melted through. Closer examination of the completed seam welds showed that a small patch of Cu/Zr alloy had been formed at the electrode contact with each surge of current. Since contamination with Cu reduces the resistance of Zircaloy-2 to hot water corrosion, it was necessary to abandon this otherwise promising approach to the closure problem.

Five specimens processed to generate the in-situ closure braze were autoclaved at $\sim$ 360 C with frequent examination. After four hours, four of them had developed one or more white oxide spots on the inner weld head. Since the caps had been left with a plating-free rim to ensure freedom from contamination in the weld bead, it seemed improbable that these white oxide spots should be due to Cu or Ni in the weld metal. Three of the four were re-vapor-blasted, re-etched, and the entire group returned to the autoclave for the remainder of 72 hours. The white spots on the one that had not been recleaned had increased in size, but they did not recur on those that had been recleaned. Thus, it appears from the standpoint of both location and behavior of the spots, that the contamination resulted from incomplete cleaning. To avoid compromising the process, another batch of 12 closures is being prepared. These have a wider unplated band at each rim, which will further ensure freedom from contamination with braze alloy. In addition, a small number of closure specimens is being prepared for in-reactor irradiation. These are to be six-inch lengths of standard enriched coextruded uranium.

Hot-Headed Closure Studies. A series of hot-headed end closures were made using several thicknesses of copper interface material (approximately 0.2, 1.0, 2.0, and 2.5 mils thick) between the end cap and the uranium and with four- and six-ring projections machined on the end caps. Bonds between the uranium and the end cap were obtained with all thicknesses; however, the bonds obtained with the thinnest copper appeared to be the best. Metallographic examination of samples with the thicker copper showed small islands of copper-rich uranium alloy in the uranium near the bonds. The projection welds on the four- and six-ring projection caps were sound and prevented movement of the uranium into the end cap and element cladding interface. Cross sections of the welded four-ring projection type showed that the areas between the projections at the Zircaloy-to-Zircaloy interface were unbonded. The six-ring projection type caps were made in an attempt to eliminate these void areas between projection welds. The six-ring projection design which was selected has not completely eliminated unbonding but has significantly reduced the amount of unbonding. A modification of the six-ring projection is in process.

<u>N-Reactor Fuel Support Development</u>. The development of Zircaloy-2 strip with superior bend ductility was undertaken because inner supports made from commercially available strip were cracking in the bends.

Strip with sufficient bend ductility to make supports was made reproducibly in the laboratory by: (1) starting with a forged, beta rolled and beta quenched slab and rolling this slab through a carefully controlled schedule of annealing, hot rolling, and cold rolling; (2) starting with extruded rod and reducing the rod by Turks Head rolling to strip; (3) starting with extruded thick walled tubing, slitting the tubing and then rolling the segments to the required strip thickness; and (4) starting with extruded flat plate and reducing the plate to strip by rolling.

Over 100 supports were made from strip formed by using each of the above first three methods and none were rejected because of cracking in the bends. Ductility tests on strip made by the fourth (extruded plate) method indicate it will also make supports. In all the schedules, a relatively low (600 C) annealing temperature during final reduction was found to be necessary. This low annealing temperature permits air annealing and avoids the costly and time-consuming vacuum annealing steps.

Accelerated fatigue tests were made on a large sample of supports from each of three strip processes. All had good fatigue strength, but the strip produced from rolled plate and that produced by the Turks Head process had two times the fatigue life of strip produced from rolled tube segments. Samples from all the candidate strip processes are now undergoing corrosion tests. Mechanical tests on supports made from material from each of the strip processes indicate that in the sized up condition the supports will meet proposed spring constant requirements of greater than 20,000 pounds per inch of deflection.

Two types of machines for forming the supports were constructed. Each operates satisfactorily and is now being fitted with air cylinders so that the equipment will be self-powered.

One of the goals of this work was to determine if inner supports could be made from Zircaloy-2 strip having what is now considered

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marginal or poor bend ductility. Annealing studies in which strip ductility, temperature, and bending strain were parameters show it is technically possible to make crack-free supports from commercially available strip. The question to be resolved is whether or not supports made from this strip will have enough residual ductility to withstand the sizing operation after assembly.

The success of the bend test in predicting the support making capability of thin sheet had led to the application of the test to specimens taken from thick plate. From preliminary results of these tests it appears likely that an undesirable texture in thick plate may be detected. This would permit diversion of the plate to other applications before it is rolled into thin strip. A correlation of tensile test data with bend ductility of annealed Zircaloy-2 strip reveals that the strip will make supports if the ratio of fracture strain in the thickness direction to the strain in the width direction is 0.5 or greater. If the ratio is between 0.25 and 0.5, the material may or may not make supports and if the ratio is less than 0.25 the material will not make supports.

Fabrication of pilot quantities of ductile Zircaloy-2 strip destined for use as N-Reactor fuel element supports was started.

Fuel element supports are required to fulfill mechanical criteria of strength, stiffness, and energy absorption. Various designs have been proposed and fabrication and testing of some designs is being done. In order to provide a method of correlating results and selecting promising new designs, a method of analysis has been developed which can account for various shapes and conditions of restraint. The effect of shift of neutral axis, shear, and normal stresses as well as the bending stresses is included in the analyses. The method has been programmed and numerical results from the first analysis are being checked. Since some of the proposed designs will restrict coolant flow, a method of analysis for the temperature rise under a rectangular area with increased surface resistance to heat flow has been developed. The equations are ready for programming.

Fluted Single Tube Fuel Element. A small diameter fluted tubular fuel element, charged into the ETR P-7 loop on September 3, 1962, has successfully completed one cycle of irradiation. This fuel element contains the same amount of uranium per foot as an N-Reactor inner fuel tube, and irradiation conditions during the test approximated those of N-Reactor. Irradiation of this element will be continued to high exposure to study the swelling capabilities of the fluted cladding. Interim measurements and examinations will be made during the course of the testing.

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Experimental fluted single tube N-Reactor fuel elements for irradiation testing in an ETR pressurized loop facility are being fabricated. The $10\frac{1}{2}$-inch long elements have been prepared for end closures and fabrication of the caps was started. Material for the caps was obtained by extruding Zircaloy-2 to the approximate desired cross section with cleanup machining and etching to give the final end cap fit.

Extrusion Behavior and Properties of Zirconium Alloys. Corrosion tests in 750 F, 1500 psi steam and 680 F water are continuing for a series of zirconium alloys of two oxygen levels and varying tin content. Current exposure is approximately 120 days. After 98 days of exposure in steam, the low oxygen series had generally lower weight gains (47-64 mg/dm^2) than the high oxygen series (50-80 mg/dm^2) and both show generally lower weight gains with increasing tin content. After 90 days in 680 F water test, there are essentially no effects of tin or oxygen content and weight gains are 25-28 mg/dm^2. Tensile testing of these alloys at room temperature, 150 C, 250 C, and 350 C is in progress.

Cladding Deformation Studies. Fabrication of components for a second irradiation test of 125 fuel rods with non-uniform cladding thickness has continued. Twenty-one capsules, each containing three sample fuel rods, have been filled with NaK, helium leak checked, and x-rayed for NaK level. Two of these capsules have been rejected. Ten other capsules have been NaK filled and are awaiting final inspection. Six capsules remain to be filled. Aluminum spacer tubes must yet be welded to each of these capsules before they are ready to be irradiated.

Progress on thermocouple capsules designed to measure the maximum uranium temperature in one capsule of each capsule train, has been slow because of problems in sealing stainless steel thermocouples into the Zircaloy-2 capsules. Modification of parts for swagelok seals has been completed and assembly of the thermocouple capsules will start immediately.

Process Tube Scratching. Test charging of prototype N-Reactor fuel elements in full sized N-Reactor Zircaloy-2 process tubes has shown that scratching of the process tubes may occur under some conditions even with low carbon steel shoes on the supports. An examination is now being made on the section of the full length N-Reactor process in which the scratches started with the hope of learning what initiated them. A test apparatus has been designed and fabricated which will allow direct observation of scratches as they start and progress, and on which fuel element support shoe materials and designs may be tested under controlled conditions.

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KVNS Self-Support. Testing of the first KV rail design (inverted channel section) indicated that it was unsatisfactory. The channel section, upon collapse of the support, provided a potential steam pocket between the support and the fuel cladding. The support was weak at the transition between the tab and the crown. The die was modified to produce a self-support of the same general shape, but with the crown cross section a tee structural member. The tee section strengthens the transition from the crown to the tab, and the shape does not create a possible steam pocket upon collapse of the support. Preliminary strength tests indicate the support will touch the fuel jacket with a load of 100 pounds on the support.

Load deflection curves, obtained by cycling the load to produce incremental deflections of 5 mils, indicate a 50 percent recovery of support height upon removal of the load for deflections in the range of 5 through 20 mils (25 through 50 pounds). Recovery in the 20- through 35-mil deflection range varies from 50 percent at 20 mils to 20 percent at 35 mils (loadings from 50 through 100 pounds). Some recovery is seen even after the tee section is firmly bottomed on the clad. Die modifications are in progress to produce this support configuration for more extensive testing.

Fuel Fouling Detector. Six 6.5-inch long, Zircaloy-2 clad, 1.6 percent enriched uranium rods are being fabricated into fuel fouling detectors (crud probes). Rod ends have been recessed and end caps brazed and welded into place on one end of each. Radiography showed all six brazes to be good, with no evidence of uranium contamination or porosity. Thermocouple wells are being drilled into the other ends. The pieces will have Zircaloy-2 clad thermocouples brazed into place with two pieces to be used to study the brazing process and the other four elements to be evaluated for reactor use.

Some evidence indicated that failure of previous probes could be attributed to corrosion through a Zircaloy-2 stainless steel joint formed by diffusion bonding. Five such joints were fabricated at the time that the fouling probe was prepared. The four remaining joints are now being subjected to 60-day autoclave tests, with provision for three intermediate metallographic examinations during the test period. The corrosion data obtained should indicate the useful life that can be obtained from such a joint in a pressurized water environment.

Welded joints made by fusion welding thin wall tubing to the thermocouple sheath are difficult to make. A method of resistance welding a flange to the thermocouple sheath of like material has been developed. The thermocouple is inserted through hollow electrodes and through one or more flanges or washers. A conventional spot weld is made. The thermocouple transfixes the weld nugget and with proper conditions is welded to it. The resulting flange can then be fusion welded into a fuel element end cap. Stainless steel and Zircaloy thermocouples have been flanged by this method.

Irradiated Fuel Measuring Equipment. The KE Basin installation of remote measuring equipment for NPR fuels has been completed. This remote measuring equipment is capable of precisely measuring ID, OD, wall thickness, warp, and ovality at any position along the length of either component of a two-foot long fuel assembly. Use of this equipment in conjunction with complementary pre-irradiation equipment allows complete evaluation of the dimensional changes in N fuel components as a result of irradiation. Measuring of irradiated NPR fuels will be started in November 1962.

2. REACTOR PROGRAM

Gas Atmosphere Studies

Oxidation of Zirconium in CO_2. The oxidation rate of crystal-bar zirconium has been measured at 550 and 700 C in CO_2 in which the O_2 and H_2O had been removed by passing high-purity CO_2 over graphite at 800 C. The oxidation rate at 550 C was about a factor of three lower than that found for two literature references. This difference in corrosion rate is probably due to the care exercised in the current tests to remove H_2O and O_2. Very low H_2O and O_2 contents are required because of their greater reactivity with zirconium which would tend to mask the Zr-CO_2 reaction.

No literature data for 700 C were found; however, the measured oxidation rate is considerably lower than given for 750 C in a literature reference. A breakaway corrosion process began after about six hours at a weight gain of 15 to 20 mg/dm^2. This is attributed to carbon diffusion into the ZrO_2 destroying the protective character of the film.

Resistance-Measurement Capsule. Although the in-reactor capsule experiment for determining the electrical resistance of zirconium oxide films has been generally very informative, recent observations indicate the capsule should be replaced. After three months of exposure to temperatures of 475 C, the oxide film appears to be spalling, interfering with the electrical resistance response to gas atmosphere composition.

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A somewhat more versatile capsule containing three Zircaloy samples with three graphite electrodes on each sample is being fabricated to replace the capsule now in the reactor.

Graphite Compatibility with Helium Containing Water Vapor. A sample of TSX graphite was oxidized in a helium stream containing approximately 740 ppm of water vapor at 811 C and 890 C. The weight loss rates were 1.71×10^{-6} and 1.51×10^{-5} g/g-hr, respectively. These rates are somewhat lower than those reported previously (see HW-69950). They are, however, more consistent with expected rates according to a review recently made by R. E. Woodley (see HW-75072). It is possible that the water analyses were in error before. Other methods of obtaining water analyses are being considered. The effect of the addition of hydrogen to the gas stream will be investigated as soon as the rates are established with certainty.

Corrosion and Coolant System Development

Decontamination. Buffered oxalate-peroxide solutions are very efficient reagents for dissolving and removing uranium and uranium oxide residues from a reactor system after a rupture. In comparative tests these reagents have been shown to be superior to the carbonate-bicarbonate-peroxide mixtures formerly recommended for NPR.

One disadvantage of the oxalate-peroxide solutions has been the lack of a satisfactory buffer system. If the pH goes too low, excessive corrosion of carbon steel and certain other metals will occur; if the pH goes too high, excessive decomposition of the peroxide takes place. Acetic acid has been used in the past, but reaction with the peroxide is vigorous and causes operating difficulties. Peracetic acid appears to be satisfactory, even though it is explosive above 120 C and cannot be stored for long periods of time. Gluconic acid is also satisfactory, is not explosive, and can be stored.

Both the peracetic and gluconic buffer systems were used in decontamination of PRTR. No operational difficulties were observed. Laboratory studies have shown that both reagents readily dissolve UO_2 at 40 C. If additional tests are satisfactory, one of these reagents will be recommended for use in NPR after a rupture.

Corrosion Testing in Low pH Water with Low Dichromate. Results from coupon samples of X-8001 aluminum alloy show that equilibrium corrosion rates for this alloy are the same in treated water at pH 6.6 with 1.0 ppm sodium dichromate as in water at the same pH with 1.8 ppm sodium dichromate. Measurements of uniform and non-uniform corrosion are being made in the 6.6 pH water with 1.8 and 1.0 ppm dichromate in the 1706-KE in-reactor single pass facilities. Visual examination of K4NS and K4N test elements indicates no difference in non-uniform corrosion in the two types of water. Measurements of uniform corrosion of these test elements and of carbon steel and aluminum alloy coupon samples have not been completed. X-8001 alloy samples continue to corrode at an equilibrium rate of 0.3 mil/month in either the 1.8 or 1.0 ppm dichromate treated water. This is the first testing in which results have been obtained from multiple-exposure periods in a single pass in-reactor facility, thus permitting accurate calculation of equilibrium corrosion rates. The second group of K4NS elements has completed 14 weeks of exposure under this production test.

Corrosion in Graphite Cooling System Water. A test in TF-4 to determine the corrosion rate and crud release rate of A212 carbon steel and 304 stainless steel in simulated NPR graphite cooling system water has been completed. Total exposure was 3000 hours. The carbon steel corrosion rate was 0.1 mil/year. The corrosion rate of the stainless steel and Zr-2 was nil. Crud release did not appear to be a problem. Samples of 17-4 pH stainless steel stressed to yield did not exhibit cracking after 800 hours in this environment.

Structural Materials Development

Ductile-toBrittle Fracture Transition Temperature for NPR Primary Pipe. This work was an outgrowth from the NPR Pipe Weld Evaluation Program,[1] and it consisted of the performance of load-deflection tests using samples which after having been subjected to fatigue tests were cracked in the area of but on the side opposite to the point of load application. The purpose was to learn something about the crack propagation properties of NPR pipe. Test temperature was varied between room temperature and -33 F. The deflection rate was maintained at two inches per minute for each specimen tested.

The data obtained consisted of curves showing the relationship between the load and the deflection at mid-length of the end-supported specimens, and of ratios of brittle and ductile area on the fractured surfaces.

A curve showing a transition from ductile failure to brittle failure was the result of plotting test temperature versus percent ductile area. The load-deflection data have not as yet been found to fit a behavior pattern.

(1) Letter JS McMahon to WJ Love, dated 1/8/62, "Project CAI-816, New Production Reactor Primary Loop Piping Responsibility for Program of Cyclic Testing of Weld Quality A-155 Pipe."

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The transition temperature, which may be called a "fracture appearance transition temperature," is between 0 and 20 F. That is, above 20 F crack propagation occurs by shear and below 0 F crack propagation occurs by cleavage.

Brittle-ductile transition temperatures measured for this steel range from -50 F to +80 F depending upon manufacturing history. Thus, the behavior of these samples of NPR pipe falls in mid-range for the base metal.

Graphite Studies

NPR Graphite Irradiations. Post-irradiation examination of the graphite samples from H-6-1, the third first-generation capsule in the series of long-term irradiations of NPR graphite, is continuing. Preliminary data agree with the data from the samples irradiated in the H-4-1 and H-5-1 capsules. The surface oxidation on the samples in Positions 4 and 5 mentioned last month is being examined in detail. The samples in Position 4 had an average weight loss of 0.4 percent, and those in Position 5 a weight loss of 0.8 percent. Both positions operated at 775 C. The oxidation was apparently catalyzed by some material external to the samples. The material was deposited on the samples after they were assembled in groups of four, since the oxidation patterns show on adjacent samples. Powder scrapings were taken from the oxidized areas. Analyses of these scrapings showed only the same elements and approximate quantities as those from the surface of unoxidized samples. Consequently, the catalyst associated with the oxidation has not been identified.

The probable source of the oxygen was from air trapped in the annulus between the leadout tubes when the lead shielding was melted into place. During this melt-down operation an original crack in the inner leadout tube may have become sealed so the air was not removed when the capsule was evacuated and filled with helium (see March 1962 Monthly Report). During operation the crack may have again opened and allowed air into the capsule. Although there is no other confirming evidence, it is also possible, but not likely, that the thermocouple seal may have developed a leak and allowed air into the capsule from the non-purified nitrogen and helium used as the cover gas on the extension leads.

Construction has started on the third second-generation capsule, H-6-2, of the NPR series. Due to the oxidation mentioned above, only one sample from Positions 4 and 5 will be re-irradiated in H-6-2. The new capsule will contain 11 samples from H-6-1, four from H-5-1, three from H-4-1, and six virgin samples. The capsule is scheduled for insertion in the GETR on November 12.

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Both second-generation capsules in the GETR, H-4-2 and H-5-2, are operating satisfactorily. The former has completed three cycles of irradiation and the latter one cycle.

Thermal Hydraulic Studies

Visual Studies of the Effects of Fuel Supports on Boiling. Laboratory experiments were continued in the study of heat transfer conditions as affected by devices used to center fuel elements in the process tubes of the Hanford production reactors. The test section for these experiments consisted of a 1.340-inch OD electrically heated tube placed inside of a 1.504-inch ID glass tube with water coolant flowing through the annulus between the two tubes. Three different types of centering devices were used in this series of experiments to produce an annulus of 75 percent eccentricity (percent eccentricity is the fraction of the normal annulus thickness that the heated tube is displaced from a coaxial position toward the wall of the process tube). The first consisted of 0.5 inch long ceramic ribs, 0.094-inch wide and ground down to a height of 0.024-inch. The second consisted of BDF reactor type bumpers which were 1-7/8 inches long, 3/16 inch wide and ground down from their normal 0.040 inch height to 0.024 inch. The third type was elliptical shaped bumpers (also called submarine bumpers) which were one inch long, 0.2 inch wide at the middle and ground down to a height of 0.024 inch.

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

Run	Conditions	Heat Flux Btu/hr-sq ft	Comments
42	Fringe zone - ceramic ribs, 0.5x0.094x0.024,	125,000	Bubbles started to form on heated surface at downstream end.
	75% eccentricity	177,000	A continuous stream of bubbles issued from end of most downstream rib.
		229,000	Stream of bubbles formed at downstream rib covered approximately 150 degrees of the circumference of the heated surface.
		306,000	Violent vapor generation took place over downstream 8" of heated surface.

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Run	Conditions	Heat Flux Btu/hr-sq ft	Comments
43	Same as Run 42 except central zone conditions	250,000	No bubbles were visible.
		360,000	A stream of very fine bubbles issued from most downstream rib.
		458,000	The bubble stream at most downstream rib was vigorous and spread to cover approximately 180 degrees of the heated surface 4" downstream of the rib.
		600,000	Violent vapor generation was observed over the downstream 8" of the test section. A high noise level was noticed with the formation and collapse of vapor voids.
44	Fringe zone - alternate BDF rail bumpers and elliptical shaped bumpers ground to a height of 0.024 inch	101,000	A few bubbles formed from a rail type device located in the second position up from the downstream end. No other bubbles were visible.
		155,000	Some general surface boiling was observed and the bubble formation at the rail in the number 2 location increased in frequency.
		207,000	Bubbles remained attached to the downstream end of the rail in the number 2 location. Some bubbles were generated at the other rail devices but none at the elliptical shaped devices.
		232,000	Bubbles began to form at the most downstream elliptical shaped device.
		285,000	The run was terminated when one of the devices turned sideways.

The observations showed very definitely that the elliptical shaped devices resulted in less vapor generation than the rail type, and that the accumulation of a vapor or bubble wake was not formed nearly as readily behind the elliptical shaped devices. The results also supported the conclusions reached earlier in the study that considerable boiling would be initiated around fuel centering devices when the fuel was allowed to become greater than 40 percent eccentric within the process tube.

Plans were made to run similar experiments with no centering devices and the heated surface resting against one side of the glass process tube.

K-Reactor Zircaloy Tube Program. The hydraulic characteristics of the front and rear nozzle assemblies of the K-Reactor Zircaloy tube replacement program were re-examined after final design modifications. The pressure drop characteristics of a variety of downstream support charges were also determined. The results of these tests are presented in HW-74247-2. In summary, it was concluded that: (1) the nozzle designs which allow charge-discharge of self-supported fuel pieces will not cause increased pressure drop losses over the present inlet and outlet assemblies; (2) hydraulic considerations cannot be used to justify re-design of the non-supported type of downstream support charge; (3) the downstream support charge which results in the least pressure drop would probably be the most expensive to procure of those studied, but the cost of the support could be minimized by accepting a pressure drop increase across the support charge of about 0.9 percent of the header to header pressure drop.

Leak Rates Through Failed Panellit or Nozzle Pressure Taps. Failure of Panellit pressure tap connections or nozzle pressure tap connections would allow water to leak out on the front face of the reactors and would affect the flow rate through the process tube. An additional concern is that if the tube flow is reduced below certain values, boiling and hydraulic instability could occur in the tube with resultant melting of the fuel pieces unless reactor power was immediately reduced.

Experiments were conducted in the hydarulics laboratory to determine the resultant tube flow rate following such pressure tap failures. The results can be summarized as follows: (1) For Panellit pressure tap failure on B, D, F, DR, H, or K reactors, the resultant leak rate would be quite small, the reduction in tube flow minor, and the failure would not result in hydraulic instability under present operating conditions. (2) Nozzle pressure tap failures at B, D, F,

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DR, H or K reactors would result in only minor reductions in tube flow rate and would not result in hydraulic instability. (3) For a C-Reactor Panellit pressure tap failure, the severity of the tube flow reduction would depend greatly on the type of flow monitor and the location of the failure. For a venturi zone, the tube flow reductions would be minor and would not result in hydraulic instability. For an orifice assembly, the tube flow rate could readily be reduced as much as 50 percent and thus could result in mild or severe fuel damage (depending on tube power) unless the reactor power level was immediately reduced.

NPR Heat Transfer Tests. Copies of the experimental heat transfer program using a test section representing one-half length of a N-Reactor fuel column were distributed to NPR personnel for comment as to its adequacy in supplying information needed for operating limits and hazards analyses. Comments have been received from most of the interested parties and are being incorporated in a final program of steady state and transient experiments.

Pressure Drop Characteristics of NPR Fuel Support Devices. In the course of making pressure drop-flow measurement on NPR fuel elements, data have been obtained with two different types of devices used to support the inner fuel piece within the outer fuel piece. One of these support types was a "suitcase handle" support while the other was a "buggy spring" support. The projected dimensions of both types were the same; 0.246-inch high and 0.25-inch wide.

A drag coefficient as defined by the following equation was calculated for both types of supports.

$$f_D = \frac{A_a}{A_p} \frac{2g}{\varphi V^2} \Delta P$$

where: f_D = drag coefficient
A_a = duct area
A_p = projected support area
g = dimensional constant
φ = density of flowing fluid
V = velocity of fluid in duct
ΔP = pressure drop across support.

It was found that the drag coefficient for the "buggy spring" support was as much as 43 percent less than for the "suitcase handle" support. The reason for this was not immediately apparent but one consequence to be expected would be a higher flow rate through the middle annulus of those NPR fuel elements equipped with "buggy spring" supports.

B. WEAPONS - 3000 PROGRAM

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

C. REACTOR DEVELOPMENT - 4000 PROGRAM

1. PLUTONIUM RECYCLE PROGRAM

Plutonium Fuels Development

Irradiation of Special Elements in the PRTR. Further metallographic examination of the $MgO-PuO_2$ element, which ruptured in the PRTR, shows that massive localized hydriding is present on the outside surface of the Zircaloy cladding in the rupture area and that severe localized corrosion has taken place on the inside surface in the rupture area.

Based on information obtained to date, the cause of failure may be tentatively hypothesized as follows: A high temperature caused by plutonium segregation and/or a localized concentration of impurity, possibly fluoride, in an internal cladding defect combined with a large release of water by the MgO core material at operating temperature led to accelerated corrosion. Corrosion-generated hydrogen dissolved in the cladding and migrated to the cooler outside surface where it precipitated to form a massive hydride layer. The cladding, weakened by corrosion and hydride formation, failed in a brittle manner producing a "leaker". Much later, waterlogging and/or hydration of the MgO caused further splitting of the cladding in a ductile manner, resulting in exposure of a section of the core and a significant amount of core washout. Investigation of this failure is continuing in an effort to corroborate or refute this hypothesis.

Cluster Engineering for PRTR Fuel. Two UO_2 19-rod fuel elements were fabricated for ex-reactor loop testing. One cluster will be used to develop designs of end bracket hardware to eliminate the wear occurring in the PRTR between fuel elements and process tubes. The first cluster assembled uses a one-half-inch wide ring on both end brackets with a diameter about 20 mils smaller than the inside diameter of the process tubes. The other cluster assembled will be used to test the pads ($\frac{1}{4}$-inch wide x $\frac{1}{2}$-inch long contact area) to be added to the irradiated UO_2-PuO_2 fuel elements under eight feet of water in the PRTR storage basin.

Advanced Fuel Element Design. A dummy prototype 19-channel water tube fuel element has been fabricated (H-3-14513). This experimental element was designed as a backup for the 19-rod clusters presently being tested in the PRTR. It compares favorably in every way with the 19-rod cluster, but has the added advantage of less complicated fabrication. By utilizing the vibrational compaction process of loading the element (with a core material such as UO_2-PuO_2) many process steps may be eliminated. The empty fuel element shell and the associated end bracket hardware can be fabricated with no exposure of personnel to radiation hazards. After the shell is completely finished, including autoclaving if desired, the core material is added through three holes in the top head sheet. After loading, only those three openings need to be decontaminated and three plugs welded into place to essentially complete the element. The end brackets are then attached to the three welded plugs at each end of the fuel element.

Extended Surface Plutonium Fuels. A modified design of the zirconium components permitted the assembly and roll-cladding of a bare plutonium-zirconium core which weighed 25 grams.

The finished plate was not contaminated. In the modified design the core was placed in a recessed slot and a close fitting cover was placed on top of the core. Two outer zirconium plates completed the assembly. After rolling, the core was four inches wide x 20 inches long and 0.002-inch thick.

An examination of the clad plates reported on last month which contained copper-plated cores showed no stringers of the core material.

These plates had showed widespread contamination in the bond zone. It had been hypothesized that eutectic Cu-Pu alloy had exuded into the bond zone; however, no visual evidence of plutonium outside the core area was detected radiographically. Since the modified design of the components was successful in preventing contamination, no further work on the plating techniques was planned.

UO_2-PuO_2 Irradiation Capsules. Examination of the last two high density UO_2-PuO_2 capsules is continuing. Data on the capsules are shown below:

	GEH-14-85	GEH-14-86
UO_2-PuO_2, (Mole % PuO_2)	2.57	4.13
Initial Density, (% of theo.)	91	91
Average exposure, (fissions/cm^3)	2.69×10^{20}	1.66×10^{20}
Avg. exposure, (MWD/T UO_2-PuO_2)	9,066	5,600
Max. exposure, (MWD/T UO_2-PuO_2)	10,001	7,080
Total gas release* (ml at STP)	6.41	4.10
Central void formation	Yes	No

*Fission gas, helium, and others.

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The first (GEH-21-13) of the four specimens, each containing UO_2 - 0.154 mole % PuO_2 and UO_2 (1.00 mole % U-235), was irradiated. It was in the active zone of the VH-4 Hydraulic Rabbit Facility of the MTR for three minutes. Radiation readings indicate a level of 35 mr/hr at a distance of three feet in air. The second specimen (GEH-21-14) will be irradiated about October 29, 1962, and both specimens shipped to HAPO this month.

Phoenix Experiment. Current status of the three Al-Pu capsules being irradiated to investigate the effect of burnup on the reactivity of high exposure plutonium is as follows: the sample which contains plutonium which initially had 6.25 percent Pu-240 (GEH-21-1) is now being irradiated for its sixth cycle; the sample containing plutonium with 16.33 percent Pu-240 (GEH-21-3) has been irradiated for six cycles and ARMF transient measurements are being made; and the sample containing plutonium with 27.17 percent Pu-240 (GEH-21-19) has been irradiated for six MTR cycles and ARMF measurements are being made.

ARMF calibration measurements using the special Al-Pu sample which incorporates various flux monitors has been approved.

MgO-PuO_2 and ZrO_2-PuO_2 Irradiation Capsules. The four Zircaloy-clad MgO-PuO_2 capsules irradiated to high exposure (about 100×10^{18} fissions/cc) are being examined. Metallography of this high density sintered pellet core material is about the same as for the lower exposure samples. Center voids and columnar grains have formed on the two high power generation capsules. The PuO_2 second phase in the MgO-13.5 w/o PuO_2 material is randomly and uniformly distributed in this high exposure sample the same as it was in the lower exposure sample. This effect seems to be independent of exposure within the range considered.

ThO_2-PuO_2 Irradiation Capsules. An experiment which will investigate the irradiation characteristics of ThO_2-PuO_2 solid solution fuel materials has been started. ThO_2 enriched with 2, 5, 10, 15, and 20 w/o PuO_2 in the form of high density sintered pellets will be irradiated at different core temperatures to different exposures. Some preliminary sintering experiments have been performed prior to fabricating the fuel pellets. The pellets are being fabricated by wet ball milling ThO_2 and plutonium oxylate together to obtain an intimate mixture of the fine powders. This mixture is then calcined at 500 C to obtain PuO_2. The powder mixture is double pressed since no binder is used and a green density of 60 percent TD is achieved. A strong dependence between the green and sintered density was found. Sintering experiments have been performed at 1500 and 1600 C

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for six hours in a helium atmosphere. ThO_2 containing about 200 ppm CaO sintered to 88 percent TD, whereas ThO_2 containing about 0.1 percent CaO sintered to 95 percent of TD under the same conditions. With 8 w/o PuO_2 added, both the high and low CaO-containing ThO_2 sintered to 94 percent TD. Low CaO containing ThO_2 with 1.8, 5.1, and 15.0 w/o PuO_2 added sintered to about 94 percent TD. From these experiments it appears that PuO_2 additions enhance the sinterability of ThO_2 the same way CaO additions do, and the sinterability of ThO_2 is independent of the PuO_2 concentration within the range considered.

Uranium Fuels Development

High Rate Densification Studies. A series of tungsten-UO_2 cermets for fundamental studies was fabricated by high rate densification at 1200 C and 277,000 psi on the Dynapak machine. Evenly blended mixes of -65 +200 mesh UO_2 and -325 mesh tungsten were compacted with UO_2 loadings of 50, 60, 70, 80, 90, and 95 weight percent. A uniform and continuous tungsten matrix surrounds the UO_2 particles in the 50 w/o tungsten cermet. Above 50 w/o UO_2, contact is apparent between UO_2 particles with increased contact area as the percentage of tungsten in the cermet decreased. The density of the cermets increased from 96.0 percent to 98.4 percent theoretical density as the UO_2 content increased from 50 to 95 w/o.

Fabrication of UO_2-Tungsten Cermet Irradiation Capsule. A tungsten-clad, UO_2-tungsten cermet fuel capsule was fabricated for irradiation in the MTR (GEH-14-361) as part of a general evaluation of uranium and plutonium fuels. The fuel material was prepared by high rate densification of a mixture of 50 w/o tungsten powder (-325 mesh) and UO_2 particles (-65 +200 mesh). A uniform and continuous tungsten matrix surrounds the UO_2 particles. The high density (96 percent TD) cermet was machined and encapsulated in 0.200-inch ID x 0.025-inch thick wall tungsten cladding. The closure of the evacuated tungsten cladding was completed by electron beam welding. The tungsten clad capsule is contained within a larger, evacuated aluminum capsule to provide a low conductivity heat transfer path to the MTR coolant. During the proposed short term irradiation test the tungsten capsule surface temperature is calculated to be 2450 C.

Thermal Expansion of UO_2. The density of UO_2 was measured in the temperature range 1200 to 3100 C by applying Co-60 radiography to make exact measurements of single crystals in sealed, resistance heated, tungsten vessels. Excellent reproducibility and precision were achieved at all temperatures. Density measurements at lower temperatures agree with those of other investigators. The density of the solid at the melting point (2800 C) was 9.76 g/cc, that of liquid was 8.74 g/cc. (The density of UO_2 at 0 C is 10.97 g/cc.) The decrease in density during melting was 9.6 percent, and the coefficient of linear expansion of the liquid in the range 2800 C to 3100 C was 37 x 10^{-6}/C.

Heat of Fusion of UO_2. The heat of fusion of UO_2 was determined to be 3300 cal/mole by using data from measurements of the change in volume on melting and of the surface tension of UO_2. These data also can be used to calculate the dependence of melting point on pressure. The results indicate that an increase of 8 C in the melting point should occur under 100 atms pressure.

Electron Microscopy of UO_2. It is highly desirable to obtain a method of replicating large areas of irradiated fuel rods. Experiments with two new replicating materials provided promising results. Room temperature curing silicone rubber or polystyrene dissolved in ethylene dichloride are applied to the specimens as liquids that solidify in a reasonable time to yield a cast replica. The polystyrene replica method is more convenient than the commonly used cellulose acetate or polyvinyl alcohol replica. The silicone rubber replicas appear to be useful for optical microscopy but have not yet been obtained with sufficient resolution for electron microscopy. Investigation of both methods is continuing.

UO_2 Relocation and Melting Experiment. Continued examination of an irradiated UO_2 fuel capsule which originally contained a spiral array of small tungsten "marker wires" (0.014-inch OD x $\frac{1}{4}$-inch long) revealed the presence of tungsten in the lower regions of the initially molten zone. Some wires (in their original shape) were discernible, but also present were a number of small, odd-shaped pieces of tungsten from which one can infer that the melting point of tungsten (3410 C) was reached, at least momentarily, during irradiation. The capsule, originally containing sintered UO_2 pellets (1.25-inch OD) operated for a very short time with a maximum surface heat flux of 550,000 Btu/hr-ft^2.

Release of Sorbed Gases by Ionizing Radiation. Gamma irradiations of UO_2 specimens with a Co-60 source (8.9 x 10^5 R/hr) at room temperature revealed a surprisingly high release of sorbed gases. Specimens were sealed in pyrex capsules under vacuum or under helium at a pressure of approximately 0.3 atm. One hundred to two hundred hour exposures caused a release of sorbed gases comparable to that normally expected from thermal effects at temperatures of 800-1000 C. Mass spectrometric analyses of released gases revealed a high percentage of H_2, with N_2, CO_2, and H_2O also present. These data indicate that release of sorbed gases in fuel elements may be increased by reactor radiation. It is also possible that ionizing radiations might aid removal of sorbed gases from ceramic fuel materials before in-reactor use. It was also attempted to accelerate the release of gas with a high frequency (400 KC) field, but the results were negative.

Determinations of Pore Size Distribution in UO_2. The angle of contact between mercury and UO_2 has been determined by capillary depression techniques to be 111 degrees. This measurement is important because mercury porosimetry techniques are commonly used to determine the pore size distribution in UO_2. In the calculation of pore size the contact angle between mercury and UO_2 is generally assumed to be ~ 135 degrees, the same as the angle between mercury and glass. The new value creates a 30 percent correction in the pore size determination.

Fused UO_2 Analyses. Analytical techniques for commercially fused UO_2 are being investigated. Determinations of density, stoichiometry, carbon and nitrogen content were checked against the vendor's values on 15 samples representing 1500 pounds of recently purchased arc fused UO_2. Density, stoichiometry and carbon values were in good agreement. However, nitrogen values often differed by a factor of two. A new dissolution technique for Kjeldahl analysis for nitrogen was devised at Hanford; it will be evaluated by the vendor.

Irradiation of UO_2 Single Crystals. Capsule GEH-14-360, containing single crystal and bicrystal UO_2 cylinders ($\frac{1}{2}$-inch diameter), is being irradiated in the MTR to reveal information about thermal conductivity, fission fragment retention and mobility, and recrystallization phenomena. The irradiation behavior of this material will be compared with that of polycrystalline UO_2. The capsule is transferring approximately 1×10^6 Btu/hr-ft^2 to the coolant.

Irradiation Alteration of UO_2. The effects of fuel atmosphere on void migration rates and on heat transfer through vibrationally compacted, fused UO_2 are being tested by the irradiations of eight capsules (GEH-14-384 through 391). The vibrationally compacted fuel (87.5 percent TD, 0.505-inch diameter) is generating a calculated, maximum surface heat flux of 900,000 Btu/hr-ft^2.

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Thermal Hydraulic Studies

Shutdown Cooling of the PRTR. Studies pertaining to PRTR cooling during a total power outage were continued with emphasis on boiling convection. In this case, if liquid natural convection should fail, a steam void would begin to form in the reactor and would increase until the void filled the upper primary system piping and part of the steam generator tubes. At this time the steam generator would act as a condenser for the steam in the primary system. The condensate would be available to return to the reactor. Forming of the initial void would result in blowing some 80 to 100 cu ft of D_2O from the primary system through the pressurizer pressure relief valves.

Calculations indicate that the potential condensing capacity of the steam generator is far greater than that required to remove fission product decay heat generated in the reactor, even with the steam generator shell at normal operating pressure and temperature. As the steam generator shell is depressurized to allow injection of well water to maintain an adequate heat sink, the water in the shell will cool, providing an even greater condensing capacity. In addition, if the primary system high pressure helium supply were valved off, the cooling capacity of the steam generator would be sufficient to result in depressurization and cooling of the primary system. It was calculated that depressurization of the primary system would follow depressurization of the steam generator shell fairly closely. By the time that steam generator depressurization is completed (about 20 minutes at the emergency rate), pressure in the primary system should be low enough that the diesel well pump could supply make-up water to keep fuel elements covered.

Primary coolant volume loss during the initial blow-out and subsequent "shrinking" due to cooling would reduce the liquid level in the reactor to no lower than 1.5 to 2 feet above the top of the fuel, and no excessive fuel temperatures would occur.

Thus, if there is little or no leakage from the primary system through valves, fittings, etc., proper emergency procedures should allow adequate cooling by boiling. Leakage from the system would complicate matters, raising the possibility of uncovering some of the fuel before the pressure had dropped sufficiently for water to be added from the well pump. In this case venting of the primary system would be required to reduce the pressure more rapidly. For some range of leak sizes, it is doubtful that the present 2-inch vent valve on the pressurizer would provide an adequate depressurization rate.

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A re-examination was made of the emergency procedures prepared previously in HW-73292. It was concluded, tentatively, that with slight modifications these procedures probably would be adequate; the major question being the effect of leaks.

Component Testing and Equipment Development

Mechanical Shim Rods. Fabrication of the Environmental Control Facility has been temporarily discontinued until an estimate can be obtained for completion of the facility. The facility will be completed in time for testing one of the new, first generation, shim rods. Detailed design, procurement, and fabrication of the second generation shim rod has been stopped as a result of insufficient funds.

EDEL-I Renovation. Last month's report stated that EDEL-I pump motor and adjustable speed drive repair had been completed and the unit shipped on September 19, 1962. A later inspection report, however, disclosed that the unit had not been shipped because the inspector discovered that the faces of the motor housing were out of square to the primary axis of the motor and would have to be machined. The vendor has now completed repair of the unit and has shipped the unit without General Electric inspection. The vendor has been notified that the unit will be returned to his factory for the required inspection.

Installation of the deionized water piping system is complete except for the installation of a "Magnetrol" liquid level control switch in the storage tank. Pressure testing of the piping system has disclosed that approximately 50 to 60 percent of the piping connections leak. This piping is a stainless steel, 1-inch nominal pipe size, all screwed system using teflon tape as the thread lubricant and sealant. No explanation is offered at this time as to why so many joints did not seal. A remedial course of action is now being studied. Modifications to the pressure piping have been temporarily stopped due to the shortage of funds. At present, a detailed cost estimate of the work yet to be done is being prepared.

Fretting Corrosion Investigation. A mockup of a short section of PRTR process tube and fuel element has been fabricated. This mockup will be used to determine whether the eddy current technique of measuring relative vibratory motions is stable over wide temperature ranges.

Purchase requisitions for a vibration analyzer, velocity pick-ups, and an oscillographic recorder have been issued.

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Shroud Tube Replacement Mockup. The first phase of this facility, which includes excavating the pit, installing the liner, pit floor and cover, is completed.

Pressure Tube Inlet and Outlet Gas Seals. Tests were run on solid zirconium O-rings supplied by Test Reactor and Auxiliaries Operation. Low leakage rates which ranged between 1.8 and 0.4 liters per hour were recorded.

Three tests were run: (a) zirconium O-ring with normal flange and approximately 0.001-inch clearance between O-ring and tube, (b) zirconium O-ring as above with a stainless steel flange modified by cutting a circumferential groove in the ID, and (c) zirconium O-ring with normal flange and the maximum clearance allowed by reactor tolerances between tube and ring. Tests (a) and (c) were run for six cycles and test (b) for four cycles.

Results for all tests showed low leakage rates with a tendency for leakage to decrease as the number of cycles increased.

All proposed testing of these seals has been completed and a final report is being prepared.

PRTR Rupture Loop . A detailed cost estimate for the work necessary to complete the design tests on the in-reactor test section and the fabrication of the discharge equipment was prepared and submitted to Test Reactor and Auxiliaries Operation.

PRTR Gas Loop. Work has resumed on modifications to the test section nozzle caps with receipt of thermocouples and seals for the gamma heat test assembly.

Hazards Analysis

Reactor kinetics studies of a uniformly enriched PRTR core, consisting entirely of UO_2-PuO_2 fuel elements, have been completed. The lower value of β, 3.26 mk, in the uniformly enriched core, about half the calculated value used for the studies of the spike enriched core (natural UO_2 fuel elements and PuAl fuel elements), is compensated by the stronger negative fuel temperature coefficient which is about twice the value for a spike enrichment core. For all of the accidents studied, the consequences would be mild when the excursion is terminated by a reactor scram. A comparison of the peak power levels reached in similar accidents is given for both cores in the table below:

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Accident	Initial Power Level Mw	Peak Power Level, Mw Uniformly Enriched	Spike* Enriched
Startup	0	158	300
Controller Malfunction	10	92	96
Controller Malfunction	70	92	94
Shim Failure	70	100	92
Experiment Failure	70	130	92

*HW-61236, PRTR Final Safeguards Analysis, October 1, 1959.

Plutonium Recycle Critical Facility. The PRCF Final Safeguards Analysis is being reviewed by the AEC Division of Licensing and Regulation. This review resulted in a number of questions which were answered in document HW-75102, "PRCF, Answers to Safeguards Review Questions."

Work was begun on the formulation of a reactor model to study the kinetics characteristics of a light water moderated core for the PRCF.

All of the planned process specifications for the PRCF were issued for approval.

Materials Development

Impact (Fretting) Corrosion. Evaluation continued at 300 C and in lithiated water at a pH of 10.0 of a prototype spring-loaded support designed to eliminate impact (fretting) corrosion by the PRTR fuel element supports. The test specimen was a rod 1-inch in diameter by 2-feet long held tightly in the test section by spring-loaded supports. This arrangement was tested in the TF-2 apparatus for 625 hours and then removed for examination. No impact or fretting corrosion was observed. Apparently the spring arrangement had kept the specimen rigidly held in the test section. The test is continuing and will be studied with applied vibration to further evaluate the design. The program is continuing to determine more exact information about the effects of variables on impact corrosion.

In a static test in deionized water at room temperature, a Zircaloy coupon was subjected to impacts of known energy at a frequency of three times per second. No weight loss occurred until after 40 hours. After this induction period the oxide-covered sample began to lose weight in direct proportion to the number of impacts received. This continued for 165 hours when the test was terminated. The weight loss in milligrams was found to be equal to 0.295×10^{-6} times the number of impacts. The 40-hour induction time apparently was caused by the protective oxide formed by prior high temperature operation. Penetration of the Zircaloy-2 coupon at the end of the test was 1.0 mil. Further testing is under way to obtain data at high temperatures to define the parameters which govern the penetration due to the impaction of Zircaloy-2 specimens.

Decontamination of PRTR. The PRTR was decontaminated during October. The entire decontamination, from the introduction of the first solution until completion of flushes after the final solution, took 10 days. The procedure included: 2 buffered-oxalate flushes, 2 cycles of alkaline permanganate and oxalic acid, and one cycle of alkaline permanganate and ammonium citrate.

The buffered oxalate solutions were used to dissolve the sintered PuO_2 and keep it in solution. Previous tests had shown that of all compounds considered, it was the only one which would dissolve the sintered PuO_2 and still not be excessively corrosive to PRTR components. Two formulations were used: one solution buffered with acetate and one buffered with gluconate. The acetic acid was added as peracetic acid to escape the difficulties of foaming, heat of reaction and volume change occurring when acetic acid reacts with peroxide. No difficulties were observed with either the acetate-buffered or gluconate-buffered solutions.

Oxalic acid solutions are not usually recommended for reactor decontamination because a film of ferrous oxalate tends to form and redeposit on the surfaces. As it deposits, it carries with it some of the activities. This effect is especially troublesome if appreciable amounts of carbon steel or 400-series stainless steel are present. This effect was noted in the IRP tests on proprietary oxalic solutions and was brought to the attention of the vendor. The formulation was changed to give a new compound containing a complexing agent which delayed this re-precipitation. This compound was chosen for decontamination of PRTR because previous tests had shown that of all compounds available it was the only one which would remove film and contamination from PRTR components. In laboratory and pilot plant (IRP) tests, the improved oxalic solution did not cause films to form and redeposit on the stainless steel. However, in the PRTR decontamination, the solutions were made in advance since the cycles could not be scheduled too tightly; moreover, it required a long time to fill, heat, circulate, and drain the PRTR. Consequently, the ferrous oxalate films did form and precipitate on the reactor surfaces, thus resulting in lower decontamination factors.

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It was necessary to add an additional step to remove these films. Some tests were run in the laboratory using spiders from PRTR to determine the best procedure for removing the films and re-adsorbed activity. Four reagents were tried: dibasic ammonium citrate, a proprietary phosphoric acid compound, nitric acid, and a proprietary bisulfate compound. Of these four reagents, the ammonium citrate was most efficient, giving decontamination factors ranging from four to eight and removing the films. Further testing showed that if the ammonium citrate alkaline were preceded by alkaline permanganate, even higher decontamination factors could be obtained along with complete removal of all film. On the basis of these results, a two-step procedure, alkaline permanganate followed by ammonium citrate, was used to remove the oxalate film.

During the laboratory tests, a small amount of precipitate formed and remained partially suspended. This precipitate was quite radioactive showing adsorption or co-precipitation of a large portion of the activities. Since this precipitate would also form in the reactors, it appeared advisable to put a full flow filter in the system. One method suggested was to place glass wool in the thumb screens. This appeared to be of questionable value since there was no good method for keeping the glass wool in place. As a test, during the rinses following the APACE procedure, glass wool was placed in the thumb screen in one tube. This filter picked up considerable activity, increasing from 40 mr to 120 mr (as read from outside the tube) in a relatively short interval. When the thumb screen was removed, it was noted that the glass wool had collected some precipitate (highly active). However, a considerable portion of the glass wool had escaped, so it was decided this procedure could not be used. Instead, wool cartridge filters were placed in twelve of the tubes. All twelve filters picked up activity.

A complete series of radioactivity surveys was made before, during and after the decontamination. The readings will be used to assess the degree of decontamination. In some places, the decontamination factor was quite high (~ 9); in others, such as the inlet to the lower ring header, the decontamination factor was low. The lower decontamination factors were observed in those parts of the reactor where the geometry was such that precipitates or crud would be trapped. This emphasizes the importance of providing full-flow filters during some part of the decontamination process.

The data are being analyzed and a more complete report on the decontamination efficiency will be reported soon. From preliminary observations, it appears that the inner surfaces were bright and free from all film.

Corrosion of Zirconium and Zircaloy in D_2O. The corrosion rates of Zircaloy-2, Zircaloy-4, and crystal bar zirconium (all alpha annealed) have been measured for 57 days in static 400 C, 1500 psi D_2O steam. The purity of the D_2O was 99 percent. The corrosion rates and D_2 pickups (expressed as H_2 equivalent) are the same as have been measured for these metals in H_2O. The Zircaloy-4 and Zircaloy-2 reached transition in about 41 days as is typical of H_2O corrosion. The crystal bar has not reached transition at 57 days. The Zircaloy-2 D_2 pickup fractions are about 30% compared with 15% for the Zircaloy-4 and crystal bar. This is in contrast to Canadian work where a lower hydrogen absorption was found in D_2O. This test is continuing to measure post-transition rates.

Properties of Irradiated PRTR Process Tubes. Tube 5679 was removed from channel 1643 on May 2, 1962 (369 days after installation). Its exposure was about 3×10^{20} nvt (E>1 mev). The tube was removed because a mark about eleven feet from the top was observed during in-reactor tube examination that could not be attributed to end bracket or bundle or rod wrap effects. The tube was cut into two-foot lengths and the two-foot piece (4C) containing this mark was burst at a temperature of about 280 C and a pressure of 4650 psig. Visual examination of the burst piece showed that the flaw did not affect the fracture characteristics of the tube.

Metallographic examination of the flaw is being performed. The kinescope recorder have been found to affect the TV picture linearity. Magnetic shielding is being investigated to eliminate this pickup from the synchros. A new light mount for the Omniscope incorporating air cooling to prevent overheating of the lenses and permitting use of the new dial gage for measuring the defect depth has been designed and is ready for fabrication.

2. PLUTONIUM UTILIZATION STUDIES

Plutonium Carbide

Several x-ray capillaries of PuC which had been loaded as long ago as 14 months were rerun in the Debye camera to study a lattice expansion with storage which had been noticed previously. All of these samples show a continued growth which is much greater than the experimental error of the technique. It seems quite likely that the mechanism for this growth is self-induced alpha damage. There seems to be an increase in the growth rate with time although there are not enough data points yet to confirm this. The cumulative rates are as shown below. The rates normalized to 100 days do not indicate any significant trend with composition.

Atom Percent Carbon	$\frac{\Delta a}{a}$	Total Time (Days)	$\Delta a/a/100$ Days
29.1	1.93×10^{-3}	426	0.46×10^{-3}
33.0	1.13×10^{-3}	279	0.41×10^{-3}
35.4	2.27×10^{-3}	426	0.53×10^{-3}
48.2	1.33×10^{-3}	273	0.49×10^{-3}
48.2	1.51×10^{-3}	273	0.55×10^{-3}

Plutonium Nitride

One hundred milligrams of plutonium nitride were placed in 10 milliliters of each of five concentrated mineral acids, HCl, HF, HNO_3, H_3PO_4, and H_2SO_4. Complete and immediate dissolution was observed in hydrochloric and phosphoric acids, giving a clear green and a clear blue solution, respectively. The temperature prior to mixing was 32 C. Reaction products were identified by color only. The blue and green colors are attributable to the Pu^{+3} ion. No immediate reactions were seen in hydrofluoric, nitric, or sulphuric acids. After 30 minutes in hydrofluoric acid, bubbles formed on the plutonium nitride; 24 hours later a clear chalky blue solution above a fine blue powder was observed. The 24-hour observation of the nitric acid solution showed a clear straw

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colored solution above a black powder. The sulphuric acid solution after 24 hours also was straw colored, but the sediment was blue.

Plutonium Sulfides

The melting point of Pu_2S_3 was measured in vacuo (approximately 10 microns) on a tungsten ribbon and found to be 1725 ± 5 C. X-ray analysis of the melting point sample showed Pu_2S_3 lines only. Heating to 2000 C in the same vacuo resulted in the formation of a weak phase identified as Pu_2O_2S. With an argon atmosphere, the Pu_2S_3 was stable to 2300 C, the limit of the tungsten ribbon furnace.

X-ray analysis of corrosion samples exposed to boiling demineralized water showed no signs of chemical instability.

3. UO_2 FUELS RESEARCH

Low Temperature Sintering of Swaged UO_2 During Irradiation

Ceramographic examination of cold swaged PRTR fuel revealed that sintering had taken place during irradiation at remarkably low bulk UO_2 temperatures (300-400 C). This low temperature bonding between fuel particles accounts for thermal conductivity increases noted during irradiation of a thermocoupled, 19-rod element in the PRTR. In earlier reports it was hypothesized that stressed point contacts between fuel particles (resulting from the swaging process) were bonded during fuel irradiation, presumably as a result of localized, transient, high temperatures in the regions of individual fission events.

Because much of the volume of PRTR fuel element core material operates at relatively low temperatures, the observation of a sintered structure also has important implications with respect to erosion resistance of the UO_2 in the event of fuel rod cladding failure.

Cladding Studies

A comparison of welds made using low voltage (30 Kv) and high voltage (150 Kv) electron beam welding equipment was made on four joint designs (tangential and "burn-through") used in Zircaloy cladding fabrication. A satisfactory tangential weld between two parallel tubes could not be made with the low voltage equipment. Heat input to the work was, in all cases, much greater during low

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voltage welding, and the sizes of the heat affected zones were excessive. It is concluded that the high voltage electron beam welding process is more suitable for Zircaloy fuel element cladding fabrication.

Transverse Vibrational Compaction Studies

A new method of vibrational compaction involving transverse excitation of cladding suspended vertically from a horizontal bar may simplify fabrication of fuel elements, particularly those containing plutonium-bearing materials and those clad in SAP or thin wall stainless steel.

Transverse excitation simplifies coupling of applied vibrational energy through the wall of a shielded facility to the fuel element cladding. It also minimizes stresses in the cladding at the point of coupling. For example, an eight-foot long UO_2 rod clad in SAP (0.551-inch OD, 0.122 inch wall thickness) withstood prolonged transverse vibration at accelerations greater than 50 G's.

A 5000-pound force vibrator was modified for horizontal operation. A horizontal steel vibration coupling bar attached to the vibrator projects into the remote fabrication cell and is clamped to the upper end of the vertically positioned fuel rod. Compaction efficiencies of 91 percent (equivalent to that obtained by axial vibration) were achieved with fused UO_2 in eight-foot long, 0.505-inch ID Zircaloy tubes.

Hot Vibrational Compaction Studies

Hot vibrational compaction of fuels clad in aluminum alloys or stainless steel may increase the bulk fuel density and reduce fabrication time. A rapid density increase of 2.5 percent TD was obtained by resistance heating a previously vibrationally-compacted, SAP-clad UO_2 rod to 150 C during transverse vibrational compaction. No distortion of the cladding was apparent after cooling. Resistance heating Zircaloy clad UO_2 rods to 850-900 C during transverse vibrational compaction caused only a minor increase in density. The ability to achieve higher density by elevated temperature vibrational compaction is directly related to the thermal expansion characteristics of the cladding and core material.

4. BASIC SWELLING PROGRAM

Irradiation Program

One general swelling capsule has been discharged this month after reaching its goal exposure. The control temperature on the fissionable specimens in this capsule was 575 C regardless of reactor operating conditions. One additional capsule has been charged and its control temperature is 525 C during full power reactor operation. However, it has been established that the maximum possible temperature for this capsule when the reactor is down will be approximately 365 C. This capsule contains a single 1/16-inch OD heating element instead of a single 1/8-inch OD element. As previously reported, a new order for 1/8-inch heating elements has been placed after cancellation of a previous order because of the failure of the assigned manufacturer to produce satisfactory elements. Meanwhile, one additional capsule has been assembled and laboratory tested, and two more capsules are partially complete. Some difficulties experienced with the syphons in the latter two capsules are delaying their completion. The above three capsules contain 1/8-inch developmental heating elements. Capsule component parts are being fabricated off-site for future general swelling capsules. The re-worked instrumentation, electrical wiring and saturable core reactors are being installed in the capsule instrument room at the reactor. The digital read-out system has arrived and replacement thermocouple extension wire will arrive shortly. These monitoring and control units will be checked out with a prototype capsule before being used on capsules being irradiated. Two previously irradiated capsules (13 and 14) are being disassembled for specimen recovery in Radiometallurgy.

Post-Irradiation Examination

The density determinations have been completed on the uranium specimens removed from capsules 11 and 12, irradiated at 400 C and 625 C, respectively, to a burnup of 0.16 a/o. The physical condition of these samples has been described in previous monthly reports. The density of specimens in capsule 11 decreased from an initial value of 18.96 g/cc to a value of $\sim$ 12.9 g/cc. The density change in specimens in capsule 12, on the other hand, was less, namely, from an initial value of 18.96 g/cc to a final value of $\sim$ 16.3 g/cc.

It is somewhat surprising that the samples from the lower temperature irradiation exhibited the greater density loss. Roughening of specimen surfaces due to irradiation growth may have introduced errors in the density measurement. The operating temperature was such that an appreciable amount of internal tearing may also have occurred, in which case the values indicated are accurate representations of the volume changes incurred. These samples are being processed for metallography to determine their internal condition. The density values of the samples from capsule 12 agree fairly well with the metallographic observations of porosity that were made on these samples.

5. IRRADIATION DAMAGE TO REACTOR METALS

Alloy Selection

A nickel, iron, chromium alloy, R-27, produced by Allegheny Ludlum Steel Corporation is presently being studied to determine its suitability for nuclear application. A cursory examination of the effects of various oxidizing environments and irradiation at high temperatures is now in progress. Three oxidation tests have now been completed, using CO_2 at 1700 F, 1800 F, and 1900 F. No catastrophic oxidation has been found under these conditions and tests indicate that the R-27 alloy withstands these conditions as well as the austenitic stainless steels.

Tensile tests performed on the R-27 heat treated by both single and double aging indicate only slight differences in strength properties. Similar specimens will be irradiated in the ETR hot water loop facility.

Sheets of three additional alloys -- Hastelloy N, Haynes R-41, and Haynes R-235 -- have been hot rolled. Tensile, oxidation, and corrosion specimens have also been fabricated. After heat treating tensile specimens will be scheduled for irradiation in the ETR hot water loop facility. Oxidation tests in CO_2 at 1700 F on Haynes R-41 alloy have also been initiated.

Hanford Laboratories is to be responsible for the procurement, storage, and disbursement of structural materials to be used in the Irradiation Effects on Reactor Structural Materials Program where use of a material is by more than one site. Design of suitable storage racks and procurement of handling equipment has been initiated. In addition, specifications for A302B, A212B, 304 SS, 347 SS, Inconel 600, Inconel X-750, and AM-355 alloy have been prepared. The A302B and A212B pressure vessel steel has been ordered in the form of four-inch plate.

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In 1959, the United States Steel Corporation produced standard documented heats of A212B, A302B, HY 80, and T-1 steels for testing purposes. Three square feet of four-inch A212B, four square feet of six-inch A302B, and 20 square feet of three-inch HY 80 plate have been obtained from these standard heats for the Irradiation Effects on Reactor Structural Materials Program.

In-Reactor Measurement of Mechanical Properties

Another significant data point in the spectrum of activation energies controlling creep of cold worked Zircaloy-2 was obtained during the month. This point was obtained on an irradiated specimen during a reactor outage when a value of 55,800 cal/mole was measured for the 20 percent cold worked Zircaloy-2 specimen at 350 C. A value of 85,000 cal/mole had been observed at this temperature during reactor operation. The difference between the two values is apparently related to radiation induced defects which anneal rapidly after the neutron irradiation is discontinued and the temperature is maintained at 350 C. The activation energy obtained during the reactor outage was measured by the same temperature cycle method used to determine the activation energies for creep during irradiation; the creep rate was measured at 350 C and the temperature then abruptly changed by 20 degrees to 370 C and the new creep rate measured. The activation energy was calculated from the creep rates just before and just after the abrupt change in temperature. The value obtained, 55800 cal/mole, is in excellent agreement with the ex-reactor activation energy which is about 58,500 cal/mole. The reduction in activation energy when the neutrons are shut off is consistent with observed increases in creep rate shortly after reactor shutdowns and also consistent with the theory of in-reactor creep behavior which was described last month.

The in-reactor creep test on 20 percent cold worked Zircaloy-2 at the test conditions of 350 C and 30,000 psi stress was discontinued at 340 hours as the specimen went into third stage creep. This test was described in last month's report to the time the specimen had accumulated 320 hours at test conditions. In another 20 hours the specimen went into third stage and the test was stopped to permit the continuation of activation energy measurements before the specimen fractured. The total plastic creep strain at the time the test was discontinued was slightly over two percent.

Calibrational procedures have been achieved that can correct for the apparent strain exhibited by the thermal expansion change of the extensometers in the capsule during a reactor outage. Previously, the temperature change during the time the reactor went up or down resulted in a change in the thermal expansion of the micropositioner in the capsule and produced an apparent strain which masked the true strain. The change in temperature resulted from a change in gamma heat between the two reactor operating conditions. This apparent strain could not be corrected from the data until the test was completed, sometimes several months after the first data were obtained. The corrections for the apparent strains exhibited became possible when the correlations between total power to the heaters and the gamma heat were made. A plot of heater voltage versus apparent strain was constructed and used as a correction chart during reactor transients. The creep data during transients when corrected in this manner represents a smooth curve which tends to confirm the validity of the correction. Additional correlations during startups can be made after the creep test is terminated to verify the original correction data. In this manner creep curves can be obtained that show the actual strain without the apparent strain during reactor outages.

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

A data processing program has been completed that will provide an exceptionally versatile and fully automatic method of analyzing calibration data of any type for the in-reactor measurements program. The program is filed with the Special Procedures Library under the name TCP. The program is capable of processing up to 20 runs of a given group. Each run may have up to 25 observations for each of seven variables. The variables may be either positive or negative, in ascending or descending order, and missing points are tested and ignored, rather than being treated as zeros. The output is a set of coefficients for the calibration polynomial, calculated for calibration curves up to and including the third order equation. These polynomials are generated for any pair, or number of pairs, of pre-selected input variables from each run and for each group of runs collectively. In other words, a polynomial is generated which relates a base variable to any, or all, of the other variables for each run. All the runs are then lumped together and a complete composite polynomial is generated for the entire group of runs. At the time of lumping together, each run is tested and if it falls outside of certain preset limits, due to one or more bad data points, this run is omitted and identified as being omitted. The program also statistically evaluates each run and each composite group and determines the standard deviation, the reproducibility at the 95 percent confidence level, the zero reset range, and the over-all precision at the 95 percent confidence level, for each generated polynomial. The program is useful in preparing each day's data points from an in-reactor capsule to fit the composite curve indicating the creep behavior of the specimen. The variables affecting the data are handled separately and tested to be within confidence limits before the point is used in the curve. It is expected that this program, due to its versatility, will be useful in evaluating calibration data of many forms. As many as seven interrelated variables can be equated to produce the polynomial describing the process. Missing points are ignored and off-limit points are rejected.

Irradiation Effects in Structural Materials

The purpose of this program is to investigate the combined effects of irradiation 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 temperature.

During the month 85 Zircaloy-2 specimens were tested at room temperature. The major portion of these specimens were bend test specimens which had been irradiated in the ETR G-7 hot water loop. Control bend test specimens were also tested; some were instrumented with strain gages or engraved using a photographic technique in an attempt to measure the strain on the tensile surface of the bend specimens. The specimens tested represent four levels of cold work (0, 10, 20, and 40 percent) and were machined from both the transverse and longitudinal direction with respect to rolling.

The geometry of the bend test specimens make them ideal for determining weight and hardness changes due to irradiation and environment. Both weight gain and hardness measurements were made on each specimen prior to testing. Specimens which are subjected to different fluxes but receive approximately the same total dose are expected to show some difference in weight gain due to flux rate effects. Preliminary data tend to confirm this observation.

In addition to the bend tests, 26 Zircaloy-2 tensile specimens were also tested. These tensile specimens were exposed in the ex-reactor hot water loop which duplicates the ETR loop history exclusive of the neutron irradiation. Comparison of data obtained from specimens exposed in both the ETR and ex-reactor loops provides a means of separating environment effects from changes produced by neutron irradiation. The data from the tensile tests is being processed by computer.

Notch tensile tests on plate specimens of Zircaloy-2 containing a 45 percent notch depth, a 60 degree notch flank angle, and a notch root radius of less than 0.001 inch have been made. They verify that in the unirradiated condition the criteria for failure of Zircaloy-2 would be gross plastic deformation prior to separation by fast crack propagation for the geometry and stress state existing in this case. There is a tendency for reduced ductility and a lower ultimate stress to yield stress ratio at higher levels of cold work and higher strain rates, as would be expected.

A shift from failure by yielding to failure by fracture for cold worked Zircaloy-2 was observed after irradiation at 540 F to 6×10^{19} nvt. A notched tensile specimen containing 40 percent cold work failed in a brittle manner with relatively little plastic strain. Specimens containing lesser amounts of cold work at this exposure level exhibited somewhat greater amounts of plastic strain prior to failure. These observations demonstrate the importance of fracture mechanics studies of Zircaloy-2 in both the unirradiated and irradiated states.

A set of grips for remote tensile testing have been fabricated and checked for axial alignment. These grips are to be used in conjunction with an Instron testing mechine now located in "I" cell in Radiometallurgy Laboratory.

Damage Mechanisms

The objective of this program is to establish the nature of the interaction between defects present prior to irradiation and those produced by radiation. The investigation is presently concerned with high purity iron and its low carbon and nitrogen alloys.

During this period several one-fourth-inch rods of Ferrovac E have been zone refined up to four passes with the electron beam unit. Vaccum and induction heating equipment is under design and construction for zone refinement of large diameter stock.

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Dies have been received for cold drawing the high purity iron and experiments to determine the heat treating parameters for grain szie control will be initiated upon receipt of the cold drawn stock.

Construction and modification of the resistivity apparatus neared completion during the month.

Foils of high purity Johnson-Matthey iron have been successfully thinned for transmission electron microscope studies. The specimens, 0.003-inch thick, were electrolytically polished in a bath composed of ten parts glacial acetic acid and one part perchloric acid (60 percent) at a potential of 42 volts. The thinned sections showed many dislocations despite a two-hour anneal at 600 C. Also present were a few foreign particles, the origin of which is unknown at this time. It is hoped that polishing conditions can be further refined to eliminate the mottled structure which is produced by the above polishing conditions.

Irradiation Damage to Inconel

The DR-1 gas loop consisting of an Inconel outer tube and a finned stainless steel inner tube has been removed from the reactor. The in-reactor section was removed by four men pulling on a cable attached to a fitting welded to the end of the outer tube. This relatively light stress was enough to separate the Inconel tube near the center, leaving the re-entry inner section in the reactor. The break was nearly square, almost as though the tube had been separated by a saw cut. Both the Inconel and the stainless steel tubes will be thoroughly examined in the Radiometallurgy Facility to determine the effects on these materials of the integrated flux of about 10^{21} nvt ($E > 1$ Mev).

A possible means of measuring the neutron exposure to which a specimen or irradiated Zircaloy-2 was exposed is being explored. This method involves counting the Mn-54 activity arising from transmutation of the original iron content of the alloy. Specimens from a Zircaloy-2 pressure tube exposed to three different flux levels are being tested.

Experimental fabrication has started on TD Nickel, a new high temperature (up to 2200 F) dispersion hardened alloy containing 98 percent nickel and 2 percent thoria. With a preheat at 800 F, the metal rolled with no difficulty from one inch and one-half inch bar stock to 0.030-inch thick strip. The warm-rolled TD nickel had a hardness of about 99 R_B, or essentially the same as

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warm or cold rolled pure nickel. However, when annealed, the dispersion hardened TD nickel softens to only about 90 R_B, whereas the pure nickel softens to about 50 R_B.

Burst Properties at Elevated Temperature - PRTR Pressure Tubes

After improving the temperature control to yield a maximum differential of 5 C over the length of the specimen, two irradiated PRTR pressure tubes were burst at elevated temperature. Data on the elevated temperature burst tests are shown in the table below:

Elevated Temperature Burst Strength
Of Irradiated Zircaloy-2 PRTR Pressure Tubes

Tube No.	Condition of Tube, % Cold Work	Ultimate Stress, psig	Nvt ($E > 1$ Mev) (Calculated)	Temp., C	Strength Increase Over Unirradiated, %
Control	45	72,500	--	297	--
5702-56	45	74,000	1×10^{17}	296	2
5540-6B	Annealed	56,900	3×10^{17}	295	24
6061-2B	Annealed	58,200	1×10^{20}	297	28
5679-4C	Annealed	62,000	3×10^{20}	286	32

The section of the PRTR pressure tube that was annealed prior to installation shows an increase in strength with increasing exposure for the elevated temperature tests. The elevated temperature ultimate strength has increased 32 percent over the unirradiated in comparison to a maximum increase of 14 percent for similar tests run at room temperature.

During the elevated temperature burst testing, of the three irradiated specimens from the annealed section, the pressure continued to increase with time up to a maximum, leveled out for a short time, then dropped abruptly about 200-300 psig. The slope of the pressure versus time curve then became negative, indicating a slowly decreasing pressure to the point where the tube ruptured. This abrupt pressure drop appears to be similar to the drop in load noted on preformed, irradiated tensile specimens. The pressure versus time curve for unirradiated samples shows a continually increasing pressure until the slope begins to decrease when the material starts to yield. The pressure curve on the irradiated cold-worked specimen did not change from that of the unirradiated sample.

6. GAS GRAPHITE STUDIES

Additives in Graphite

Preliminary results have been obtained from hot capsule irradiations of graphites containing additives supplied by Speer Carbon Company under Contract DDR-118. Duration of the irradiation in position E-14 of the ETR was limited to cycle 47 due to reactor outages. Flux monitor data are not yet available. Irradiation temperature is estimated to be in the range of 500 to 700 C. Each hot capsule contained one standard graphite sample containing no additive and three samples prepared with additive. The largest difference in dimensional changes was noted in a capsule containing transverse samples prepared with increasing levels of Fe_2O_3 addition. The standard sample contracted 0.007 percent, whereas growths of 0.022, 0.024 and C.016 were noted for samples containing 2, 4, and 5 percent Fe_2O_3 addition, respectively. Of the additives other than iron, Cr_2O_3 produced the largest effect: 0.015 percent contraction for the sample with Cr_2O_3 addition versus 0.032 percent contraction for the standard in the same capsule. With the exception of one capsule in which the standard sample was broken during disassembly, all additive systems apparently reduced the transverse contraction to some degree. The effects were less pronounced and sometimes reversed in parallel samples.

Investigation of Tensile Creep in EGCR Graphite

The first tensile creep capsule, GEH-13-9, was irradiated in ETR during cycle 47. The capsule contained a sample of EGCR graphite under a tensile load of 800 psi. The goal temperature for the sample was 1000 F, to be achieved by gamma heating of the components. The atmosphere in the capsule was kept at a constant helium pressure and the tensile load maintained by a helium-pressurized bellows.

During early operation the temperature profile of the sample varied from 700 F to 850 F. After a week of operation the thermocouples started to fail and by mid-cycle no temperature read-out was available. Because of the lack of temperature instrumentation, the experiment was removed at the end of the cycle. During disassembly it was found that the thermocouples had broken due to chemical reaction. It is possible that this was due to excess flux that had not been removed during formation of the couple bead.

The sample length was measured and it was found that the tensile specimen and the unstressed control specimen had both contracted 0.037 percent. The exposure was estimated to be 7×10^{20} nvt

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(E > 0.18 Mev). This contraction compares well with data given in HW-71500 A. The data obtained from the test indicate that the basic capsule design was sound.

Several mechanisms can be postulated to account for the contraction observed in the stressed specimen. The control specimen has a smaller diameter and runs at a higher temperature (approximately 100 F) than the tensile specimen. In this temperature range there is less contraction noted at the higher temperatures. There is, too, some preliminary evidence that small samples contract less than do large ones. Therefore, it is possible that the intrinsic contraction of the specimen under tension may have been less than the control specimen due to a strain-release mechanism under load. This, however, is an unsubstantiated preliminary conclusion.

The second capsule, GEH-13-91, is installed in the ETR and is functioning well. This capsule is scheduled for three cycles and should provide a more reliable measure of any radiation-induced deformation that may exist at this stress and temperature.

EGCR Graphite Irradiation

The fifth capsule, H-3-5, in the series of irradiations of EGCR graphite has operated satisfactorily for one cycle in the GETR. All thermocouples are functioning properly. Sample temperatures match those for the first four H-3 capsules.

Data from the H-3-4 samples are being analyzed. A delay in receipt of flux-monitor results due to unavailability of the 7090 computer has held up the final exposure determinations for these samples.

Flux Intensity Test

The graphite irradiation capsule, GEH-13-8, designed to study the effect of flux intensity on property changes at a controlled temperature was removed from the ETR on August 9. The samples received an exposure of 87.8 effective full power days. Sample temperatures of the four positions having heaters were controlled at 650 C. Charts of the sample temperatures at the three remaining positions are being analyzed.

The capsule was successfully disassembled in the hot cell during the week of September 17. The general appearance of the as-opened capsule was excellent. A slight amount of carbon deposit was observed on the inner shell wall between positions 6 and 7. There

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was no visible damage to the samples, all of which were recovered. All the pyrolytic graphite insulators were also recovered. Three of these insulators were found to have partially delaminated. All three were pieces adjacent to their respective aluminum base plates and consequently had the highest temperature gradients across them. The insulator from position 1 exhibited the greatest delamination, but still did not cleave apart.

Removal of the flux monitors and the vials containing diamond samples was greatly facilitated by the new base plate lid. The monitors were positively identified as to location in the base plate and all were recovered. All items from the capsule have been returned to HAPO and measurements on them are in progress. Preliminary optical examination of the diamonds has been completed. All three types, two types of synthetic and one of natural diamonds, turned black under the irradiation and are completely opaque. Further examinations are continuing.

Gas Phase Inhibitor Studies

A study is being made on the value of using Freon 12 (dichlorodifluoromethane) as an inhibitor of graphite oxidation in dry air. Weight-loss rates as a function of temperature are being determined. The flame from a graphite sample that ignited between 800 and 1000 C in pure oxygen flowing at 2.5 SCFH was extinguished by approximately 3 percent Freon 12 within 5 seconds. Between 500 and 650 C, the addition of approximately 2.5 percent Freon 12 apparently increased the rate of oxidation. This result is not understood as yet but has been reproduced in several tests. One typical result at 620 C is shown below:

Gas	Weight-loss Rate, g/g/hr
Dry air	9.5×10^{-3}
Dry air + ~ 2.5% Freon 12	5.2×10^{-2}

Infrared analyses were made of the exhaust gases, both with and without a graphite sample present, to determine the effect of temperature on the Freon 12. With graphite in the system CO was produced and the absorption peak due to CO_2 increased; other absorption peaks were unchanged. Below 650 C only the Freon 12 spectrum was observed. As the temperature was increased, the Freon 12 disappeared until at approximately 925 C it was no longer present in the spectra. SiF_4 (absorbing at ~ 1030 cm^{-1}) was observed to increase in proportion to the loss of Freon 12. A peak observed at 1215 cm^{-1}, identified

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as $CClF_3$, appeared at 650 C, went through a maximum at 857 C and was essentially gone at 1000 C.

These results suggest that the inhibition observed at higher temperatures is due to the decomposition of the Freon 12. This decomposition liberates chlorine and fluorine. It is well established that Cl_2 acts as an inhibitor. The fluorine attacks the quartz to give the SiF_4.

Surface Measurements on Cellulose Carbon

Two attempts were made to measure the surface area of a 3-g tubular sample of an impermeable carbon prepared by direct carbonization of a cellulose material by the British General Electric Company. The measurements were performed by the standard technique of nitrogen adsorption after outgassing the sample for three hours at 400 C and a pressure of 10^{-6} cm Hg. It was found that the surface area of the sample is too small to measure with the equipment normally employed for similar measurements on ordinary nuclear graphite.

An attempt was also made to force nitrogen into any pores that might be accessible to the gas. However, no measurable condensation occurred in the pores. This is the result expected on a graphite with a non-connected pore system.

7. GRAPHITE RADIATION DAMAGE STUDIES

Graphite Irradiations at 650 C

The effect of wide variation of filler materials used with a conventional coal-tar-pitch binder is being studied in K-Reactor irradiations at approximately 650 C. The samples are from graphites prepared by Battelle Memorial Institute at the time when low-temperature radiation effects were under extensive study. All samples were formed by molding. After an exposure of 5200 Mwd/At_K (1.2 x nvt, E > 0.18 Mev), the following dimensional changes were noted:

Filler Material	Final Processing Temp., C	Length Change, %	
		Transverse	Parallel
Coal-tar-pitch coke	2570	-0.09	-0.12
Thermax carbon black	2570	-0.17	-0.19
Fluid coke	2570	-0.17	--
Korite petroleum-asphalt coke	2800	-0.77	-0.78

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Irradiation of these materials at 30 C resulted in growths interrelated in the same manner noted above; the coal-tar-pitch coke graphite grew at a high rate, Thermax and fluid coke graphites grew at intermediate rates, and the Korite coke graphite grew at a very low rate.

These results agree with previous observations of reactor-grade graphites that indicated lower growth rates occur in room temperature irradiations for those graphites that contract at high rates in high temperature irradiations. It is also evident that the contribution of filler material to contraction can mask the beneficial effect of a binder such as coal-tar pitch that is known to have good dimensional stability. Contraction of the Korite material was significantly higher than that of carbon black which has previously been considered the least stable in high temperature irradiations. This series of samples has been recharged for further exposure.

Irradiation of Lampblack-based Graphite

In a search for evidence of saturation in high-temperature, radiation-induced contraction, samples of lampblack heat treated to 1400 C have been irradiated in the ETR. Irradiations were conducted in non-instrumented capsules in which the samples are estimated to have been heated to 500 to 700 C by gamma irradiation. A contraction of 3.0 percent has been accumulated after a total exposure of 1.6×10^{21} nvt ($E > 0.18$ Mev). After the first irradiation of 0.9×10^{21} nvt, the sample had contracted 2.2 percent. After the second irradiation of 0.7×10^{21} nvt, however, the sample contracted only an additional 0.8 percent. Hence, the rate of contraction was much less during the second irradiation. Part of this lower rate may be due to a slightly higher irradiation temperature during the second irradiation. However, current data on the effect of temperature indicate that this could not account for such a large difference in rate. Hence, a tendency toward saturation for this material is inferred from these data.

National Carbon Company R&D Contract

Samples of ZT-4130, a hot worked material with an apparent density of about 1.95 g/cm^3, were irradiated to a maximum exposure of 5020 Mwd/At (1.2×10^{21} nvt, $E > 0.18$ Mev), at about 600 C. Transverse samples displayed a net expansion of 0.03 percent and parallel samples a contraction of 0.08 percent. The expansion in the transverse direction probably indicates that this material is significantly more stable than most reactor-grade graphites since CSF would

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display a contraction of about 0.07 percent in the transverse direction after this exposure.

8. ALUMINUM CORROSION AND ALLOY DEVELOPMENT

Coupon Testing in H-1 Loop

Following a period of conditioning of the loop with phosphoric acid, the second coupon test was charged in H-1 Loop on October 10. This will continue for three months at high temperature. Coupons of X-8001 and X-8003 aluminum, Zircaloy-2 and stainless steel are being exposed both in and out of flux to water at 290 C, 25 ft/sec velocity, and pH of 4.5 adjusted with phosphoric acid. Several of the in-flux coupons of each type are being subjected to strong beta radiation from rhodium foils of intensity equal or greater to that present at the surface of typical fuel elements. Rhodium is being used in place of silver because it gives higher radiation intensity, has better corrosion resistance, and has much lower residual activity after discharge.

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

Thermal Hydraulic Studies

Thirty-five experimental heat transfer runs were made with a full scale, electrically heated model of a 19-rod fuel bundle constructed from 6.3 feet long Inconel rods, 0.587-inch OD and spaced 0.050-inch apart. This section was built to demonstrate the steam generation characteristics of a 19-rod bundle in the horizontal position and to substantiate burnout points collected from a 19.5-inch long, 50-mil spaced, 19-rod bundle test section. The bundle was instrumented with thermocouples in 14 of the rods and in 10 of the flow channels. Pressure drops were measured along each third of the heated length as well as over the entire test section.

Operation of the test section was divided into four categories of interest, these being (1) isothermal characteristics, (2) pressure drop-flow characteristics, especially during boiling, (3) coolant mixing behavior, and (4) boiling burnout determination. All runs were made at 1200 psig. The isothermal runs established the flow-pressure drop behavior of the section at temperatures of 232 F and 550 F and at mass flow rates of 500,000 to 4,000,000 lbs/hr-sq ft. Pressure drop measurements were made at constant heat fluxes of 200,000; 300,000; 400,000 and 500,000 B/hr-sq ft and over a range of inlet temperatures during step reductions in flow. Boiling burnout indications terminated nine out of eleven of these runs. The mixing behavior as indicated by the coolant channel water thermocouples was studied at several inlet temperatures and flow rates by increasing the power to the section. Inlet temperatures and flow rates were selected so that full power of 2700 KW would just raise the outlet conditions to bulk boiling. Boiling burnout determinations followed the same approach as the mixing studies but with the inlet temperatures and flow rates selected to provide higher outlet qualities. Burnouts were obtained at mass flow rates of from 500,000 to 3,000,000 lbs/hr-sq ft, steam qualities from 25% to 5% and heat fluxes from 175,000 to 500,000 Btu/hr-sq ft. Eleven burnout points were obtained in these runs making a total of 20 for the test section program. Burnout was always obtained on one of the inner seven rods except for two instances at flow rates when it occurred on one of the outer rods 45° from the top of the bundle. This is likely the result of stratification at these low flows.

10. REACTOR AND NUCLEAR SAFETY STUDIES

Advanced Reactor Concept Studies

Fast Supercritical Pressure Power Reactor. Physics calculations have been completed on a reference core design for the PuO_2-UO_2 fueled Fast Supercritical Pressure Power Reactor. The core is 5 feet in diameter and 5 feet high, surrounded by a 15-inch blanket. Calculations on the reference design have shown large positive coolant temperature coefficient of reactivity. A reactivity increase of about 35 percent $\Delta k/k$ was obtained for the transition from full density to supercritical coolant conditions. This is a desirable characteristic for the case of core flooding with cold water from the equilibrium operating condition. A further increase of about 18% $\Delta k/k$ was calculated for the void case, which is highly undesirable for control stability and cases involving power excursions. A similar void effect has been computed for other steam-cooled fast reactor designs. However, the large decrease in reactivity upon flooding was not observed in these designs.

The positive void effect in these systems is attributable to elastic scattering from hydrogen degrading the spectrum sufficiently to (1) increase the capture to fission ratio (α) of Pu-239, (2) increase resonance absorption at lower energies, and (3) decrease fast fission in U-238 and Pu-240. Thus, as the coolant density decreases, the spectrum hardens which increases the neutron economy and, hence, reactivity of the system. In smaller cores the increased neutron leakage upon coolant loss is sufficient to compensate these positive

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components. The coolant coefficient may also be reduced by using D_2O coolant which has a much smaller moderating power or by using U-233 fuel, which has a more nearly constant α in the fast spectrum. However, the economic incentives for the use of H_2O cooling in a fast plutonium breeder are high enough to warrant further core design studies seeking a void effect of less than $1 and identifying the effects on reactor and fuel cycle costs.

To study the effect of in-core water inventory on the void effect, calculations were made for a partially moderated core. This core resembled the thermal spectrum SPPR concept except for very close packing of the fuel elements to limit water inventory to the desired value. A void effect of +3% $\Delta k/k$ was found for this core. This indicated that increasing moderation sufficiently to reach a void effect of < $1 would probably be an unattractive course because of the associated economic penalties of decreased breeding ratio and increased fuel inventory.

Core designs are now being investigated which will enhance the leakage effect by (a) reduction in core volume and (b) assuming larger surface to volume ratios. In addition, the total amount of H_2O and other non-fuel materials are being minimized by revising fuel design parameters to further reduce spectral degradation.

Fuel Re-use. Economic analyses of direct interchange of fuel between thermal and fast reactors lead to the conclusion that fuel cycle cost savings up to 1 mill/kw-hr are possible for the thermal reactors without increasing the fast reactor fuel cycle cost.

Application of Plutonium to Compact Reactors. Some preliminary results have been obtained in the study of plutonium fueling of existing compact reactor concepts. Critical mass calculations performed by Applied Physics during the month furnished a basis for estimating potential reductions in reactor size and weight. For an ORNL boiling potassium reactor concept, the substitution of plutonium for uranium as fuel may provide savings up to 20 percent in the weight of a reactor equipped with shadow shield. In the case of a GE-NMPO gas-cooled reactor, savings of 50 percent or more in reactor core weight are possible, although it may not be feasible from engineering considerations to make such a drastic cut in weight. These preliminary estimates will be refined consistent with the information available on the reactors involved. Studies are continuing on the effects of plutonium substitution in compact reactors; the next reactors to be investigated are the LCRE and SNAP-50.

Plutonium Fueled Spacecraft Reactor. This Mwe reactor concept has been used as a reference system to study the potential for plutonium fuel for spacecraft reactor applications. Work during the month was redirected toward study of plutonium fuel in other systems presently under development and toward summarizing and reporting the work done in this study.

D. RADIATION EFFECTS ON METALS - 5000 PROGRAM

Foil specimens of molybdenum obtained from Johnson-Matthey as well as molybdenum containing three levels of carbon as intentional impurity obtained from Materials Research Corporation were annealed at 1850 C for 0.5 hour and then irradiated at 40 C to $\sim 10^{19}$ nvt (fast). Defects in all of the as-irradiated foils are detected by transmission electron microscopy. The defects present in the as-irradiated state appear to be unaffected by a two-hour, post-irradiation anneal at 575 C. Control unirradiated specimens having identical thermal histories do not show the defect structures. As reported in the previous monthly report, high purity cold worked molybdenum foils showed defects only after a post-irradiation anneal at $\sim$ 500 C. The defects are of two types, namely, distinct location loops similar to those found in quenched metals, and black spots. Density counts of the two defect types are in progress. Similar irradiated specimens will be annealed at higher temperatures and subsequently examined in the electron microscope. No new dislocations formed in the foils as a result of the 10^{19} nvt (fast) irradiation. However, dislocations present in the pre-irradiated foils have become jogged. An attempt to induce dislocation motion in the irradiated 0.003-inch thick foils will be made in order that defect-dislocation interactions can be studied. Foils of molybdenum are currently undergoing irradiation to goal exposures of 10^{20} nvt (fast).

Samples of polycrystalline molybdenum foils of four carbon impurity levels have been examined by x-ray diffraction. The samples were as follows: high purity molybdenum, low carbon (10-30 ppm C), medium carbon (100-200 ppm C), and high carbon (400-500 ppm C). Specimens were available in the pre-irradiated state, irradiated approximately 10^{19} nvt, and annealed for two hours at 575 C after irradiation. The diffraction lines were broadened after irradiation by an amount proportional to the carbon content, the high carbon sample showing an increase in half-height breadth from 0.24 degrees to 0.33 degrees for the (400) reflection. The corresponding change for the high purity sample was from 0.23 degrees to 0.235 degrees. Annealing decreased the line breadth somewhat for all samples.

Lattice parameters also increased with irradiation, the amount of increase being almost linear with carbon content. The lattice parameter decreased during annealing. Apparently, the carbon exists as a separate phase in the unirradiated molybdenum and is forced into a non-equilibrium solution by the high energy irradiation. Annealing permits a partial precipitation of the excess carbon.

A total of 22 single crystal tensile specimens have been examined. The x-ray technique for observing the deformation has been modified to permit faster and more unequivocal measurements. The failed specimen is mounted so that the x-ray beam is parallel to the tensile axis and a succession of pictures is made as the crystal is shortened in predetermined decrements. In this manner a series of back-reflection Laue photographs may be made from the failure point to the relatively undeformed material some distance away.

One crystal, designed 6-2, has been carefully studied by this method. This crystal contained 100-200 ppm C. The specimen axis before testing was [011]. Far from the failure point the material deforms by slip on $(31\bar{4})$ planes in the [111] direction. At the point of failure slip occurs on $(11\bar{2})$ planes. Why the $(31\bar{4})$ plane is the slip plane rather than the more favorably oriented $(11\bar{2})$ is unknown.

A molybdenum polycrystalline specimen which had been annealed at 1050 C for 16 hours and then irradiated at $\sim$ 50 C to 10^{18} nvt (fast) has been subjected to successive isochronal anneals at temperatures which increase in steps of 25 C. After each anneal a series of microhardness measurements, expressed as Diamond Pyramid Hardness, D.P.H., were made on the specimen surface. The hardness was observed to increase from an initial value of 194 to a maximum value of 213 at 175 C, to be essentially constant at a value of 202 between the temperatures of 250 through 550 C, and then to decrease gradually to a value of 182 at a temperature of 750 C. Annealing at higher temperatures is continuing.

Attempts have been made at reducing the diameter of one-eighth-inch diameter molybdenum rod stock to final diameters of 0.020-inch, 0.010-inch, and 0.005-inch by a drawing process. Such wire specimens are to be used for quenching experiments and electrical resistivity measurements. Metallographic studies show that the reduction and annealing schedules used were incorrect. The wire specimens contained longitudinal cracks. Wires must therefore be refabricated by an appropriate schedule.

Testing of single crystals of molybdenum irradiated to $\sim 10^{18}$ nvt (fast) is being scheduled for the coming month. Capsules containing these crystals and polycrystalline specimens will be opened by a remote lathe facility which has just been put into operation in the Radiometallurgy building.

E. CUSTOMER WORK

1. RADIOMETALLURGY EXAMINATIONS

Examination was completed on an enriched split failure from tube #0567-H. No actual water entry point was found. It is believed that the most likely entry was in the spire at the base of the female cap (RM C-402).

Dye penetrant tests disclosed that no leaks were present in the water annuli of the damaged section of a horizontal safety rod from 105-H (RM 458).

Severe groove corrosion was found in the internal annulus of the hole failure from tube #2583-D (RM C408).

2. EQUIPMENT PROJECTS

Project CGH-858 (High Level Utility Cell)

Two operators have been trained in the use and operation of the cell equipment. A radiation check was performed on the cell and three small outside areas showed a higher-than-average reading from a source placed inside the cell. The activity of the source was 500 R/hr at two feet in air. The cell will be ready for operation by the end of October 1962.

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

All castings have been received and installed with the first coat of paint. Modification drawings for the remotization of the rotating-beam fatigue tester are complete, and a work order was issued for the fabrication work. Delivery of the remote plate fatigue tester is expected to be delayed because of revisions.

3. METALLOGRAPHY LABORATORIES

An excellent etch for the thorium-2½% uranium and thorium-2½% uranium-1% zirconium alloys was obtained from a recent literature source. The etch consists of eight volumes of 70% perchloric acid in 130 volumes of ethyl alcohol. It is used electrolytically with a stainless steel cathode at 30 volts for the thorium-uranium alloy and 25 volts for the thorium-uranium-zirconium alloy. Time required varies with the individual sample from two to five seconds. Excellent grain detail is obtained along with retention of inclusions. The interfaces with cladding materials are attacked somewhat but not badly. Diffusion layers are clearly outlined, but the Zircaloy cladding material is left in a rough-etched condition. Difficulties are encountered when a sample still has the outer coextrusion layer of copper still in place. This combination has not yet yielded satisfactorily to an etch.

A spot-welding development program has been undertaken by Coextruded Product Engineering Operation, FPD. Its purpose is to develop techniques for spot welding Zircaloy supports of 0.050-inch thickness onto Zircaloy clad fuel elements. First attempts produced a heavily heat-affected region in the uranium beneath the weld. Later tests have produced smaller welds and smaller heat-affected areas.

Pits in the surface of uranium have been found after chemical milling of the uranium. These pits definitely interfere with brazing or welding to the point of causing rejects.

Sections of type 304 stainless steel from two crossheaders of 105-DR were examined in areas where cracks had been observed on the outside of the pipe. Transgranular stress corrosion was found in these areas. One crack was about 0.110-inch long or nearly half way through the cross section of the pipe. Intergranular stress corrosion cracking was found on one of the Parker fittings attached to the crossheader. Extensive grain boundary carbide precipitation was present in the Parker fitting in this area as the result of welding the fitting to the crossheader.

4. N-REACTOR CHARGING MACHINE

Modifications

Modification of the transfer arm presence indicators has been completed. The indicators now work satisfactorily. Modification of the lower portion of the vertical life jack transmissions was started and includes installing a 1/8-inch thick rubber gasket between the transmission and the bearing block and installing Belville washers on the transmission hold-down studs. This modification is required to allow the transmissions some flexibility so that the ball screws will not be rigidly restrained against lateral motion by the transmissions. It will eliminate any danger of permanently deforming the ball screws when the machine is loaded by the charging forces.

Fabrication of the new limit switch assemblies which control the vertical movement of the machine while loading or unloading

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magazines has been completed. Installation of these assemblies is not yet started.

The rear pressure roller assembly was installed on the charging machine.

The clearance between the transfer arm racks and the transfer arm driving gears was checked and shims added where required.

Testing

Rough drafts of reports on Design Test No. 8, Idler Roller System, and Design Test No. 19, Filtered Water System, have been completed.

The majority of the work, with the exception of the charging cycles, of Design Test No. 16, Magazine Functional Testing, has been completed. A report is being prepared. The portion of Design Test No. 5, Vertical Life Drive, which concerns vertical lift movement while varying load, pressure, and relief valve setting has been completed.

The magazine piston removal equipment has been tested successfully.

5. SPECIAL PLUTONIUM FABRICATIONS

High Exposure Al-Pu Fuel for Physics Program

Thirty percent of the core material for the PRCF light water critical experiment has been extruded and cut to length. The end caps are finished and machining will begin on the tubing as soon as it arrives.

Preparation of High Exposure UO_2-PuO_2 Pellets Fuel for PCTR

Physics experiments require approximately 16,000 pellets ($\frac{1}{2}$-inch diameter, $\frac{1}{2}$-inch long) clad in Zircaloy for measurements in the PCTR. The pellets are to be made from depleted UO_2 containing 0.90 w/o PuO_2. Four kilograms of plutonium metal were oxidized and blended to form a uniform feed material for the pellets. Development work was begun to establish the process to be used in making the pellets.

Fission Product Transient Samples for Phillips

Twenty-four fission product transient samples containing U-235 Al alloy cores have been completed. Preparations for shipment are in progress.

Final machining and extrusion of elements containing U-235 Li-Al and Pu-Li-Al cores is continuing. One coextrusion with an Al - 12 w/o U-233 alloy core has been successfully completed.

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PHYSICS AND INSTRUMENT RESEARCH AND DEVELOPMENT OPERATION

MONTHLY REPORT

OCTOBER 1962

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

N-Reactor Exponential Experiments

The experimental data pertaining to control strengths in the N-reactor have been analyzed. Much of the further analysis required to translate these exponential pile results to "local control" strengths in the N-reactor is complete. The errors on the control strengths are about ± 8 μB. An additional error which is difficult to estimate on the basis of present knowledge is incurred in obtaining the local control strength.

Results are now available on the following types of experiments:

1. Horizontal rod strengths in the mockup pile with water coolant. The strengths were measured with all possible combinations (4) of internal and external control rod coolant.

2. Horizontal rod strength in the flooded (5 different degrees and positions of flooding) mockup pile with water coolant.

3. Horizontal rod strength in the exponential pile as a function of the position of the rod in the pile.

4. Control strength of samarium ball channels in the mockup pile with water coolant--no flooding.

5. Horizontal rod strength in the mockup pile with natural uranium fuel with water coolant.

Results obtained from the measurements show the following: the rod strength is decreased by 6% when the control rod coolant is lost, secondly the rod strength is increased by 7 to 11% when the pile is flooded to varying degrees, and thirdly the strength of the samarium ball channels is less than the strength of the control rod by about 14%.

Angular Distribution of Thermal Neutrons

A directional neutron detector has been designed for use in measuring angular fluxes. A prototype has been fabricated from steel. Gadolinium has been ordered for the operating model. An invention report on the detector has been submitted. Quarterly reports were prepared on the derivation of angular fluxes from radial fluxes and on the scattering cross section matrix derived for calculations of angular fluxes and hardened spectra. Trial runs on an 18-group S-X calculation have been made for $\phi(r,\theta,E)$ with a copper bar in the center of the PCTR.

Code Development

COMPTAP, a program to prepare a data tape, was written and checked out. The program will be used to originate and maintain a composite tape of libraries used by several nuclear codes. The codes will then use subroutine LILLEY to find the appropriate library on the composite tape. LILLEY is in debug.

Instrumentation

A method was developed and reported of using static laboratory data to determine the inherent limitations in the value of the period derived from a logarithmic response amplifier. The method was developed to aid in the evaluation of various NPR nuclear instrument systems.

The final, detailed report for Design Test 1133 for the gamma energy spectrometer prototype of the NPR Fuel Rupture Monitor System is 90% completed, and the detailed appendices on various component tests are done. The final-form prototype was received back from GE-APED after considerable modification.

Specifications were completed for the solid-state multichannel analyzer to be used at the experimental fuels testing loop which is to be installed in the PRTR. Good progress was made on the detailed drawings for the detector assembly, drain facility, and the general sampling port sections.

One of seven NPR beta-gamma, scintillation, solid-state air monitors, fabricated off-site to a Hanford design, was tested and modified to provide easier filter-paper removal, maintenance, and a cooler interior. Following modifications, the unit was used in a long-term test which was successful until the Gast Company air pump suddenly lost its oil; the motor burned up and will be replaced. General performance, including all circuitry operation, was successful during the test period.

Testing was started by GE-APED on the production units of the Source Range and Intermediate Range nuclear instrument systems for NPR. Recommendations were made regarding a number of circuit changes and further testing will definitely be required on the systems. A comprehensive procedure for testing was proposed in an effort to reduce the visitation time to GE-APED by Hanford engineers. The work was done in cooperation with Electrical and Instrumentation Design, CE&UO and with Instrumentation and Electrical Design, IPD.

Investigations and calculations were started regarding a special bore-gauge instrument to be used at K-reactor. In essence, the instrument is to measure the internal diameter of certain process tubes and channels with an accuracy of ± 0.002 inches for internal diameters from 1.800 to 2.600 inches. The work was requested by Irradiation Testing, IPD.

Systems Studies

Additional reactor instrumentation analog study by means of the 11-node reactor model was requested by IPD. As in previous studies, the purpose was to determine, for various instrument trip point settings, how fast the control rods can be withdrawn without causing coolant boiling at the reactor outlet. The study was made at several power levels. The reactor scram was initiated by a signal from an instrument which monitored the rate of rise of reactor power. The study was completed and the results forwarded to IPD.

Methods of simulating various parts of the NPR and its primary loops on the existing analog computers were studied. The system simulation is separated into parts which can be tested independently for scaling or operational problems, thus reducing the time required to set up a comprehensive NPR system simulation if and when additional computing equipment becomes available.

Four pieces of the Bailey control system ordered by NPR Project Operation for testing and training purposes have arrived for checkout by Systems Research Operation.

The existing analog facility is far too small to simulate the entire NPR system. However, the system can be broken up into a number of smaller sub-systems, which can be simulated separately. Since many of the circuits have not been previously tried or tested on the computer, this would seem to be a necessary prerequisite to the eventual simulation of the entire system. A four-node model of the open loop reactor kinetics including an experimental method of producing the effects of control rod movements was devised and is now in the process of being tested on the computer.

Continued assistance was given NPR Project Section in the review of vendor performance of the Integrated Temperature Monitor and Data Logger acceptance tests.

SEPARATIONS

Experiments with Plutonium Solutions

Criticality experiments were continued with plutonium nitrate solutions in a 14-inch diameter sphere. Plutonium concentrations were in the range of 36.5 to 69 g Pu/ℓ with nitric acid molarities ranging from ~ 4.3 to 6.7; the Pu^{240} content of the plutonium was 4.6 w/o.

Criticality data were obtained for the vessel with reflectors of water and concrete, and with a reflector consisting of a six-inch layer of concrete separated from the core by a four-inch air gap.

The data from the current experiments are summarized in Table I.

The experiments with the 10-inch thick spherical concrete reflector were conducted with Pu solutions at several different acid molarities not previously used in the concrete reflected assembly. The data permit a more accurate comparison with the water reflected unit. The results from these experiments verify that a 10-inch thick layer of concrete is a better reflector than water; in the 14-inch sphere reflected with concrete, the critical concentration of Pu is about eight percent less than when the sphere is reflected with water.

The effect of an air gap between the concrete reflector and the core on the criticality of the unit is currently being studied. The results will be of use in evaluating the nuclear safety of in-plant equipment proximate to reflectors. In these experiments a four-inch air gap has been provided between the vessel surface and a six-inch-thick concrete shell reflector.

The four-inch air gap (a width 57% of the sphere radius) between the core and the reflector is seen to increase the critical mass by about 80%, i.e., if the nitrate is held constant at ~ 336 g NO_3/ℓ, the critical concentration in the 14-inch sphere is increased from ~ 38 g Pu/ℓ to 67.9 g Pu/ℓ by the presence of the air gap. These preliminary results indicate this reflector combination (air gap plus concrete) to be about equivalent to a nominal reflector, or to approximately an inch of paraffin. Further criticality data is being obtained for the effect of an air gap between the reflector and the core.

[redacted]

TABLE I

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

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

Experiment Number	Date	Reflector	Pu Conc. (g/l)	Acid Molarity	Sp.Gr.	H_2O (g/l)	Total NO_3 (g/l)	H/Pu Atomic Ratio	Critical Volume (liters)	Critical* Mass (Kg Pu)
1142124	10-3-62	Full Water	63.0	6.64	1.303	760	477	345.4	$21.45^{+.03}_{-.04}$	1.35
1142125	10-4-62	Full Water	58.5	6.51	1.295	770	465	376.0	$21.69^{+.03}_{-.03}$	1.27
1142126	10-5-62	Full Water	53.6	6.27	1.280	770	445	409.3	$22.35^{+.02}_{-.02}$	1.20
1142127	10-8-62	Full Water	52.3	6.47	1.285	775	455	422.9	$22.61^{+.08}_{-.10}$	1.18
1142128	10-9-62	Full Water	47.9	6.70	1.294	775	465	462.9	$23.13^{+.05}_{-.06}$	1.11
1142129	10-10-62	Full Water	46.8	6.62	1.284	770	459	470.6	$23.38^{+.06}_{-.10}$	1.09
1143130	10-16-62	10" Concrete	47.9	6.46	1.281	773	450	460.6	$22.22^{+.06}_{-.09}$	1.06
1143131	10-16-62	10" Concrete	44.8	6.46	1.279	780	447	496.7	$22.84^{+.03}_{-.03}$	1.02
1143132	10-17-62	10" Concrete	43.3	6.37	1.286	791	445	520.1	$23.25^{+.07}_{-.10}$	1.01
1143133	10-18-62	10" Concrete	49.8	4.57	1.225	834	335	466.5	$19.95^{+.03}_{-.04}$	0.99

[redacted]

[redacted]

TABLE I (Cont'd)

Experiment Number	Date	Reflector	Pu Conc. (g/l)	Acid Molarity	Sp.Gr.	H_2O (g/l)	Total NO_3 (g/l)	H/Pu Atomic Ratio	Critical Volume (liters)		Critical* Mass (Kg Pu)
1143134	10-19-62	10" Concrete	40.5	4.40	1.214	850	315	583.1	21.93	+.02 -.02	0.89
1143135	10-19-62	10" Concrete	38.4	4.46	1.205	844	315	611.2	22.52	+.02 -.03	0.86
1143136	10-22-62	10" Concrete	36.8	4.42	1.200	848	312	640.4	22.96	+.03 -.04	0.84
1143137	10-23-62	10" Concrete	36.5	4.38	1.213	864	309	657.0	23.27	+.05 -.09	0.85
1143138	10-24-62	4" Air Gap + 6" Outer Shell of Concrete	36.8	4.43	1.214	862	308	650.6	29.33		1.08
1143139	10-25-62	4" Air Gap + 6" Outer Shell of Concrete	47.5	4.27	1.225	859	314	501.6	25.68		1.22
1143140	10-26-62	4" Air Gap + 6" Outer Shell of Concrete	69.0	4.34	1.259	845	341	340.1	23.09	+.07 -.09	1.59
1143141**	10-29-62	4" Air Gap + 6" Outer Shell of Concrete	67.9	4.29	1.250		336		23.20	+.07 -.11	1.58

* Includes 4.6 w/o Pu^{240}

** Chemical analyses incomplete

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The critical concentrations of Pu in the full sphere, evaluated from the data in Table I, for various experimental conditions are given in the following table:

TABLE II

CRITICAL CONCENTRATION OF Pu IN 14-INCH SPHERE

(Measured Sphere Volume 23.22 Liters)

Reflector	Concentration (g/l)	Acid Molarity	Total Nitrate (g/l)	Critical Mass Kg*
Full Water	47.5	6.66	462	1.10
10-inch Concrete	36.6	4.39	310	0.85
10-inch Concrete	43.4	6.37	445	1.01
4-inch Air Gap + 6-inches of Concrete	67.9	4.29	336	1.58

* Includes 4.6 w/o Pu^{240}

During the month of October (in a period of 23 working days) a total of 20 critical approach experiments were completed. Only eighteen of these are listed in this report, since the chemical analyses for the last two experiments were not available. From a purely operational viewpoint, this has been the best month to date since beginning experiments in the Plutonium Critical Mass Laboratory.

Experiments with Plutonium Oxide-Plastic Mixtures

Work continues in preparation for the installation of the remotely operated split table machine in the second hood of the critical assembly room. The neutron source drive for use in the critical approach experiments with this device was received during the month.

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Calculations were made on a poison type rod for the PuO_2-plastic fuels. The results from GAM, Tempest, and the HFN codes show a 1.2 cm diameter solid stainless steel rod would have a control rod strength of approximately 75 cents for plutonium fuel with an H/Pu ratio of 5; the control strength would be greater at lower concentrations of plutonium in the plastic mixtures. Tentative plans are to use a blade type control rod of 3.2 cm width and 0.3 cm thickness.

Input Data for GAM - Tempest Chain

It is possible to determine the critical geometry of heterogeneous assemblies, with special emphasis on slightly enriched uranium-water lattices, using a multigroup diffusion calculation provided a set of homogenized multigroup constants can be obtained that effectively represent the heterogeneous system. On providing the GAM - Tempest chain with proper weighting factors, a set of homogenized multigroup constants that represent a particular heterogeneous system can be obtained. The coding of an input data code to provide these necessary weighting factors for the GAM - Tempest chain has been initiated and is proceeding.

This code can provide homogenization factors for assemblies having regular cell structures containing up to seventeen concentric regions in a unit cell. A P_3 approximation to the transport equation provides the neutron flux values for determining the thermal group weighting factors. The fast fission weighting factors are obtained from first flight collision probabilities. The collision probabilities are calculated by the techniques of R. Bonalumi (Energia Nucleare, Vol 8/n5/1961) in concentric geometries; the fast fission interaction effect between neighboring fuel cells is determined by the model previously outlined (HW-73116).

Nuclear Safety Parameters for 0.5-inch Diameter, 1.8 w/o Pu-Al Rods in Light Water

The buckling values as reported in HW-74190, Subcritical Measurements with 1.8 w/o Pu-Al Rods in Light Water, Physics Research Quarterly Report for April, May, and June, 1962, have been used to estimate the following minimum critical parameters for water moderated and reflected arrays of these elements.

Minimum critical mass	2.28 Kg Pu
Minimum critical volume	45.4 liters
Minimum critical slab thickness	9.3 inches
Minimum critical cylinder diameter	17.6 inches
Minimum critical mass per unit of area	41.8 lbs alloy/ft^2

The above values are of use in nuclear safety evaluations concerning the handling and storing of the fuel rods.

Buckling of Partially Filled Spheres

Debugging was completed on the partial difference code representing numerically the equation

$$\frac{\int (\nabla^2 \phi + B^2 \phi)\, dV}{\int dV} = 0$$

for bare truncated spheres. Inversion of the matrix representing ∇^2 was not accurate enough to produce good eigenvalues.

A simultaneous equation method was applied to n-1 equations repeatedly to minimize the residual of the n^{th} equation in the nxn matrix. This was also unsuccessful, apparently because of the constant flux assumption used in the integration of $\int B^2 \phi \, dV$.

A new formulation using the simple wave equation has begun. Discontinuity at the origin appears to be the only serious difficulty.

A short code was written which gives the approximate buckling of a truncated sphere with water reflector.

Instrumentation and Systems Studies

Work started on preparing an analog simulation of a critical mass for use in establishing the proper characteristics for the instrumentation at the Critical Mass Laboratory. It is also expected to be useful for evaluating and demonstrating data presentation methods which might be applicable in critical mass experiments.

The neutron generator that will be used for pulsed neutron experiments is being returned to the factory. It will be rebuilt into a more modern and reliable unit. Pulse neutron experiments are expected to start within two months.

Information on the PuBe source of the Critical Mass Lab was gathered this month. Since the information contained only number of neutrons per second, a search was made for range and fraction for this particular kind of source. The information was then converted to power in watts.

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Mass Spectrometry

Isotopic analyses were provided on samples of uranium and plutonium in support of Plutonium Recycle Program studies. Three of the samples were high enrichment Pu^{240} fuel for PCTR experiments. Three uranium analyses were performed on samples from PRTR fuel element No. 1041. In addition, five uranium sample analyses were provided for Testing Methods Engineering Operation, FPD.

Some studies were made on the stability of plutonium-ion emission from different Pu sample solutions. The observed instabilities have not yet been correlated with chemical preparation procedures.

Consulting Services on Nuclear Safety - Criticality Hazards

Nuclear Safety in HLO

The nuclear safety of the SRL 70 ton shipping cask for Mark I spike fuel elements (1.8 w/c Pu-Al) was reviewed for the Programming Operation. This review is part of a feasibility study being made by the AEC. The internal dimensions of the cask are 15 ft long by 27 inches wide by 43 inches high. The following conditions were estimated to be safe:

a) A vertical slab of four stacked baskets containing four unirradiated Mark I assemblies each,

b) Two such vertical slabs separated from each other in the cask by two existing cadmium filled stainless steel spacers (32 Mark I assemblies),

c) Five Mark I filled baskets in any configuration.

A meeting with Technical Planning personnel of PRTR was held October 22, 1962. The discussions concerned the nuclear safety aspects of increasing the storage capacity for Mark I fuel elements in the PRTR storage basin. It was pointed out that based on the criticality measurements made recently on spike fuel rods (1.8 w/o Pu-Al alloy), the storage arrays could be revised. New arrays would be based on a safe slab thickness of 7.9 inches and a minimum spacing between slabs of 12 inches.

Nuclear safety was reviewed for the Plutonium Metallurgy Operation on October 3 and 10, 1962. Visits were made to the 308 Building and 231-Z Building. During this review, the nuclear safety specifications were reviewed in detail, revised, and reissued. The specifications now in effect are as follows:

J-1 General Rules for Plutonium Handling in Dry Glove Boxes
J-2 General Rules for Pu Handling in Wet Glove Boxes
J-3 Rules for the Transportation and Storage of Plutonium
J-4 Rules for Preparing Plutonium Metallographic Samples
J-5 General Rules for Plutonium Handling in Dry Glove Boxes with Internal Water Lines
J-6 Special Rules for Casting Metallic Plutonium and Plutonium Alloys in Amounts to 7.0 Kg
J-7 Rules for Processing Thin Walled Plutonium Metal Castings

The following specification was issued for the Technical Shops:

K-4 General Rules for the Storage, Handling, and Processing of Slightly Enriched Uranium, October 12, 1962.

One specification was issued to cover Pu-Al fuel rod handling in both Reactor Lattice Physics and Experimental Reactors Operation:

A-2, B-3 Rules for Storage and Handling of Pu-Al Alloy Fuel Rods

Nuclear Safety in CPD

Participation on the Recuplex Deactivation Hazard Review Committee and the Project 880 (New Recuplex) Hazards Review Committee continued throughout the month. Procedure A-32 concerning the acid flushing of the D-7 sump tank was reviewed and approved. The D-7 tank, which is 10-ft in diameter, is estimated to contain not more than 660 g Pu and probably much less than this. Cadmium nitrate will be added to the tank prior to the acid flush.

A specification covering the storage of 7.35 w/o Pu-Al fuel rods outside of the 234-5 Building was reviewed for CPD. About 113 rods are to be transferred from HLO to CPD for storage. The rods are in birdcages.

7.35 w/o Pu-Al Alloy Fuel Element Storage Outside 234-5 Building, October 23, 1962

A CPD meeting with T. E. Harrington and W. B. Stockdale of ORNL was attended October 18, 1962. Mr. Harrington and Mr. Stockdale are making an economic study of sea water desalination in large nuclear process heat reactors and visited CPD personnel to discuss the technical and economic aspects of fuel reprocessing. The fuels under consideration are natural UO_2 and 15 w/o PuO_2. In the course of the discussions, it was pointed out by HLO and CPD nuclear safety specialists present that the geometry

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limitations on fuel element dissolving equipment for 15 w/o PuO_2 would be quite restrictive. The minimum critical parameters were estimated to be about 1 Kg Pu for mass; 4-10 liters for volume; 6-7 inches for cylinder diameter; and 1.5-3 inches for slab thickness. An engineering study to devise unique dissolving equipment would be needed.

NEUTRON CROSS SECTION PROGRAM

Scattering-Law Measurements for Room-Temperature Light Water

Some progress has been made on the development of a calculational method for correcting slow-neutron scattering cross-section data for the effects of multiple scattering.

Scattering-Law Measurements for Light Water at Elevated Temperatures

Two sample holders have been designed and constructed for the measurement of inelastic scattering of slow neutrons from water at elevated temperatures. One sample holder is suitable for temperatures below the boiling point of water and the second is designed for temperatures up to 150°C. Both sample holders have operated satisfactorily although the effects of corrosion in the high-temperature sample holder are still under investigation. A temperature control system has been completed which controls the indicated sample temperature to within 1°C.

Rotating-Crystal Spectrometer

Studies of the characteristics of the rotating-crystal equipment to measure inelastic scattering of slow neutrons by time-of-flight are in progress on the triple-axis spectrometer. Satisfactory intensity and background levels for scattering measurements have not yet been achieved. The 1024 channel data-storage system is still not operating satisfactorily.

Fast-Neutron Total Cross Sections

Analysis is in progress of the fast-neutron total cross-section data obtained in September. Satisfactory analyses have been completed on about one-half of the samples which were measured including Mg, Ca, V, Nb, and Pb. The data which were taken to study possible systematic errors in measurement due to incorrect background determination and in-scattering effects have been analyzed. These results indicate that the systematic error of measurement due to these effects is less than 0.5 percent of the total cross section measured with a sample of nominal one mean-free-path thickness over the energy interval of 3 to 15 Mev.

A subprogram has been written which calculates the energy resolution of the total cross-section measurements from periodic measurements made during each run of the $Be^9(d,n)B^{10}$ neutron spectrum.

Samples of Cr, Mn, As, Sb, Se, Te, S, and I were prepared for future total cross-section measurements by compressing powder, granules, or crystals into stainless-steel sample holders.

Instrumentation

Design criteria was finished, and sketch completed for the modification of the 1024-channel slow neutron time-of-flight analyzer to a 6144-channel analyzer.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE

Lattice Parameters for Low Exposure Pu-Al Fuel

The Hanford P-3 program is being used for a final rerun to obtain the thermal flux distribution for the 10-1/2, 8-3/8, and 6-1/2 inch graphite lattices fueled with 19 rod clusters of PuAl fuel. Preliminary P-3 calculations have been done for both the poisoned and unpoisoned cells for each lattice spacing, but some modification of the original input data is necessary. The IDIOT program has not proven to be as flexible as the P-3 code on this work because exact values of the atomic mass number and source term cannot be used for input.

The 10-1/2 inch poisoned lattice is being prepared for a fourteen group HFN calculation (13 thermal and 1 epithermal group). Material parameters are being prepared in a manner such that all necessary data may be stored in the memory spaces allotted by the HFN program.

Since multigroup parameters in and near the thermal energy region are needed in plutonium lattice calculations, a set of 5 subroutines has been written which can be incorporated into the code, Spectrum V. Output consists of average cross sections, including a complete transfer matrix, for the arbitrary specified groups in the thermal energy range. Four of the subroutines have been debugged. The fifth contains the output format and needs further work. An informal report describing this code, to be called Spectre, is being written.

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High Exposure PuAl Lattice Studies

The preparation and planning work for the experimental portion of these studies is essentially complete. Special cadmium covers and holders for the PuAl foils have been fabricated.

Two thirds of the high exposure PuAl fuel rods have been completed by the Plutonium Fuels Operation. Chemical analyses of samples of the fuel core material indicate that the plutonium concentration may vary about 10% from one end of a 200" extrusion to the other.

The neutron flux data obtained in this experiment is sensitive to the plutonium concentration in the 20" elements that make up the central cell. Therefore, the relative plutonium contents of each of the 19 rods in the central cell are being compared by comparing their reactivity coefficients in the PCTR. The elements are placed in the center of the graphite core with the driver fuel loaded in a circle of about 50 cm radius.

Low Exposure PuO_2-UO_2 and PuC-UC Lattice Studies

The Hanford P-3 program has been used to obtain the thermal flux distributions for 6-1/2 inch unpoisoned graphite lattices fueled with 19 rod clusters of both PuO_2-UO_2 and PuC-UC. The quantity of Pu and U was taken to be the same for both the oxide and carbide cases. Several changes in input were made and the changes in the ratio of the average thermal flux in the graphite region near the cell boundary to the average thermal flux in the central fuel rod were noted. An increase in the source term $3\Sigma s$ in the fuel region to 130% of the original value caused the flux ratio to decrease by about 1.1%. A decrease in the atomic mass number in the fuel region (which determined $\overline{\mu}_c$) to 33% of the original value caused the flux ratio to decrease by about 0.3%. An increase in the scattering cross section Σ_s in the fuel region to 110% of the original value caused the flux ratio to increase by about 1.7%.

It appears that an experimental substitution of PuC-UC for PuO_2-UO_2 in which the quantity of Pu and U remains the same does not entail a sufficiently large change in the source term, average scattering angle per collision, or the scattering and absorption cross sections to make the experiment instructive. The main difference between the two cases is in the effective ξ since the molecular binding energy appears to be only about 1.2 ev/bond for the carbide while it is about 5.6 ev/bond for the oxide.

Development work on the ceramic pellets for the mixed-oxide fuel elements for PCTR and PRCF experiments has been started by Plutonium Fuels Development. The plutonium to be used for these elements has a Pu^{240} content of 7.57 percent.

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Pu^{240} Effective Resonance Integral Experiment

The rough draft of a report on the Pu^{240} effective resonance integral experiment has been completed. Revision of this draft is in progress. The report is being prepared for submission to an appropriate technical journal.

The Critical Facility

The final draft of the process specifications which will govern the operation of the PRCF have been reviewed. Revisions to the specifications have been suggested to the authors.

A document, "PRCF STARTUP TEST PROCEDURES", HW-71214 Supplement, has been completed and is being distributed for review and comment. The design drawings of the thimbles for the Void, H_2O and D_2O substitution experiment for startup are being prepared.

Teflon adaptors have been made to attach cadmium-covered and bare BF_3 proportional counters to the flux traverse machine.

A proposed schedule and outline of experimental programs for the first two years of operation of the PRCF has been drafted. The schedule attempts to fit the various types of experiments together so that the expected uses of the critical facility with D_2O moderator will not conflict with those which use H_2O.

In support of the measurements on irradiated fuel rods, work was resumed on the effect of large sources of neutrons from the (γ,n) reactions on reactor periods.

The feasibility of an experimental evaluation of models for cylindricizing fuel clusters is being considered.

Neutron Spectrum Studies

Foils made from Lu_2O_3 and Al_2O_3 with varying amounts of Lu_2O_3 were irradiated and counted. The results of this data can be used to determine the self-shielding properties of lutetium.

Work on the decay scheme of lutetium continued this month. An analysis of the data obtained in a measurement of the distribution of high-energy beta rays emitted by Lu^{176m} has been completed. The distribution has been divided into two components by the usual Fermi-Plot analysis. By this method, the distribution of beta rays with the highest "end-point" energy is subtracted from the total distribution. Then the "end-point" energy of the second component is determined from the remaining distribution. The "end-point" energies of the components are 1.319 ± 0.005 Mev and 1.204 ± 0.005 Mev with intensities of 0.46 and 0.54, respectively. The difference in the "end-point" energies is 115 ± 7 kev. This difference should be equal to the energy (89 ± 1 kev) of the gamma ray which also is emitted in the decay of Lu^{176m}. The reason for the disagreement is that the components are of equal intensity and have "end-point" energies which are nearly the same. In such cases the relative intensities and the difference in "end-point" energies are sensitively related.

For this reason, all of the data were reanalyzed in one process by fitting it using a subroutine written for the Generalized Least Squares program (HW-68858). During the fitting process it was required that the high-energy end point be fixed at 1.319 Mev and that the end point of the second component be fixed at 1.230, a difference of 89 kev. The relative intensities were then allowed to vary until a best fit was obtained. With these restrictions the data were fit properly and the intensities were 44% and 56% for the 1.319 Mev and 1.230 Mev components, respectively.

Calculation of Non-Maxwellian Flux Shape in PRTR Fuel Element

The shape of the non-Maxwellian component of the neutron flux in PRTR fuel elements is needed to correct experimental lutetium activities for non-1/E flux. The deviations from 1/E which significantly affect lutetium activities in the Mark I-H PRTR Al-2.0 Ni - 1.8 Pu fuel elements occur below 0.7 ev. They result from the flux depression caused by the Pu-239 cross section and from thermalization effects in the portion of the spectrum where the "1/E tail" joins the Maxwellian. The one-dimensional multi-group diffusion code HFN, and the homogeneous medium thermalization code Spectrum have been used to generate the required flux.

Phoenix Fuels for Compact, Water Moderated Reactors

Work on the derviation of suitable group constants for the compact, water-moderated Phoenix cores is proceeding. The TEMPEST, SHUSH-HFN routines for the thermal group constants are operational. The GAM slowing down code is to be used for the generation of epithermal, intermediate, and fast energy group constants. The treatment of the Pu-240 resonance in heterogeneous geometries by means of the GAM code presents some difficulties. Resonance shielding for the plutonium isotopes is not automatically handled, and appropriate self-shielding factors must be calculated separately. At the present, Dresner's methods have been used to obtain effective resonance integrals for Pu-240 for various geometries and plutonium concentrations. These methods have been checked successfully against recent experiments with

Pu-240.* The self-shielding factors are being incorporated into the group constant calculations.

Using the above cross section routine, reactivities are being calculated for reflected $Zr\text{-}H_2O$ assemblies (M/W = 1) by means of the HFN code. For a 500 liter core, total plutonium loadings of 30, 50, and 70 kg have been considered so far. All plutonium composites contain 10 percent Pu-240. The resulting multiplication factors and non-leakage probabilities are tabulated below:

kg of Pu	k_∞	k_{eff}	L
30	1.3507	1.1986	.8874
50	1.3397	1.1954	.8923
70	1.3352	1.1956	.8954

The relative constancy of k with plutonium loading should be noted. Calculations of k over a wider range of Pu loadings are in process.

Plutonium Utilization Studies

Work on the plutonium-uranium comparison for compact fast reactor systems is continuing.

Plutonium utilization in a small, 1 MW, auxiliary power source results in core size reductions of about 50 percent, compared to a fully enriched U-235 system. A Pu for U substitution in a nuclear rocket propulsion unit results in a similar 50 percent core volume reduction. The effect of these core volume reductions on over-all engine weight are presently being examined.

Physics statics calculations for a large, 30 MW unit are also being carried out. Various schemes for reducing control requirements are being examined. An annular core geometry containing a central U-238 plug may have some promise for reducing reflector control.

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* Nichols, P. F., "Summary: Reactivity Determination of the Effective Resonance Integral of Pu-240 in Pu-Al Rods," to be presented at ANS 1962 Winter Meeting.

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Nelkin Scattering Kernel for Water

The second Hanford version of KERNEL, the computer code based on Nelkin's theory of scattering in water, gives results in agreement with the original version. This is confirmatory evidence that the codes in current use do evaluate the theory correctly.

A detailed comparison of theory and experimental measurements of slow neutron scattering in water, has been prepared for the Physics Research Quarterly Report. From such a comparison, and with the evidence that KERNEL evaluates the theory correctly, one deduces that the basic constants in Nelkin's theory should be altered. This is not surprising in view of the paucity of data available at the time Nelkin formulated his theory.

Presently, Nelkin's basic constants are being modified as inferred from the Hanford and Chalk River scattering data used in this comparison. It is anticipated that much of the discrepancy between theory and experiment can be eliminated with better values of these constants.

Integration of the Egelstaff S-function

The customary use of the Egelstaff S-function representations in reporting the results of neutron scattering experiments suggests that a similar representation is appropriate for expressing theoretical results. The Egelstaff S-function, therefore, becomes a useful starting point in describing the scattering properties of a moderator. However, to obtain transfer cross sections for reactor analysis, one must integrate the S-function over the cosine of the angle.

Therefore, we are investigating integrating the Egelstaff S-function with numerical methods. In particular, we are studying Simpson's Rule and the Gaussian Quadrature Formula with the S-function for an ideal gas as the integrand. The ideal gas case provides a good means of checking the accuracy of the numerical methods since an analytic expression for the integral has been derived. Although the results are not complete, there is a strong indication that the Gaussian form will be the more accurate of the two methods.

Code Development

Variational Optimum Kinetic Normal Modes

Recent transport-theory development work has led to substantial improvement in the theory of nuclear kinetic effects in breeder reactors. A variationally optimized machine analysis of nuclear kinetics based upon multienergy transport perturbation theory has been formulated for the IBM-7090 computer, is now available for exploratory evaluative use as a part of GE-HAPO program S (version S-XI), and is expected to be made available for external distribution after completion of relevant documentation.

RBU

Values of $\bar{\mu}_0(E_0)$ and $\overline{\Delta E}(E_0)$, the average cosine of the scattering angle and the average energy loss per collision of a neutron of initial energy E_0 in a gas, were generated by the new program MOMENTS for scattering masses 1 through 31. The same quantities were calculated for hydrogen in light water using Nelkin's light water kernel, and the results plotted along with the gas model data. Effects of molecular binding on the scattering law for low energy neutrons with hydrogen are quite large.

Although some work remains in calculating $\bar{\mu}_0(E_0)$ more accurately in MOMENTS, a double mass variation to be used in the Monte Carlo has been obtained using the data now available.

RBU Cross Section Updating

The updating of the cross sections listed below is complete, and some preliminary tests are being run on the library. A document describing the methods, and references used, is forthcoming.

Plutonium - 239, 240, 241, 242
Uranium - 235, 236, 238
Others - Zr-40, He-4, H-2, O-16

TEMPEST Cross Section Updating

The program to obtain the cross sections from the RBU library and punch the library for TEMPEST is 90 percent completed, and upon completion will be utilized in obtaining the cross sections in the necessary format for updating TEMPEST.

BARNS Revision

The formulas for producing 68-group parameters for GAM-I from the RBU basic library tape have been re-examined and several errors have been found. Instructions for preparing a version of the BARNS code incorporating these formulas have been clarified. The actual programming of the revisions is being done by EDPO.

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CALX

Debugging of the SIGMA subroutines of CALX continued. Subroutine SIGMA now gives average fluxes for a nine-group, two-region problem which are consistent with SIGMA-3E, HFN results, when both use the same TAM library (combined GAM and TEMPEST output).

The SIGMAT subroutine of CALX has been modified so it will accept card input after the CALX data tape has been read. This allows various items of data from the data tape to be changed before the burnup portion of CALX is run.

Instrumentation and System Studies

A new design was started for the proposed gamma scanning facility for PRP. The original work concerned a collimator-detector system for use in air measurements; however, the new requirements are for an underwater scanning system for use in the PRTR basin. A general scope drawing for the new collimator was completed and issued for review and comment.

For use at the Plutonium Recycle Critical Facility, a preamplifier circuit was developed for direct connection to the applied fission chambers. The preamplifier uses General Electric high temperature ceramic vacuum tubes, and the complete circuit was designed to withstand the known nuclear environment at the chamber for at least one year. The preamplifier was the logical approach to improve the present inadequate signal-to-noise ratio, and laboratory tests were successful. Installation will proceed at PRCF location as soon as possible.

Both the installed first generation scintillation gamma emitters effluent monitor in use at PRTR and the final model unit, still in constant laboratory operation, performed correctly for the month. Both units have been fully satisfactory and a direct liquid radionuclide source calibration was planned for the detectors to provide better measurement information. The probes are installed in PRTR liquid effluent Manhole No. 2. All seven printed circuit masters were completed for the final model effluent monitor, and work was started on the complete circuit drawings.

The Boonshaft and Fuchs transfer function analyzer used for the analysis of signals from the PRTR was received from the manufacturer during the month and has been partially tested. The tests made to date indicate a marked improvement in operation, especially in the multiplier zeroing circuits. It was immediately used as a source of interrogation signals in the testing of a cross-spectrum analyzer set up on the EASE analog computer. Initial results of tests on the cross-spectrum analyzer circuit are encouraging. If successful, this device will allow the determination of system transfer functions by means of random signal testing. Further tests were made on equipment designed to provide rapid determination of the power density spectra of random signals. Tape speedup techniques were used to obtain the power density spectra of randomly varying signals over a three decade range in fifteen minutes.

Work continued on the problem of obtaining quantitative measurements of the fuel assembly vibrations encountered in the PRTR. An investigation was conducted, using a bench mockup, to determine the feasibility of measuring the amplitude, frequency, and direction of these vibrations with an eddy current instrument and the findings have been issued in a memo report, PM 62-14. Further investigations are being conducted to determine the behavior of the sensing coils at reactor operating temperatures of 500°F.

HIGH TEMPERATURE REACTOR LATTICE PHYSICS PROGRAM

Preliminary analysis of the High Temperature Lattice Test Reactor has been oriented to a reduced scope in an attempt to keep the cost within the desired figure of $1,900,000, which includes $100,000 for a fast chopper for spectrum measurements. The size of the graphite cube has been cut from 11 feet to 9 feet, the building has been reduced in size, and the design temperature has been cut from 1200°C to 1000°C except for permanent components, such as insulation, which would not be practical to upgrade later.

Flux traverses and cadmium ratio measurements were made near driver fuel elements in the PCTR to aid in defining the appropriate lattice cell for a non-repeating lattice like the PCTR or HTLTR driver rings. The results indicate that the cell should be considered rectangular with the long side equal to the driver spacing and the short side about half the long side.

NEUTRON FLUX MONITORS

It was estimated that the neutron temperature, T, in a reactor test facility can be determined to within about ± 10% using the mass spectrometer to determine isotopic ratios of irradiated U-235 and Pu-239 samples. By determining the activity ratio between bare and cadmium-covered cobalt samples, it is estimated that the parameter "r" can be obtained to within about ±5%. If these experimental accuracies can be obtained, the optimum isotopic composition of detectors can be calculated with sufficient accuracy to use in the experimental evaluation of the regenerating detector technique.

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To determine the cadmium ratio of cobalt samples, it is necessary to utilize several foils of different, known thickness values. An extrapolation then can be made to zero thickness, or infinite dilution, for the determination of "r". The U-235 and Pu-239 samples are scheduled to be irradiated in the form of solutions contained in quartz samples. Preliminary discussions regarding fabrication, irradiation, and analysis of the samples indicated that no unusual difficulties are to be expected.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

Fabrication and testing of the electronic units for the prototype multiparameter eddy current testing equipment are proceeding. Several three-dimensional models of eddy current distribution in molten woods metal have been made. The graphical nulling device has proven to be very useful when used in a test instrument in the field.

Twelve commercial operational amplifiers together with a power supply and a housing unit for the amplifiers have been ordered for use in the prototype multiparameter eddy current tester. In addition, the 250 Kc and 3 Mc tuned amplifiers for this tester have been modified and successfully tested. However, testing of the crystal oscillators revealed the desirability of incorporating a buffer amplifier for each oscillator. The amplifiers are necessary to reduce coupling between the various crystal oscillators. Two of the oscillators have been equipped with the required buffer stages.

The study of the propagation of eddy currents in liquid woods alloy is continuing. By using eleven different sizes of pickup loops, it was possible to construct several three-dimensional models of the eddy current distribution in the metal. These models represent the magnitude of current at various distances from the center of the drive coil and also at different depths in the metal. Each model represents the current distribution at a given time. The series of models permits one to visualize how the current amplitudes change with time. In obtaining the models, a large flat spiral coil was driven with a step function of current to produce the eddy currents.

The graphical nulling device incorporated in an eddy current tubing tester is now being used in field testing of the NPR instrument line (3/16 inch diameter 304-L stainless steel tubing). The field tests show that use of the new device aids materially in ease of adjustment of the tubing tester operating conditions and in the interpretation of test results. The use of a newly developed spring mounted coil assembly permitted the testing of tubing containing appreciable bends. Stability and reproducibility of the tester were good. Chart recordings of tests made on the same lengths of tubing on separate days showed identical traces. The standard used consisted of a 0.004 inch deep electromachined notch on the outer surface 1/4 inch in length. Under operating conditions a 0.010 inch deep notch on the internal surface resulted in a signal amplitude 1.75 times that given by the .004 inch outside surface standard notch. A 0.009 inch deep notch on the outside surface resulted in a signal amplitude 6.5 times that given by the 0.004 inch outside surface standard notch. A total of about 8000 feet of tubing has been tested.

Based upon observations of the eddy current test signal indications, estimates were made as to nature of the tubing irregularities. The tubing specimens were then destructively examined by Physical Testing. Estimates and corresponding destructive examination results are as follows:

Section 1. Estimated to be a surface type irregularity about .005 inch deep. Sectioning revealed an irregularity extending to .0065 inch deep.

Section 2. Estimated to be less than 0.009 inch deep. Sectioning revealed an area of separation along the gain boundaries extending to about 0.005 inch deep with associated metallurgical irregularities extending to the region of about .010 inch deep.

Section 3. Estimated to be an outside surface type irregularity 0.007 inch deep. Sectioning revealed an irregularity 0.0025 by 0.003 inch extending to a maximum depth of .003 inch below the surface.

Section 4. Estimated to be outside surface type .007 inch deep. Sectioning revealed a fold extending to about 0.004 inch below the surface, lying at an angle of about 45° with respect to the surface. Associated with the fold was a shallower irregularity extending about .005 along the surface.

Section 5. Estimated to be an irregularity on the internal surface extending about 0.010 inch into the wall from the internal surface. Sectioning revealed an area of internal surface irregularities extending to 0.005 inch depth extending 0.015 inch along the internal surface.

Section 6. Estimated to be an irregularity extending through the tubing wall. Sectioning revealed a crack extending from the outer surface to a depth of .013 inch.

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Section 7. Estimated to be an irregularity which extended through the wall. Sectioning revealed a crack extending from the outer surface to a depth of 0.020 inch.

Section 8. Estimated to be an irregularity extending through the wall. Sectioning revealed a crack extending from the outside surface to within .003 inch from the internal surface.

Section 9. Estimated to be an irregularity extending to a depth of .025 inch. No irregularities were revealed at first sectioning.

Section 10. Estimated to be outside surface type about .015 inch deep and of large area. Visible inclusions noted. Sectioning revealed an irregularity with maximum depth of 0.005 inch, extending along surface 0.012 inch.

Section 11. Estimated to be an outside surface irregularity extending to .020 deep. Visible inclusions. No irregularities were revealed at first sectioning.

These results show that the new instrument can be used to give fairly good estimates of the general depth extent of irregularities. On the average the actual depth was 68% of the estimated depth. It is expected that fabrication process type irregularities of the same depth extent as an electromachined notch will give a range of signal amplitudes depending on many factors including length, volume of metal affected, type of inclusion material if any, and type of irregularity.

Zirconium Hydride Detection

An eddy current probe capable of operating at 77°K was developed and was used to test hydrided Zircaloy-2 laboratory samples.

The tests were made at both room temperature and 77°K on several sets of samples having different histories. These samples contained 0, 10, 50, 100, 300, 420, 500, 700, 1000, 1500, 3000, 5000, 9000, 10,000, and 15,000 ppm hydrogen.

At room temperature, it appeared that samples containing between approximately 2000 and 5000 ppm hydrogen could be distinguished from those at the lower concentrations in a given set. However, additional samples are needed to determine whether or not the curve of signal versus concentration is monotonic in this range. Concentrations above 5000 ppm can also be detected, but the curve has multiple valued regions below 2000 ppm.

Cooling the samples to 77°K increased the test sensitivity and reduced the extent of multiple-valued regions of the signal versus hydrogen content curve below 2000 ppm. This made it possible to uniquely distinguish samples having hydrogen contents between 100 and 500 ppm and between about 1200 and 15,000 ppm within one set of the samples. Differences in signal at different points on some single samples indicate that either the hydride concentration, or some other factor, is somewhat non-uniform. Samples were submerged in liquid nitrogen during the low temperature tests. The eddy current probe developed for this purpose was repeatedly thermally cycled between room temperature and 77°K, but performed well through the test.

The eddy current test is sensitive to some variables other than hydride content at both room temperature and 77°K; samples believed to have the same hydride content, but having different histories, gave different signals. However, a difference between hydrided areas and surrounding non-hydrided metal was detected in two Zircaloy-2 process tube sections tested at room temperature. The areas contained 2000 and 5000 ppm hydrogen. These samples have not yet been tested at 77°K.

Further study is under way to determine the effects of variables other than hydride content on the test results.

<u>Heat Transfer Testing</u>

Laboratory prototype circuitry to convert outputs of the dual infrared radiometer heat transfer testing system to emissivity independent outputs was assembled. The system was evaluated by testing sixteen aluminum clad uranium fuel elements.

The system effectively reduced the emissivity dependence below a detectable level in many of the fuel elements. Variations in signal from others could have been due to actual surface temperature variations, but this was not demonstrated conclusively in the initial tests. Noise from the infrared detectors and their preamplifiers is an important factor in the new system since noise in the two radiometer outputs is additive. However, a 1/4 inch diameter mica standard heat transfer defect in the fuel core to cladding bond of an aluminum clad uranium fuel element could be detected. It should be possible to reduce radiometer noise below the present level.

The circuitry used to convert the radiometer signals consisted of an Ampex FR-100 tape recorder, which was used to record the original radiometer outputs, an Ampex continuous tape loop time delay, and a ratio circuit. One recorded dual radiometer output was fed directly from the FR-100 into the ratio circuit, and the other was fed through the time delay

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into the ratio circuit. The time delay is necessary in order to permit comparison of signals from a given area on the test piece surface at two different times.

Analysis of the results from the initial tests indicates that improvements can be made not only by reducing noise in the radiometer outputs, but also by reducing signal fluctuations arising in the time delay and associated circuits, and by compensating for the non-absolute zero reference temperature used in the radiometer.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Fabrication and replication of deep, inside surface transverse notches has proven to be somewhat of a problem. Measurements at 5 mc indicate that ultrasonic response as a function of notch depth is nonlinear. Recent evidence indicates that the optimum ultrasonic test procedure for the longitudinal test may be one that propagates a shear mode at 45° within the tubing wall. Nine tubes from the group of 200 tubes tested in a manufacturer's plant were rejected based on results obtained in the longitudinal test. Analytic studies have shown that the Lamb-wave frequency, particle vibration, and wave amplitude equations are continuous for all phase velocities and for all modes.

In the preparation of standards controlling electrode shapes during the electro-machining of deep inside surfaces, transverse notches continue to be a problem. Replication of these deep transverse notches was also found to be difficult because the replica material cannot be cleanly removed from the narrow deep notch. By using electrodes of greater thickness both the above difficulties may be resolved. One mil thick tungsten electrodes have been used to date; it is estimated 5 mil thick electrodes would provide the necessary clearance and maintain sufficient stock to keep the corners from rounding excessively. Some available 10 mil stock is being milled down to the appropriate thickness.

Fabrication of the additional notch standards for extending defect depth studies is nearly complete. The outside surface notches machined in 0.035 inch wall thickness tubing, which were originally used for transverse test studies at an ultrasound frequency of 10 mc, were used for preliminary studies at 5 mc. As was previously found at 10 mc the ultrasonic response at 5 mc also proved to be a nonlinear function of notch depth. The response curves at 5 mc for three different Lamb-wave-mode entry angles were similar in shape. The Lamb mode generation was verified, as previously, by the sharp increase in signal at specific angles as entry angles were changed. In general the response at 5 mc was found to increase rapidly for shallow notch depths, increase less rapidly for intermediate depths, and gradually taper off with little increase in signal from a notch depth of about 0.012 inches through a depth of 0.0175 inches which is equal to one-half the wall thickness. The response curve was also obtained for an entry angle which corresponded to propagation of shear waves at 45° within the wall. This curve increased less rapidly for shallow notches, more rapidly for intermediate depths, and finally tapered off near the one-half wall thickness depth. In review of the above results, Lamb-waves would appear to have a greater detection sensitivity for shallow transverse notches. Response measurements with inside diameter transverse notches are continuing.

The ultrasonic response as a function of longitudinal notch depth in 0.035 inch wall tubing was also studied at 5 mc. All entry angles, corresponding to the Lamb and shear modes which were used for the transverse notch tests, were examined during the longitudinal measurements. As in the past, no discrete signal increases were observed during entry angle changes with the longitudinal test setup, thus indicating that Lamb-wave propagation is improbable. It was found that all Lamb-wave mode entry angles resulted in confusing response signals. In contrast, the entry angle which permitted shear energy propagation at 45° within the wall resulted in a relatively clear response. Consequently, only the response as a function of notch depth for the 45° shear propagation was determined. This curve was similar in shape to the shear propagation curve obtained during the transverse test measurements. Outside and inside diameter notches gave approximately the same shape response curves. With the improved signal clarity, the 45° shear mode may be the optimum method for testing for longitudinal imperfections. However, as is usually the case during conventional 45° shear propagation, the signals from identical outside and inside surface notches are not the same amplitude. This difference in signal is probably due to the difference in metal-path distance between the locations of the O.D. and I.D. notches in the course of the zig-zag, 45° shear propagation. The difference in metal-path likewise causes the signals from the outside and inside surface notches to be displayed at different time intervals. Therefore, in order to obtain equal rejection levels for outside and inside surface imperfections it may be necessary to have a dual gate system if this method is used for the longitudinal test. With entry angles proper for Lamb-wave mode generation (in flat plates or in the transverse tube test), a point in time can be found where the responses from inside and outside surface notches are equal and, in fact, may be reversed. Generally, these points are impractical because of either rapidly falling or rising response from one of the notches, and other general confusion as found in the tests this month. It would appear the solution for the longitudinal test

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parameters on this score can best be resolved in dynamic tests followed by destructive analysis.

Response measurements as a function of notch depth in 0.017 inch thick wall tubing at frequencies of 5 mc and 10 mc are continuing.

Modifications in the Hanford production test equipment are complete and tube testing as part of the correlation program has begun. Of the approximately 200, 0.505 inch I.D., 0.030 inch thick wall tubes tested previously in a manufacturer's plant, 80 have been tested for longitudinal imperfections. Initial longitudinal tests were performed in one circumferential direction only, using identical O.D. and I.D. notches, 1/20 of the wall thickness in depth and one wall thickness in length. Based on this initial screening test, 9 of the 80 tubes were rejected. These rejected tubes were longitudinally tested in the opposite circumferential direction and transversely tested in both axial directions. Fluorescent penetrant tests followed by destructive analysis is continuing for the twofold purpose of establishing the degree of correlation that exists and determining the effectiveness of the ultrasonic test for the particular testing parameters chosen. Further tests on remaining tubes will depend on the results of the completed tests on the nine rejected tubes.

Analytical studies pertaining to Lamb-wave phase velocity, V, in the region near the longitudinal velocity, V_L, continued. A newly found treatment of the differential equation at $V = V_L$ yielded a solution to the frequency equation which does not vanish for either the symmetrical or asymmetrical modes. Previously it was assumed that the asymmetrical mode frequency equation was not solvable at $V = V_L$. In harmony with these solutions the particle vibration equations leading to amplitudes of waves within the Lamb-wave carrying specimen must also be continuous. By suitable modification of the constants in the solutions to the differential equations, it was found that the particle vibration and wave amplitude equations were also made continuous through V_L. The modification of constants does not affect the frequency equations. This result, however, does not yet explain the previously observed experimental disappearance of the symmetrical modes at $V = V_L$. Analytical studies concerning this phenomena are continuing.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Five atmospheric diffusion experiments were conducted during the month, utilizing the elevated source at a height of 185 feet on the meteorology tower. The Hanford Fluorescent Tracer Technique, employed successfully for a number of years with ground level sources, was used without change. Meteorological conditions were unstable during four of the experiments, with wind speeds at 200 feet less than 10 mph in three cases and 28 mph in the fourth. During the fifth experiment, the atmospheric stability was approximately neutral and the wind speed 15 mph.

Normalized peak concentrations, when plotted as a function of distance, show the characteristic rise to a maximum at 3 to 15 stack heights, with the distance to maximum increasing with wind speed. At large distances, theory predicts that the exposures should decrease at a rate approximately equal to that for a ground source. However, during these experiments, the rate of decrease was much greater than that for a ground source during similar meteorological conditions.

Data from the special series of diffusion experiments conducted during July and August were received from computing. The data confirm the findings reported last month for the "30 Series" experiments with the decrease of cross-wind integrated exposure with distance stratifying according to atmospheric stability. In addition, the subtle deviation from a power function relationship believed to be related to the vertical dispersion and deposition processes was also found.

A theoretical model based on the bi-variate gaussian distribution was developed which accounts for dispersion of particles by the atmosphere and preferential settling or deposition of the particles due to difference in size. It is assumed that the cross-wind distribution is gaussian, but the same restriction is not made for the vertical. Instead, it is necessary that the vertical distribution for any one particle size be gaussian, but not the composite or observed distribution at downwind points.

In Air Force supported work, the Series III diffusion data from Vandenberg Air Force Base were further analyzed. Regression equations were determined for the decrease of normalized peak exposure with distance. The average regression lines for Series III daytime experiments were essentially the same as those for Series I and II. However, nighttime experiments showed a pronounced seasonal variation with the Series I and III summer tests being similar and showing a more rapid decrease with distance than the Series II nighttime winter condition where the greatest stabilities were encountered. In addition, the ducting phenomena occurred during some of the latter experiments, causing the tracer to pass above the first arc of samplers in the stable air and come to the ground at the second arc in high concentrations. This effect is due to modification of stable maritime air as it passes over the warmer land surface, developing instability in the lower layers.

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Dosimetry

Liden and Andersson reported the presence of Cs-134 in Laplanders (Nature 195 1040 (1962)). A preprint of the paper arrived while the measurements of Alaskan Eskimos were in progress. A close examination of our data also showed a small photopeak presumably due to Cs-134 in the cases of high Cs-137 body burden. Adding the spectra of the ten Eskimos with the highest body burdens gave a well defined Cs-134 photopeak. Comparing photopeak heights and correcting for gamma ray abundances and counter efficiency gave a Cs-134/Cs-137 activity ratio of 1.18%. Matching the sum spectrum of the subjects with Cs-134 and Cs-137 sources in water gave 1.25%. These agree approximately with the ratio found in Lappland if allowance is made for the decay of Cs-134 between the measurements. The sum spectrum of all the people counted at Anaktuvuk Pass showed the Cs-134 peak but it was not as well resolved as in the spectrum of the ten highest. No Cs-134 could be seen in the sum spectrum of 200 people from the vicinity of Richland counted during 1961--it could be there but be masked by the K-40 gamma rays. There is considerable interest in Cs-134 because it is not known how it could be produced as a fission product in weapons testing.

There was a small bump in the sum spectrum of the 200 Richlanders where the photopeak from Zr-Nb-95 would be expected. This might have been external contamination (in spite of the shower and shampoo the subjects get) or might have been in the contents of the GI tract.

The body burdens of potassium were calculated for the people counted during the Alaska study. The Eskimos appear to have a few percent more potassium per unit body weight than whites; this would indicate a slightly higher proportion of muscular to fatty tissue. The data showed a decrease in potassium with age similar to that found elsewhere.

A small program under the HLO contract with Swedish Hospital, Seattle, was started to give them the benefit of our experience with whole body counting at the same time that we benefited from their medical experience with isotopes to obtain some new whole body counter calibration data.

The method of cutting scintillation crystals mentioned last month was successfully employed in preparing two thin NaI crystals. This was done partly for practice, partly to obtain some thin crystals for test purposes.

The positive ion Van de Graaff operated satisfactorily during the month.

Dr. H. H. Rossi, Professor of Radiology (Physics) at Columbia University, used our positive ion Van de Graaff for two weeks in an experiment aimed at developing a rem-meter for neutrons and gamma rays. He accomplished his planned experimental work.

A modified readout system was completed for use with tissue equivalent chambers used in neutron studies.

Mound Laboratory has completed intercomparisons of their precision long counter with the one of ours that we loaned them. The latter is now being sent to Argonne National Laboratory for intercomparisons there.

The measurement of the correction for heat loss in the gamma ray calorimeter is partly completed. It appears that the discrepancies that led to the discovery of the loss will be practically removed by the correction.

Radiation Instruments

Experiments with a number of specially-fabricated pencil-type ionization chamber dosimeters, as modified for use in the automatic recharging dose meters, indicated a considerable variation in output pulse rise time for each recharge cycle. It was determined that a re-coating with Aquadag on the center quartz rod improved the electrical characteristics; thus, either the pencil dosimeters can be re-coated or the solid-state circuitry can be modified to secure proper performance. Two more final-form dose-meters of the type which provide only a single point signaling feature at either 50 mr or 100 mr, as required, were completed, and these two will be completely tested in the laboratory before delivery. One other experimental pencil dosimeter was modified, by using the shell from a standard neutron sensitive dosimeter, to provide an experimental device for neutron dose measures. The conversion was successful and tests were planned.

Investigation and planning were started to provide a suitable method of telemetering radiological and other information from animals to central receiving and measuring instrumentation at Biology. The approach seems to be completely valid and can permit good freedom of movement for the animals.

Experiments were started in an effort to provide more shock-resistant scintillation detector probes, especially for use with portable-type general survey instruments. In addition, simpler methods of assembly are being sought to reduce costs. Instructions regarding proper handling of the probes is part of the answer.

Development was started on a new automatic sample changer mechanism to provide a reliable, simple system for the monitoring and counting of pint-size bottles of liquid radionuclides with an 8-inch diameter NaI well counter. The indexing mechanism will move the bottles into position for counting and then back to storage. The basic design and fabrication were completed and tests started. A Geneva gear is used for sample indexing with cam-operated microswitches used for control. The development was carried out following the complete inability of commercial equipment to do the work.

Considerable progress was made regarding the experimental portable mast measurement system. The gating-circuitry, to permit sequential digitization of the six measured temperatures during scanning, was designed, fabricated, and tested. All emitter follower circuits were completed and the transistor trigger circuits were redesigned to provide compatibility with the developed anemometer solid-state amplifiers. Wind speed storage resets, identification gates to provide printed output data point identification, the wind speed display frequency meter circuit, the wind direction integrator circuits, and the calibration test-pulse generator for wind speed circuitry testing were all completed in final form design and were fabricated and tested. The commercial printer and the digital voltmeter were both connected into the system and tested. The signal-to-noise ratios at logic points in the system were measured and found to be, quite satisfactorily, in excess of 30 db. A special circuit was developed to permit, at the option of the operator, one complete data group printout per minute in addition to the normal of twelve per minute; this selectivity will be useful for certain of the planned experiments. The over-all system is approximately 50% complete at this time.

All development work was completed on the scintillation, logarithmic response chopper input, solid-state area radiation monitor as fabricated in prototype form. Extensive tests showed proper performance, and the unit was calibrated to cover the dose-rate range from 1 mr/hr to 500 r/hr. Drawings of the complete circuit were started.

Experiments continued on a scintillation dose rate meter which can cover a four-decade dose-rate range with a logarithmic response, for use in either portable or line-operated applications. The unit employs a rechargeable battery for power; thus, adaptability is provided. The most promising experimental solid-state circuit to date employs only five transistors and has shown excellent temperature stability from +30° F to 130° F; some temperature compensation work remains to provide low temperature stability. Progress was satisfactory.

An improved, simple pulse generator circuit was developed which provides output pulses larger than the magnitude of power supply voltage of the battery operated circuit. The generator employs a General Electric Unijunction transistor as a relaxation oscillator plus one added transistor amplifier.

A study was performed to determine appropriate, simple, reliable methods for the detection and counting of microgram amounts of beryllium on standard air filters. The proposed methods were of the activation and product detection type of $(\alpha,\gamma n)$, using a suitable alpha source and of Sb^{124} (γ,n). It was calculated that a 0.01 curie alpha source or a 20 curie Sb^{124} gamma source would be required to permit detection of microgram quantities of filter-deposited beryllium. Suggested general instrumentation methods for both approaches were determined and a report was prepared.

Debugging is under way on the logic portion of the 400-channel analyzer for use with the positive ion accelerator. Current drivers for the coincident current magnetic core memory have been developed. These drivers will produce a three microsecond, 300 milliampere pulse with a 120 nanosecond risetime and a 90 nanosecond fall time. Transistors for evaluation have been received and final selection is being made. Layout for the printed circuit construction is finished. Work is now in progress on the sense amplifier for the memory.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples received during the month in accordance with current goals. The performance characteristics of the mass spectrometer were checked by daily analyses of a natural-uranium standard. The results of these analyses showed a daily variation in the measured U^{235}/U^{238} ratio consistent with the precision of an individual analysis and also showed that no significant change in bias has occurred during the previous three months.

TEST REACTOR OPERATIONS

The PCTR was operated intermittently during October. There were no unscheduled shutdowns.

The experiment to measure the neutron distribution in a PCTR cell was completed during the month.

Preliminary tests for the high 240 Pu-Al experiment were started.

The TTR was operated intermittently. There was one unscheduled shutdown due to faulty by-passing technique.

The TTR was made available to the University of Washington Graduate Center on a twice-a-week basis during the month.

A maintenance program was started to replace the safety sheet drives with improved equipment. Eject springs were added to the safety rods to decrease release and drop time.

CUSTOMER WORK

Weather Forecasting and Meteorological Service

Consultation service was rendered on meteorological and climatological aspects of 1) environmental consequences of reactor accidents or major fission product releases to RPO, 2) low altitude sampling of fallout to CR&D, 3) oxides of nitrogen release in 300 Area to IHO for FPD, and 4) design of heating system for new buses to CE&UO.

Off-site requests for data included 1) selected weather data on punched cards for Travelers Research Center, as requested by U. S. Weather Bureau, 2) atmospheric pressure data for Texas A&M Research Foundation, and 3) wind data pertinent to structural design criteria for Bonneville Power Administration.

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	82.7
24-Hour General	62	88.0
Special	174	83.9

Temperatures during October averaged a little below normal and precipitation totaled a little above. All measurable rain occurred during the first half of the month--most of it from the 11th to the 14th.

There was an unusual amount of fog for October. The number of days of occurrence (9), the total time of observance (63.6 hours), and the longest period of continuous observance (39.0 hours), were all 18-year-record highs for the month.

Instrumentation

Trial application of the Automatic Conveyor-Type Laundry Monitor System over several eight-hour shifts showed that one person, using the system, can perform the work of six persons using the usual monitoring methods. All modifications have now been satisfactorily completed and general operation has been quite satisfactory. The hand-written draft of the operation, instruction, and maintenance manual has been completed and is being reviewed. Since October 17, the complete system has operated for 16 hours per day with good performance with about 50 garments per hour being monitored and processed. The technical report was started.

The field-model continuous coincidence-count alpha air monitor, designed for Radiation Protection Operation, completed another month of satisfactory test operation.

Calibrations were completed on the Columbia River Monitor, designed for use by Radiation Protection Operation, and ranges (three) were adjusted to provide, as desired by operating personnel, values of approximately 0-10, 0-100, and 0-1000 microroentgens per hour. The integrating time constant was increased to reduce the chart-recorded statistical variations. Except for some minor additional work to provide alarming for pump failure (actually not a part of the monitor), the complete system is done.

A special amplifier circuit was designed for use with glass scintillation detector probes in use by Plutonium Process Engineering, CPD. A 40:1 signal-to-noise ratio was obtained for the probe-amplifier combination. The circuitry was installed and is performing correctly.

Final stages of design of the creep capsule data logging system to be used by Physical Metallurgy Operation for in-reactor creep measurements are being completed. Consultation was provided on the design of a differential temperature control system for the creep capsule. Specifications for a differential amplifier-integrator unit were determined and the availability and cost of such a unit were investigated. Such a unit with two percent accuracy, .3 uv/day drift, and good response to 40 uv differential thermocouple signals (1°C differential using ASA type K) can be obtained for about $400 per channel for 16 channels. The swelling capsule Minneapolis-Honeywell data logger arrived this month. It is now located in the 326 Building for acceptance testing prior to installation at the 100-KW test facility.

Calibration of micro-displacement readout systems to be used by Physical Metallurgy Operation for in-reactor creep measurements has continued during October.

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Calibration of both types of systems in use in the third generation creep capsules has been completed, including tests to determine the interchangeability of various models and the effects of varying the zero control settings on the translators.

Data obtained during the calibration of an Electromicrometer readout system has been processed using the 7090 transducer calibration program, and the results were transmitted to the customer in a memorandum report, PM 62-15.

The synthesized frequency characteristics for the Gorton tracer lathe were programmed for the GEDA analog computer. The simulation was checked out in a 10 and 100 times slowed down system. Frequency response curves taken showed the validity and workability of the system. Potential trouble due to instability of the system was observed. Checking the system with a ramp input showed that the error output system works.

A digital program for cylindrical heat transfer based on a formulation by A. L. Ruiz was revamped. The program is now faster and the time saved is $(T-1)/2$ where T = number of timesteps to be calculated by the program. In addition, a program was written to calculate coefficients for an analog simulation using variable radial distance steps. The program was compiled and is expected to be used soon.

Cost estimates for the ground water analog system were prepared and sent to Chemical Effluents Technology. Recommendation of taper pin connections throughout the system have been made for reliability purposes. This will also allow changes to be made easily to any of the passive resistor components. Purchase specifications for equipment and components are now being initiated.

A method of limiting the charge accumulated on the autoclave controller integrating capacitor during startup operations is being investigated in an attempt to improve temperature control of the FPD autoclave vessels. Methods of improving the temperature measurements on the autoclaves were also discussed with FPD personnel. For control purposes it is desirable to reduce the measurement time lag of the control thermocouple. Also, in many cases the sidewall thermocouples develop a high thermal resistance contact with the autoclave vessel and are severely influenced by the adjacent heater strips. When this happens, the thermocouple reads high, due to the radiated energy from the heaters, and may cause false excess temperature alarms.

Development and testing were completed on modifications to Fuels Development's tape punch readout for fuel element dimensions. The changes will permit the use of the readout device at the K-East underwater measurement facility.

Optics

A small (4 foot long x 2 inch diameter) wide angle borescope was designed and assembled for Process Equipment Development Operation. The periscope has a field of view of 80° and is to be used to study the rate of travel of bubbles of immiscible fluid in water.

A camera was designed for photographing the interior surfaces of two inch and 3/4 inch I.D. tubes. Components of the camera are now being fabricated in Tech Shops and in the Optical Shop. Some test photographs were taken using the set of optics for the large bore tubes.

Modifications to a new 1-1/8 inch diameter process tube borescope have been designed to permit photography using a 35 mm camera. This work is being done for Irradiation Testing Operation, IPD.

During the five-week period (September 23-October 28) included in this report, a total of 464 man-hours work was performed. The work load increased considerably over the previous month.

The work performed during this month included:

1. Machining of three NaI crystals.
2. Fabrication of four lenses for the Purex crane periscopes.
3. Modification of a borescope for Irradiation Testing Operation, IPD.
4. Fabrication of a quartz cylinder for a high temperature UO_2 pressurized test section to be used by Ceramic Fuels Development Operation.
5. Repair of four camera shutters for the Metallography Labs and for Radiometallurgy Operation.
6. Repair of an underwater periscope for 105-B Building.
7. Fabrication of seven glass bearings for waste storage tank pumps for CPD.
8. Fabrication of components of a Fuel Element Bore Camera.
9. Fabrication of components for a borescope camera.
10. Fabrication of windows for ten bell jars for Production Maintenance, FPD.

Physical Testing

Testing service work in general remained steady with a slight slack off in tube shop work. The tubes for K-reactor replacement have been delayed. A total of 2,174 tests were made on 1,891 items representing some 16,149 feet of material. Though work on tubular components has been restricted,

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testing of such components still accounts for over 25% of the material tested. A good base of customers was maintained with thirty-two different components serviced this month, representing all of the HAPO operating departments and service organizations and other AEC contractors. Advice was given on fifty-two different occasions on general testing theory and applications.

ASME Boiler Code approval was instituted and obtained for operation of the Tube Shop autoclaves. Kaiser Engineering operating personnel were retained for assistance in establishing complete and detailed operating procedures. These preparations were made to be able to complete the work required on spare PRTR process tubes.

Eddy current testing of 8,300 feet of 3/16" O.D. s/s tubing for use as instrument lead lines at the 100-N site was completed. The tubing was to have been used in the rupture-monitor system; subjected to operating pressures and temperatures. The high pressures required that minimum wall tolerances for the tubing be maintained. Concern with the removal of discontinuities of a size which would reduce wall thickness below the minimum allowable caused the instigation of the eddy current testing. Unfortunately, only one length of tubing (40 feet) was found to be free of discontinuities. The standard established was a 4 mil notch. Metallographic examination confirmed the existence of a variety of discontinuities including corrosion, carbon depletion, cracks, inclusions, cavities, laps, folds, and seams. Some cracks were found to have almost completely penetrated the tube wall. Use was made in the eddy current test, of a novel balancing system to determine phase relationships which in turn gives information as to whetheran indicated discontinuity is at the inside or outside surface of the tubing. Metallographic verification of this aspect of the test was obtained also. The defective tubing has been returned to the manufacturer and new material will be similarly tested when it arrives.

Work continued on direct visualization radiography techniques in connection with a simulated nuclear heating project being done by Ceramic Fuels. A closed circuit T.V., zoom lens and vidicon tube were used to determine its capabilities of visualizing the effects of uranium changes at high temperatures. Poor response at optimum conditions proved the need for an image orthicon tube in order to complete the experiment satisfactorily.

In the work on NPR vibratory fatigue samples ultrasonic testing was continued and fracture tests were made over a range of temperatures to observe the effects on brittle fracture. In addition, ultrasonic examination was made of a seven foot long by 18 inch O.D. pressure vessel (for hydrostatic pressure testing at Southwest Research Institute) fabricated from NPR primary loop pipe. The weld was continuously scanned on both sides for the full length of the longitudinal weld. Eleven indications were detected equaling or exceeding indications on previously tested vessels. The ultrasonic amplitude measurements were recorded along with the location of the discontinuity in the weld. The pipe from which the vessel had been fabricated contained an I.D. replacement weld made by approved procedures and should not have contained discontinuities of the size indicated. However, from the ultrasonic evidence it would appear that the discontinuities were slag pockets and not cracks. Review of available radiographs (in finished vessel and on original pipe) did not reveal slag pockets to be present.

Work was started to nondestructively and destructively analyze the mechanical and metallurgical properties of a cross-header pipe and associated Parker fittings which had been removed from DR reactor. Fluorescent penetrant examinations of 8 of 12 of these pipe sections disclosed indications which appeared to be stress corrosion cracking, particularly in the sections of pipe which had been in the center of the reactor. The cracking also occurred in highly stressed areas adjacent to welds. Radiographs of the Parker fitting welds, thread relief area, and threads were taken to determine the integrity of each area. The pipe was marked to establish orientation in the reactor and photographed. Photographs were taken of fluorescent penetrant indications and then mapped on an overlay. The Parker fittings were ultrasonically examined to determine if any cracking had occurred in the thread relief area. A hydrostatic test on one section to 5000 psi did not reveal any leaks. Sections were taken through typical cracked areas and metallographic examination confirmed the existence of stress-corrosion cracking. In some areas, the cracks extended a quarter of the way into the wall. Mechanical tests samples are being prepared for flattening tests, tensile tests, and bending and tensile tests on the Parker fittings. A field examination was also made on a H-reactor cross-header using fluorescent penetrant. A similar condition was found to exist as found on the DR cross-header; that is, evidence of stress corrosion cracking was evident on the Parker fittings and on the cross-header pipe jumper.

Analog Computer Facility Operation

The major problems considered during the month were:

1. Reactor Instrumentation.
2. NPR Simulation.
3. Cross-Spectrum Analysis Study.
4. Critical Mass Simulation.

Since the tube noise tester has shown promise in detecting tubes which will be potential sources of trouble in the analog computer, a plan was worked out with maintenance personnel to check all 12AX7 tubes in the analog computers. Since not sufficient down time was available, this has not yet been put into practice.

A miniature temperature recorder has been received and installed in the computer to provide a continuous record of the oven temperature. Several weeks of operation of the recorder show that a variation of about 8°F in the oven temperature is occurring in 24-hour cycles. The specifications call for oven temperature control within ±1°F. Since this causes variations in the computing capacitors of about 0.1%, it seems advisable to schedule some down time to correct this.

Eighty-one percent of the GEDA and ninety-two percent of the EASE equipment were in good operating condition during the month. Computer utilization was as follows:

GEDA	EASE	
112	140	Hours Up
56	28	Hours Scheduled Down Time
0	0	Hours Unscheduled Down Time
0	0	Hours Idle
168	168	Hours Total

A check of all routines and other general maintenance for the last two months was made. The check indicated that routine maintenance was completed as per schedule except for a few instances where the equipment could not be removed because of usage.

The reference power supply on the EASE kept burning out fuses. The trouble was traced to a loose connector on the back of the patchbay.

Instrument Evaluation

The Model II Scintran instruments in use at HAPO are performing satisfactorily. The scintillation alpha probes, which can be used with the instruments, have only occasional troubles with pinhole light leaks. Word was received that Instrument Laboratories, Inc. of Seattle, who fabricated the last order of 65 Model II Scintrans, is planning to fabricate and market them commercially. One such unit was displayed at the October Instrument Society of America Conference in New York.

The last two on-site fabricated scintillation, solid-state, alpha-only hand counters continue to perform satisfactorily, after about six weeks use, at Biology and at B-plant.

Evaluation tests were performed on seven scintillation combined alpha-beta-gamma hand and shoe counters being fabricated, on an order from Radiation Protection Operation, by Instrument Laboratories, Inc. of Seattle. The tests were performed at Seattle in an effort to accelerate the acquisition of the instruments. One unit, a "final" prototype is scheduled for shipment to HAPO for more complete testing before the other six instruments are finished. One of these instruments was also displayed by Instrument Laboratories, Inc. at the Instrument Society of America Conference in New York in October.

All calibration tests were satisfactorily completed on the experimental, scintillation, solid-state, chopper input, logarithmic-response area radiation monitor. The full covered range was about 1 mr/hr to 500 r/hr.

Assistance was rendered in calibrating an older-model, quasi-logarithmic response scintillation area monitor which has been in use with two other units, for several years on the front elevator at H-reactor. It was suggested that the newer models be obtained as replacements in the interests of continually upgrading the plant radiation monitoring system.

Fifteen of the new solid-state circuitry, portable, BF_3 tube neutron monitors are now in plant service.

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CHEMICAL RESEARCH AND DEVELOPMENT OPERATION

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Ground Water Temperature Studies

North of the Umtanum Ridge-Gable Butte-Gable Mountain axis several regions are known to exist which have very good hydraulic connections with the Columbia River. Water level records from wells drilled in these regions reflect almost immediately changes which occur in the water level of the river. The only locality, however, where temperature data indicate physical exchange of river water with ground water at a distance greater than several hundred feet inland is in the region between 100-H and 100-F Areas. There the river water can be detected up to one mile inland.

Disposal of NPR Decontamination Wastes

Digestion of iron-scavenged ammonium citrate waste at 85 C, simulating conditions expected in the NPR waste treatment tanks, develops a solid which exhibits poor settling characteristics. The effect of this peptized iron hydroxide on soil infiltration of ammonium citrate waste was determined in laboratory tests. A sandy aeolian surface soil retained all of the solids (340 ppm) from a waste throughput equivalent to about 9000 gal/ft^2. The 2 cm of digested waste sludge which accumulated on the soil surface slowed the rate of waste percolation from 4 to 8 gal/ft^2/day, expected for solids-free waste supernate, to about 0.5 gal/ft^2/day. In marked contrast, retention of undigested (room temperature) sludge by this soil was previously found to be poor (HW-73482).

The concentration of sulfate ion in the supernate of a simulated three-step decontamination waste was found to be 9800 ppm. Design Engineering personnel have indicated that this sulfate concentration precludes the use of concrete for underground and side-wall construction of the NPR decontamination waste crib.

Effluent Monitoring

The development work on the As-76 monitor was completed. An instruction manual and a final report are being written.

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The radioiodine monitor operated with no malfunction during the month. The equipment was left unattended for ten days, and no trouble occurred. A rupture caused an off-scale recorder reading for 1-1/2 hours. The reading prior to rupture was 1/3 full scale, and no gradual buildup occurred prior to the rupture. No estimate of the maximum value can presently be made. This experience indicates that a logarithmic count rate meter is required in the radioiodine monitor instrumentation.

Carbon-14 Decontamination Studies - Coolant Gas Drier Condensate

A program was initiated to determine a suitable method for C-14 decontamination of condensate from pile coolant gas drying beds. An initial sample of drier condensate, received from 100-KE Area, is being analyzed to ascertain the composition of this waste. The sample received had a pH of 9.2; it was slightly discolored (light tan) and had some light brown sediment. The solution was 2.3 M NH_4^+, 0.5 M $CO_3^=$, and the presence of HCN was detected when the solution was acidified. Precipitation and counting of $BaCO_3$ shows an approximate C-14 concentration of 1.5 μc/ml of condensate.

Efficiency of Charcoal in Reactor Confinement Halogen Traps

Iodine removal efficiency tests of charcoal exposed to reactor building ventilation air were completed. All charcoal samples proved to be 99+ percent efficient for laboratory-generated I-131 vapor.

Reactor Studies

During the past month, equipment outages required interruption of the addition of deionized water to experimental reactor tubes and addition of normal process water for a period of 12 hours. Substitution of process water for deionized water produced an immediate increase in all radioisotopes monitored in the effluent from the two experimental tubes. During the same outage it was found that a leaking valve was allowing a flow of about four gallons per minute of process water into the tank supplying deionized water to the two experimental tubes. The degree to which the deionized water test was compromised by this valve failure cannot be unequivocally established but the indication from analyses of the effluent water is that this failure occurred only shortly before the forced total substitution of process water for deionized water.

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Discarding data taken during the 12 hour addition of process water, the concentrations of Na-24 and Ga-72 in the effluent from the two experimental tubes increased continuously throughout the test, while their ratio remained nearly constant. Since under the conditions of this test these radioisotopes must arise in the water stream largely through corrosion of the aluminum surfaces, their concentrations in the effluent water may prove a useful index of aluminum corrosion phenomena.

If it is assumed that the gallium content of the aluminum in these tests is the average value measured in other tubes and cans, and if it is assumed that the corrosion products are released directly to the water, then an apparent corrosion rate can be computed from the Ga-72 content of the reactor effluent. Computed values were 2.8 to 5.1 mils per month and were still slowly increasing. These are very high corrosion rates compared with those found when process water is used containing a dichromate corrosion inhibitor. In this and other tests, Ga-72 concentrations in the reactor effluent water will be measured and compared with aluminum corrosion rates. If a correlation can be found, Ga-72 measurements may provide valuable aluminum corrosion rate estimates on a current basis during the course of reactor tests.

Reactor drier condensate samples from two reactors were analyzed for tritium and other weak beta emitting radioisotopes (C-14 and S-35). From these measurements it was estimated that tritium is disposed to this waste at a rate of about 2 curies per day at each K reactor and about one curie per day at the older reactors. Some C-14 and S-35 are also produced in amounts equal to 10 percent or less than that of the tritium.

FUEL PREPARATION PROCESSES

Electrodeposition of Nickel on Uranium

Scouting studies of the electrochemical characterization of uranium metal surfaces have continued with measurements of the capacitance and resistance of the cell U/0.1 M $KClO_4$/Pt. Apparently significant differences in capacitance of a particular specimen have been found to result from variations in treatment of the uranium surface prior to the measurement. Thus far, studies have been made of the relationship between capacitance and the concentration of nitric acid used to clean the surface produced by defilming in 10 M HNO_3 at 65 C, then chemically etching in a 6 M H_2SO_4, 2 M HNO_3, 0.25 M $CuSO_4$ solution at 50 C. It was found that when the final nitric acid cleaning is done

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at room temperature, 9 M HNO_3 gives a surface with minimum capacitance (compared to clearing with other acidities). If the cleaning is done at 40 C, the minimum capacitance is found for 8 M HNO_3. The latter observation appears particularly significant in view of the reports from Savannah River that the best uranium surface for nickel plating is one which has received a final cleaning in 8 M HNO_3 at 40 C.

Measurement of Nickel Plate Thickness on Uranium

Tests have shown that the thickness of nickel plating on uranium metal can be determined to ± 0.5 percent by measurement of the fluorescent X-ray intensities of four uranium lines after their attenuation by the nickel plating. The four lines give calibration curves with different slopes since their wavelengths and associated absorption coefficients are different. The method is sensitive to a fraction of a micron variation in plating thickness because the linear absorption of X-rays is an exponential function and it operates twice; once on the primary X-ray beam penetrating the nickel plate and again on the uranium fluorescence from the base metal.

Preliminary studies have been made of the use of these measurement techniques to indicate the degree to which plated nickel has diffused into base uranium metal as a result of heat treatment. The attenuation of the uranium fluorescence X-ray lines was measured for several nickel-plated uranium coupons, before and after heat treatment, and evidence was found that changes had occurred in the effective thickness of the nickel layer. Although the method has not been calibrated as yet to permit calculation of the extent of diffusion, some of the observations made in the course of these tests indicate that the uranium diffused through the nickel much more rapidly than the nickel diffused through the uranium. It is not yet known if this conclusion is supported by optical observation of diffusion zone boundaries for this system.

SEPARATIONS PROCESSES

Solid State Electrorefining of Metals

Solid state electrorefining occurs when passage of a direct current through a heated solid metal specimen causes impurities to migrate toward the anode or cathode end of the specimen. Since purification can be accomplished without melting the metal, this technique may prove attractive for purification of plutonium and other metals which in the molten state are corrosive toward most containment materials.

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Preliminary to work with plutonium, studies have been conducted with cerium traced with Fe-59. Extensive migration of iron in cerium has been achieved, in agreement with the work of Marchant, et. al.[1], and further, the use of higher current densities has been shown to yield greater purification in a shorter time. In a 21.5 hour in vacuo run with the specimen temperature ranging from 650 C to 790 C (the melting point) and a current density of about 600 amp/cm^2, the initially uniformly distributed Fe-59 assumed the following distribution:

% of Total Fe Contained in 2 cm Lengths									
(Cathode)	9.0	0.4	0.4	1.3	2.6	6.9	14.4	65.9	(Anode)

Despite cooling effects at the ends due to conduction through the electrode clamps, the iron content of 25 percent of the rod was reduced by a factor of 30 and 50 percent of the rod by a factor of 10. Cobalt-60, which was present as an impurity in the Fe-59, distributed similarly to iron.

Preliminary studies have established the feasibility of using a molten chloride salt bath as a heat sink around a metal specimen, to maintain more nearly isothermal conditions and likewise to allow higher current densities and presumably likewise faster migration rates.

Equilibria in Molten Metal-Salt Systems

To test equipment and techniques, distribution of iron-59 between molten cerium metal and molten cerous chloride was preliminarily investigated. The results showed that reasonably good precisions could be obtained for equilibrations of three hours duration at 875 C followed by separation of the salt phase with dilute acid, dissolution of the metal in concentrated HCl solution and radiochemical assay. Values for $E_s^m(Fe)$ ranged from 0.02 to 0.05. The induction furnace set-up which was used in these experiments has been transferred to a gloved box for systematic study of the distribution of metals between metallic plutonium and plutonium trichloride bearing molten chloride salts. Such studies are expected to yield thermodynamic information useful in connection with present development work on plutonium electrowinning from molten chloride salt solutions.

1. Marchant, J.D., E.S. Shedd and T.A. Henrie, "Solid State Electrorefining of Rare Earth Metals," Rare Earth Research, Nachmann and Lundin, eds., Gordon and Breach, Science Publishers, New York, 1962, p. 143-150.

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Disposal to Ground

Analyses of water samples from well 699-17-5 show the presence of beta-gamma emitters (presumably ruthenium) slightly above the detection limit of 8×10^{-8} μc/cc; this was verified by analyses of resamples. The inclusion of this well within the contamination pattern extending from the 200 East Area places the detectable limit of beta-gamma contamination to within three miles of the Columbia River. Tritium and nitrate ion concentrations above background have previously been noted in this well.

Additional core sampling attempted at the Recuplex 216-Z-9 crib was only partially successful with the recovery of a two-foot and a five-foot core. Two other subsequent sampling attempts resulted in the recovery of shorter cores. Other sampling methods, an auger bit and a split-barrel drive sampler, are being considered. Casing of the hole will be required if either of these methods is used to permit re-entry to obtain samples from successively greater depths. These methods would also be more time consuming than the rotary drill method.

Iodine Removal Processes

Further studies were performed in an attempt to explain the anomalous behavior of I-131 released in a specially-controlled Redox dissolving. Iodine-131 in the stack gas stream, filtered to remove particulates, was not efficiently held on charcoal, and more unusual, was removed in a reducing scrubber ($NH_2OH.HCl$) largely in a form other than elemental iodine. The conclusion reached from these observations and other tests to confirm the presence of other than elemental iodine is that I-131 under some plant operating situations may be "conditioned" into forms or possibly react with trace organics to give compounds which react quite differently than elemental iodine. Charcoal traps for I-131 in plant streams should be recommended only after detailed study of the forms of iodine occurring throughout the range of operating conditions anticipated.

Foam Suppression During Formaldehyde Treatment of Purex 1WW

Several non-silicone antifoam agents were tested on a laboratory scale as foam suppressants. As previously, foaming tendency was induced in synthetic 1WW by refluxing it with 30 percent TBP-Soltrol. Tributyl phosphate was found to be an effective antifoam agent. During continuous operation TBP had to be added to the reactor in 5 gm/l increments every 80 minutes to maintain nonfoaming operation; General Electric Antifoam B, the most effective

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silicone-base antifoam agent tested, had to be added in 0.5 gm/l increments every 20 minutes to prevent foaming. Polyethylene glycol reduced foaming somewhat, while heptadecanol was completely ineffective. During "foaming" operation of the laboratory formaldehyde reactor, cyclic surging of the foam occurs. It was found that if the formaldehyde was introduced above a packed section in the tower, instead of at the base of the tower as is normally done, the foam level remained steady at only about one-half the maximum height reached when surging.

Denitration of Purex 1WW with Sugar

Batch denitration of 1WW with sugar was investigated in the pilot plant denitration unit. A 2.5 molar sugar solution was continuously added to hot (100 C) 1WW. After the sugar addition was complete, the pot contents were digested at 100 C for several hours. The reaction proceeded smoothly and was easily controlled. At the highest sugar addition rate used (3.4 moles/min./sq.ft. of tower cross section), the maximum pressure drop across the tower was 1.5 inches of water and the maximum decrease in pot vacuum was about eight inches of water. In all runs, an induction period of four to six minutes was observed before the reaction started. Gentle air sparging reduced the induction period by about a factor of two.

With a feed ratio of 28 moles of free acid per mole of sugar and an 11-hour digestion period, the HNO_3 concentration of the product was reduced to 1.3 M. About 22 moles of nitrate were destroyed per mole of sugar fed and about 1.5 percent of the sugar was left in the product. At a feed ratio of 15.5 moles of free acid per mole of sugar, the product was acid deficient after six hours of digestion even though about 30 percent of the sugar was still present in the product.

A decrease in pot vacuum of 30 inches of water was produced in the pilot plant unit when 20 liters of 1WW and 2.7 liters of 2.5 M sugar solution were mixed together at room temperature and then heated rapidly to 100 C.

WASTE MANAGEMENT AND FISSION PRODUCT EXTRACTION

Purex Strontium Flowsheet Studies

Hot-cell tests of a peroxide-tartrate process for producing a strontium crude decontaminated from cerium - without the need for a byproduct oxalate precipitation - were reported in previous monthly reports. Results with actual Purex waste were very poor.

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The reason for this disappointing behavior (free radicals formed by radiation) and an effective means to prevent it (addition of ethylene glycol, a free radical scavenger) are now believed to have been found. Glycol addition was tested in two runs with well aged (ca. 200 days) somewhat dilute (95 gal. per ton) 1WW. Twenty milliliters of ethylene glocol per liter of feed were added immediately following addition of lead nitrate and about one-half hour after peroxide addition. The mixture was digested at 60 C for three hours and filtered. Decontamination factor from cerium was over 100 in the run at pH 3 and about 10 at pH 2.5, both quite saitsfactory. The experiments will be repeated with fresh, concentrated 1WW as soon as the latter is available.

A large number of supporting, tracer-level, laboratory experiments were performed during the month, mostly with FTW, in an effort to elucidate the chemistry of the process and maximize strontium recovery. Principal conclusions were:

1. That strontium recovery was primarily dependent on digestion temperatures. Losses were about 15 percent at room temperature and 9 percent at 60 C when cerium was not precipitated and even lower when cerium was also precipitated.

2. Fraction of cerium retained in solution increased with amount of peroxide added. Best separation of cerium and strontium was observed at room temperature. The cerium DF exceeded 20 at room temperature and was about 10 at elevated temperature.

3. Rate of peroxide addition is a significant variable, rapid addition giving best results. Reactions which destroy peroxide (without yielding a solution which complexes cerium) proceed more rapidly at higher temperature but are slower at room temperature than the desired reaction. Hydrazine slows down both the desired reaction and the side reactions proportionately and renders rate of peroxide addition less critical.

4. Addition of ethylene glycol to scavenge free radicals effectively stopped the action of peroxide on the tartrate. The desired peroxide-tartrate reaction was found to be essentially complete within one minute of peroxide addition.

A detailed topical report on this work will be issued as soon as the hot-cell program is completed.

Cesium Recovery by Ion Exchange

Linde AW-400 synthetic zeolite has been incorporated into the flowsheet for cesium removal from alkaline supernatant waste and has been found to be very satisfactory in the 14-30 mesh particle size range. However, the iron-containing binder presently used does not hold up well for acid wastes, especially where complexing agents such as oxalic or citric acid are present.

Cesium Packaging

The binary equilibrium system cesium-silver was determined for Linde AW-400 because of current interest in storage of cesium, recovered by the cesium nickel ferrocyanide process (HW-75051). If we assume the effect of silver and sodium to be additive, cesium loading would be approximately 0.5 meq Cs/g, from solutions containing 0.26 g Cs/l, 0.20 g Ag/l, and 0.62 g Na/l. Eliminating the silver from the above solution gives a cesium loading of 1.5 meq Cs/g for the remaining 0.26 g Cs/l and 0.62 g Na/l. The actual silver concentration in the cesium nickel ferrocyanide process is known only to be less than 0.20 g Ag/l, so additional data are necessary to define the expected significance of silver competition in the cesium loading operation.

Strontium Packaging

Shallow bed experiments were conducted to determine the strontium loading rate on Linde 4A zeolite that had been treated to replace interstitial gas with water. A 0.2 $\underline{N}$ $SrCl_2$ solution traced with Sr-85 was passed through 0.5 gram beds of 30-35 mesh 4A for varying time intervals. Flow rates were maintained to keep the effluent strontium concentrations essentially the same as the influent. By this method each 4A particle was bathed in solution having very nearly the same concentration of strontium chloride.

The loading curves obtained showed that the time to 50 percent strontium loading decreased from 17 minutes for untreated 4A to 14 minutes for 4A that had been soaked in water for 5 days. The kinetics were not significantly improved by soaking in water for longer periods of time or by using dilute Dupanol or sodium hydroxide solutions. In addition, vacuum outgassing under water for two hours showed no significant improvement in kinetics over untreated 4A.

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Cesium Solvent Extraction

Evaluation of BAMBP (4-sec-butyl-2-α methylbenzylphenol) for extraction of cesium from Hanford wastes and synthesis and testing of related compounds was the subject of intensive laboratory investigation during the month.

Compounds synthesized included BAMBP itself (to verify the technique of synthesis and purification), 4-tert-butyl-2-α methylbenzylphenol (ter BAMBP), α-methylbenzylphenol (AMBP) and its bis- and tris-isomers. The tertiary butyl compound (ter BAMBP) was found to have cesium extraction characteristics identical to BAMBP, but should be only about a fourth as expensive to manufacture as BAMBP itself based on cost of starting materials (quoted prices for BAMBP are in the range of $5 to $20 per pound, depending on size of order). The AMBP compounds either gave low cesium extraction or three-phase formation, and were not investigated further.

A variety of diluents were tested for BAMBP with the results shown in the following table. BAMBP concentration was 1 M in all cases and initial aqueous was 0.001 M Cs, 1 M NaOH.

Diluent	Cs E_a^o
Amsco - D95	25
Nitrobenzene	19
Diisopropylbenzene	100
Hexone	0.05
Shell 2342	260
CCl_4	94
Soltrol 170	245

Unlike the dipicrylamine system, there appears to be no correlation with dielectric constant. Saturation of BAMBP (in Soltrol) with cesium was observed at a ratio of 4 moles of BAMBP to 1 mole of cesium (versus a 1:1 ratio with dipicrylamine). Both observations suggest a drastically different mechanism of extraction. Scouting experiments indicated that NMR measurements may prove valuable in elucidating the extraction mechanism.

D2EHPA Solvent Extraction of Strontium and Rare Earths from PTW

Laboratory Studies - In the chemical flowsheet presently envisioned 0.03 M HNO_3 is used in the partition column to separate strontium from co-extracted rare earths. A succeeding strip column employing 2 M HNO_3 as the stripping agent is used to remove rare earths from

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the organic phase. Contrary to results obtained in mixer-settler runs, high strontium and cerium losses have been observed in pilot plant pulse column studies of the partitioning and stripping operations. Results of batch contacts suggest that the high waste losses may be the result of insufficient residence and/or contact time. In contacts simulating partition and strip column conditions attainment of equilibrium distribution ratios for both strontium and cerium required three minutes contact at either 25 or 60 C. Vigorous mixing conditions were used. The equilibrium distribution ratio for both strontium and cerium was lower at 60 than at 25 C.

To test the effects of solvent cross-contamination, extraction column feed prepared from synthetic FTW of the estimated 1965 composition, was contacted with 0.18 to 0.20 M D2EHPA - 0.18 to 0.20 M TBP - Soltrol solutions containing dipicrylamine (DPA) and nitrobenzene. (One of the methods for removing cesium from FTW involves extraction with DPA-Nitrobenzene.) The presence of as much as 0.0005 M DPA or 3.7 volume percent nitrobenzene did not affect distribution ratios significantly. At 7.4 volume percent nitrobenzene, the highest concentration tested, strontium and cerium distribution ratios were about two-fold lower.

Pilot Plant Studies - 1A pulse column studies were continued to confirm the adverse effect of some FTW component on cerium extraction. With the FTW-type feed, it was necessary to operate the 1A column at 45 C or above to insure a cerium waste loss of less than 20 percent. Scattered strontium waste losses in the range of 10 to 20 percent at elevated temperatures suggest that a higher solvent concentration or flow rate will be required to achieve the desired two percent or less strontium loss. Operation with the aqueous phase continuous, instead of the normal organic phase continuous, lowered the cerium loss about two-fold at room temperature and six- to ten-fold at 50 C. This type of operation may not be desirable, however, because of the extensive backmixing of feed components and "cruds" throughout the scrub section, undoubtedly leading to poorer decontamination performance.

Recent strontium analyses have revealed an unexpected difficulty in stripping strontium from the solvent with 0.03 M HNO_3 in the 1B column. Waste losses as high as 30 percent were obtained under operating conditions which should have been extremely favorable for strontium removal. The main difference between the current runs and several very satisfactory runs made in 1960-61 with a somewhat similar flowsheet is the much lower strontium concentration used in the present runs.

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Cerium stripping in the organic-continuous 1C column was apparently controlled by the rate of cerium diffusion in the solvent phase. An increase in frequency from 80 to 95 percent of flooding or a two-fold increase of the aqueous-to-organic flow ratio had an appreciable effect on the cerium waste loss (five-fold or greater reduction). An increase in column temperature from 25 to 40 C or a reduction of the flow rates from 560 to 270 gph/ft^2 (sum of both flows) had a lesser but still significant effect (two- to four-fold reduction). Surprisingly, neither varying the acid concentration from 0.5 to 2 $\underline{M}$ HNO_3 nor adding oxalic acid had much, if any, effect on the cerium loss. Use of 0.3 $\underline{M}$ HNO_3 did significantly increase the loss, as would be expected from the cerium distribution ratio and extraction factor.

Engineering Studies of Processes for Cesium Recovery

Extraction with Dipicrylamine-Nitrobenzene - The use of packed columns for the dipicrylamine extraction of cesium was investigated. The three-inch-diameter glass column contained nine feet of 3/4-inch stainless steel Raschig rings. An air-purge dip tube installed about six inches below the packing was used to demonstrate the effect of air agitation on extraction efficiency and flooding. The following results were obtained.

1. When extracting cesium from a simulated supernate-type feed at a solvent rate of one liter per minute, the H.T.U. ranged from 3.8 feet at a feed rate of 0.5 liters per minute to 7.3 feet at one liter per minute. Addition of air at 0.1 to 0.2 scfh lowered the H.T.U.'s to 2.3 and 3.7 feet, respectively. Addition of slightly higher air rates caused the column to flood.

2. At a feed rate of 1.1 liters per minute, the H.T.U. decreased from 3.5 to 2.3 feet with an FTW-type feed, diluted about four-fold, as the air sparge rate was increased from 0 to 0.5 scfh.

3. The sodium decontamination factor obtained under CS (scrub) column conditions, using 0.1 $\underline{M}$ citric acid as the scrub solution and a one liter per minute organic feed rate, ranged from 4.2 to 7.5 as the air rate was increased from 0 to 0.5 scfh. At a feed rate of two liters per minute and an air rate of 1.8 scfh, the sodium decontamination factor was increased to 11.

4. The H.T.U.'s under CC (stripping) column conditions, using 0.3 $\underline{M}$ HNO_3 as the strip solution and a two liter per minute feed rate, decreased from 3.5 feet to 2.7 feet as the air rate

[redacted]

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was increased from 0 to 1.5 scfh. At a one liter per minute feed rate, the H.T.U. decreased from 3.2 to 2.7 as the air rate was increased from 0 to 0.5 scfh.

The above results indicate that packed columns with provisions for air sparge agitation should be acceptable, though not as efficient as the pulse columns previously tested. The performance of the CS and CC columns would undoubtedly be improved by using smaller packing and air rates on the order of 20 to 40 scfh/ft^2. In all cases, the efficiency was improved by operating at lower flow rates, so the final columns would probably be designed to operate with the supernate feed rate at about 160 gph/ft^2 and the solvent rate at about 320 gph/ft^2. The FTW feed rate could probably be about 320 gph/ft^2.

Sorption with Clinoptilolite - Experiments were resumed to test cesium removal from simulated Purex 241-A Tank Farm supernate in fixed beds of 20-50 mesh clinoptilolite. The columns were 23 inches high by four inches in diameter. Using a cesium concentration of 5×10^{-4} $\underline{M}$ and flow rates of 1 and 1.5 gpm/ft^2, breakthrough occurred at 5 and 4.5 column volumes, respectively. As expected, it was possible to process at least twice the feed volume, i.e., an equivalent quantity of cesium when the feed was diluted to twice its original volume. When the column length was doubled, 12 column volumes were processed before significant breakthrough. Apparently, the greater the length-to-diameter ratio the more efficient the absorption.

A preliminary laboratory experiment has demonstrated that the proposed elutant for cesium, a solution of ammonium hydroxide and ammonium carbonate, can be recovered on a continuous basis. A 5000 ml pot was half filled with the solution. The solution was continuously fed into the pot and distilled off at equal rates of about 4 ml/min. The vapor temperature was 94.5 C. After processing about 1.25 liters of solution, none of the Cs-134 tracer in the feed could be detected in the distillate. Analyses are being made of the pot liquid and distillate.

Fission Product and Waste Packaging

Studies continued on the use of inorganic zeolite materials as a means of stabilizing high level waste fission products. Emphasis is currently being placed on the hot gas through-drying portion of the process cycle. Installation of off-gas humidity measurement equipment was completed with shakedown runs indicating extreme care is necessary to eliminate tramp water when measuring dew points in the range of -100 F.

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Since the particle size distribution of fines elutriated from synthetic inorganic zeolites is of interest in process and hazards evaluation, a water suspension of fines from Linde Company, Type 4A molecular sieves was analyzed for particle size distribution over a range of one to twenty microns. Samples were prepared by tumbling 1/16-inch pellets and washing the fines off with water. Greater than ninety percent of the particles lay between one and ten microns, 2 w/o were greater than ten microns, and approximately 5 w/o were less than one micron.

Additional study was made of the incorporation of cesium-loaded synthetic zeolites into vitreous solids. Results of 39 fusions using mixtures of the zeolite pellets and varying amounts of LiF, SiO_2 and B_2O_3 indicate that glass formation containing dispersed crystals can be obtained over a wide range of compositions. At a fusion temperature of 800 C, or less, the optimum zeolite concentration is between 30 and 35 weight percent. The fused product occupies approximately the same volume as the original pellets.

Strontium Titanate Packaging Studies

Use of the Dynapak (high-energy impact) process for production of high density, massive strontium titanate was scouted briefly (with Ceramic Fuels Development Operation). The process holds promise as a simple, effective means for producing strontium-90 heat sources. Two schemes were tried. In the first, commercial strontium titanate powder was packed into a stainless steel can, heated to 1200 C for 30 minutes under vacuum, and then compacted. Best density achieved by this process was 4.2 g/cc. In the alternate method, $SrCO_3$ and TiO_2 were slurried together to insure intimate mixing, dried, packed into the can, heated to 1200 C for three hours and to 1300 C for 20 minutes, and then compacted at 280,000 psi. Density of the resulting $SrTiO_3$ was 4.79, about 94 percent of theoretical. These densities compare favorably with those reported by Oak Ridge and Martin Marietta (3.0 to 3.5 g/cc) on production pellets.

Remote Welding

The remote welding program was essentially completed by investigating the effects of the included angle of the tungsten point on the weld characteristics. Twenty samples remain to be leak tested.

The five-inch pipe weld samples with raised flush face joints and V groove raised face joints were welded with tungsten points that were ground to 20°, 30°, 40°, 60°, 90° or 120° included angles. Two welds were made with each of the 20°, 40°, 60° and 120° tungstens, 13 welds with the 30° tungsten and 19 with 90° tungsten. These tests showed that the following changes occur as the included angle of the tungsten point is increased: (1) voltage decreases at constant arc length and amperage, (2) joint centerline penetration increases, (3) tungsten erosion increases, (4) arc stability decreases, and (5) arc initiation becomes more difficult. The arc shape changes from a bell shape to a cylindrical shape as the included angle increases, thereby producing localized heat at higher included angles and wide coverage heat at lower angles.

The localized heat produced by the wider included angled tungstens yields better penetration (deeper at joint centerline) but requires finer positioning of electrode to get a joint with good external characteristics and penetration at the center of joint. The wide coverage heat from the smaller included angled tungstens produces a weld with good external characteristics over a wider range of tungsten positions (± 0.005 inch), but deep centerline penetration is hard to produce as the weld beat flattens out as the heat is increased. The sharp pointed electrode produces a weld nugget with a crescent shape while the blunt pointed electrode produces an elliptical nugget shape.

To produce welds with good external characteristics and good penetration (0.090 inch) over a wide range of tungsten positions (± 0.005 inch), a sharp pointed electrode should be used, while a blunt pointed electrode should be used if deeper penetration (0.110 inch) is required and more accurate tungsten positioning (± 0.001 inch) is feasible.

Two 5-inch samples with half-inch thick end caps were welded and pressure tested. The sample welded with a 90° tungsten point failed at 6000 psi while the one welded with 30° tungsten failed at 4000 psi. The weld joint was subjected to bending and shear forces in the pressure test.

<u>Amine Extraction of Neptunium from 1WW</u>

About 97 percent of the Np(IV) in synthesized 1WW was removed in a single equal volume contact with 0.3 <u>M</u> Alamine 336 (a tertiary amine) - 2 v/o n-octonol - Soltrol. Hydrazine was used to reduce Np(V) to Np(IV) prior to the contact. Under similar conditions tri-n-lauryl amine was almost as effective, but a third phase formed. The other amines tested, tri-n-octyl amine, and two secondary amines were less effective as neptunium extractants, and also formed third phases.

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N-octonol, 2-ethylhexanol, and heptadecanol at concentrations from 0.7 to 3.3 volume percent were tried as modifiers to prevent third phase formation in amine-Soltrol-1WW extractions, but none was effective.

EQUIPMENT AND MATERIALS

Stress Cracking in Mild Steel

The three large (3 ft. x 3 ft. x 3/8 in.) weldments exposed to 50 percent sodium nitrate at about 90 C were found to have cracked during the interval between 60 and 90 days' exposure. One (and probably two) of eight large weldments exposed to synthetic Purex alkaline waste (approximately 6 M nitrate - 0.2 M nitrite) was cracked after 42 days' exposure at about 90 C. The weldment which definitely cracked was fabricated from mild steel ASTM 283, grade

At the end of this period, the sample exhibited preferential attack near the braze at a rate of about 1000 mpm. This braze is not suitable for joining 300 series stainless steels for service in nitric acid environments because of the severe galvanic corrosion which occurs on the stainless steel.

Non-Metallic Materials

Eastman Kodak's "Polyallomer" 5B22 was irradiated and tested for damage by the flex test. Damage at 10^7 and 10^8 R was severe. The radiation damage pattern was quite similar to that of polypropylene, i.e., at these exposures the samples were embrittled. Samples irradiated to 10^9 R were very flexible.

A sample of Falls Industries, Inc., "Graph-i-tite" carbon rod was soaked in molten LiCl-KCl for 168 hours. X-ray diffraction examination indicated that the salt had penetrated 15 mils into the carbon. If this material were used as a cathode in molten salt electrolysis it is probable that the deposit would quarry out some of the carbon.

PROCESS CONTROL DEVELOPMENT

Scintillating Glass Alpha Counters

A cell for contact alpha counting of solutions has been designed and fabricated for use in a prototype installation at the 234-5 Building. A scintillating glass phosphor will be used in contact with solution in the D-6 sump receiver line. Although the average plutonium concentration on the D-6 line is about 10^{-5} g/l, higher excursions to 0.02 g/l are possible. The scintillating glass alpha particle counter is expected to be adequate as an alarming device for this application.

Tests were made using thorium solutions, in which about 9 g/l of thorium produced a count rate equal to the background count rate. Based on the ratio of specific activities of thorium and plutonium, it should be possible to detect less than 10^{-3} g/l of plutonium.

The fabrication procedure for and performance of in-solution silicon PN alpha particle counters was recently reviewed with L. Cathey of Savannah River. At the present time, the scintillating glass appears to be at least as sensitive and efficient as the PN counters, it is available commercially, it does not require regeneration in service, and no immersed metallic electrode, which dictates a lower allowable solution conductivity, is required.

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Conductivity Measurements of Dissolver Solutions

Two sets of conductivity electrodes were fabricated, assembled into a probe, tested, and delivered to Redox, together with the necessary auxiliary instrumentation. This device will be used to measure the Zirflex dissolver solution conductivity as an indication of the end point of jacket dissolution.

An exploratory study was made to determine whether this probe will also provide an indication of nitric acid concentration during fuel core dissolution. Conductivity measurements were made using nitric acid - UNH solutions in the range 0-1.0 M HNO_3 and 2.0-2.5 M UNH. Similar measurements were made with various concentrations of nitric acid, aluminum nitrate, and mercury(II) nitrate. Preliminary analysis of the results indicates that the higher conductivity of the nitric acid solutions may require redesigned electrodes to give a higher cell constant. Operating experience is needed to determine the adequacy of the probe in conductivity ranges outside those for which it is designed.

Instrument Evaluation

As part of a continuing program to make current advances in instrumentation technology available for Hanford chemical process control requirements, three types of instrumentation are currently undergoing test and evaluation:

1. A commercially available solid-state recorder-controller combination has considerable versatility, in terms of the types of input signals. Accuracy and long range stability of the system are quite satisfactory. A dead band of about 200 μV at the zero input point may limit the usefulness of the instrument in certain applications.

2. Evaluation of an operational magnetic amplifier was directed toward determination of its suitability as a control system component. With proper circuitry, this instrument can be used as a controller as well as an amplifier. Frequency response, stability, and zero drift characteristics were measured to provide data from which amplifier performance for specific applications can be calculated.

3. Turbine-type flowmeters are being evaluated as a possible replacement for the rotameter instruments that are currently being used to monitor the flow of organic process streams. Two small meters (0 - 1 gpm) installed in the experimental C-column facility have each experienced 1000 hours of operation

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over a one-year period and are still in operation. Two sets of calibration data taken during this period, one in January and the other in August, show no significant differences in the calibration curves for the two periods. In contrast to this satisfactory performance, a 0-150 gpm meter failed after successive operating periods of 200, 300 and 100 hours. To date, no satisfactory meter has been found for the range of flow rates of interest for most plant applications.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Flowsheet Development - The data for the September hot-cell Salt Cycle run have been collected and reviewed. In this run, a 2.6 LiCl-KCl melt was used as reaction medium and segments of a PRTR fuel element (cooled for about three months) were used as feed. After about 30 percent of the uranium had been recovered as UO_2 by partition-type electrodeposition at 600 C, 98 percent of the plutonium in the melt was precipitated as PuO_2 by a Cl_2-O_2 sparge at 575 C. At the conclusion of the run, a heel of undissolved, caked U_3O_8 was found in the bottom of the vessel. Examination of the data leads to the conclusion that fission product decontamination for the PuO_2 precipitation was adversely affected by the presence of the undissolved U_3O_8, but that the UO_2 electrodeposition proceeded normally. Decontamination factors measured for the product UO_2 were as follows: Pu, 75; Zr-Nb, 1; Ce-Pr, 85; Ru-Rh, 20; Pm, > 250; total rare earths, 60. About all that can be said with assurance regarding the data from the PuO_2 precipitation is that little or no decontamination from zirconium can be expected beyond that which can be obtained via prior removal of zirconium with electrodeposited UO_2.

Although not yet studied, the apparent ready incorporation of zirconium into the crystalline PuO_2 precipitated in this experiment suggests the possibility of preparing crystalline PuO_2-ZrO_2 via precipitation from molten chloride salt solutions.

Mechanical Development - In tests performed in the Radiometallurgy Laboratory, vibrationally-compacted UO_2 was successfully removed from the cladding by "vibration hammering" from a one-foot long section of PRTR fuel rod irradiated to 1200 MWD/T. Hammering was done with an Ingersoll-Rand 171 air hammer which delivers up to 5000 blows/minute with a force sufficient to break up the UO_2 but

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not enough to deform or damage the Zircaloy cladding significantly. The air hammer method appears suitable for removal of UO_2 from fuel rods for Salt Cycle reprocessing.

Cerium Stand-In for Plutonium in UO_2 Deposition - Electrolyses were conducted with graphite electrodes in two-liter quartz crucibles at 530 C and 1.5 volts with equimolal LiCl-KCl initially containing 20 w/o uranium and 1 w/o cerium. The previously reported difficulty in making cerium material balances was due to systematic errors in radioactive cerium tracer analyses and sampling techniques. Chemical analysis of the cerium in both the molten salt and the UO_2 deposit gives excellent material balances, indicating that very little cerium precipitated from the salt, even with an air sparge. Based on these new analyses, feed-to-deposit decontamination factors previously reported for cerium with air sparges and air-chlorine sparges should be revised upward to a range of 4.3 to 7.6.

Additional runs made with intermittent electrolyses and air-chlorine sparges gave cerium decontamination factors in the range 3.7 to 5.0. With intermittent electrolyses the oxygen-to-uranium ratios were generally higher and the deposit surface was much smoother than with steady electrolyses. In a steady electrolysis made with a chlorine sparge saturated with water vapor, a cerium decontamination factor of 2.8 was obtained, but the crystal structure was fine and porous with no mechanical strength.

Based on comparative behavior between plutonium and cerium (laboratory experiments) the separation factor of plutonium from the uranium should be lower than that obtained and reported here for cerium. Thus, it is concluded that a co-deposit flowsheet can be designed successfully.

RADIOACTIVE RESIDUE FIXATION

Zeolite Properties

Several additional shallow bed loading rate determinations were made on the pelletized clinoptilolite samples from Minerals and Chemicals Philipp Corporation. Sample E-282-62C, for example, was fully loaded from a 0.1 M CsCl influent in approximately 14 minutes for the 0.25 to 0.50 mm particle size range, 55 minutes for the 0.50 to 1.00 mm range, and 240 minutes for the 1.2 to 1.4 mm size range. For comparison, naturally-cemented clinoptilolite requires about 330 minutes for full loading in the 1.2 to 1.4 mm size range.

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Although the pelletized clinoptilolite loading rate represents some improvement over the natural material, loading rates are still too slow to permit column use of the 1.3 mm diameter pellets.

Condensate Treatment

MPP Run 32 evaluating the decontamination ability of a commercial grade mixed bed resin (Illco TM-1) was terminated after 1800 column volumes of Purex Tank Farm condensate were treated. The steam stripper bottoms had a pH of about 7.3 and contained 20 ppm ammonium ion, 5 ppm sodium, 25 ppm nitrate and 32 ppm nitrite. These constituents began to appear in the effluent from the resin bed after about 500 column volumes of waste had been treated.

During this early period about 99 percent of the ruthenium was being removed. When the nitrite and nitrate had saturated the anion resin portion of the bed, ruthenium removal efficiency decreased and only about 90 percent was removed for the remainder of the run. Cesium began to appear in the effluent when the cation resin portion of the bed was saturated with sodium and ammonium ions. Although strontium was being removed during the entire run, the efficiency of removal increased towards the end of the run when carbon dioxide gas was added to the feed to adjust the pH to about 4.5.

Calcination of Radioactive Wastes

A second full-level spray calcination run (third full level run) was completed in the A-Cell pilot-plant-scale calciner during the month. Object was to complete filling of the pot used in the first run and to obtain meaningful baseline analytical data - since most of the liquid samples in the first runs were rendered invalid by cross contamination due to interconnections (since eliminated) in the sampling system. Feed was 54 liters of Purex acid 1WW with added sulfate (0.64 moles/liter), and the feed rate was 4 to 4.8 liters/hour (design rate is 1 gallon per hour). Because of build-up of a minor amount of ruthenium activity on the cell filters after termination of the two previous runs, shutdown procedure was modified to include a prolonged (5 hour) steam and air flush of the calciner and off-gas train after cessation of the feed cycle. This was apparently successful since no increase of activity on the cell filters was observed when the melt pot was removed from the calciner. An increase in activity on the cell filters (from 38 mr/hr to 140 mr/hr) was noted during the first half of the run. This was traced to a pressurization of the

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calciner (which normally operates at negative pressure) caused by a too-vigorous filter blow-back. Adjustment of the blow-back cycle to eliminate the pressurization halted the activity build-up.

At completion of the run, combined feed used in the two spray runs was equivalent to about 1/3 ton of uranium, and the melt pot was filled to a depth of about 6 inches with 4.8 kilograms of solids occupying a volume of 1.7 liters, giving a density of 2.8 g/cc. Fission product decay heat maintained the centerline thermocouples in the pot at a temperature of about 240 F after heating had been terminated and the pot allowed to cool.

The results of the current run and more complete analyses of the first two runs requires some revision of last month's conclusions regarding off-gas behavior. It now appears that 12 to 20 percent of the feed ruthenium is volatilized during spray calcination and finds its way to the condensate, the fraction evolved increasing somewhat as the run progresses. No significant difference is apparent between pot calcination and spray calcination with respect to fraction of ruthenium evolved or behavior in the off-gas system. Off-gas decontamination factors are shown in the following table (only ruthenium was present in appreciable concentration in the off-gas):

RUTHENIUM OFF-GAS DECONTAMINATION FACTORS

Unit	1st Run[a]	2nd Run[b]	3rd Run[a]
To and including condenser	6×10^4	6×10^5	3×10^5
Caustic scrubber	7.1	2.5	3.6
Electrostatic scrubber	3.1	1.5	2.3
Silica gel bed	1.9	3.8	3.3
Absolute filter	$>10^3$	$>10^3$	$>10^3$
Total	$>10^9$	$>10^9$	$>10^9$

(a) Spray calciner-continuous melt pot run
(b) Pot calcination run

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Eighteen-Inch Spray Calciner

Construction of the 18-inch diameter, ten-foot long spray calciner was completed during the month. Shakedown tests were underway at the end of the month.

WASTE CALCINATION DEMONSTRATION

The Chemical Development Operation has been asked by the Atomic Energy Commission to proceed with the design and development of equipment for pilot plant scale conversion of high level radioactive wastes to a solid form. This program will utilize a portion of the forthcoming Fuels Recycle Pilot Plant for simultaneous demonstration of the pot calcination process (jointly developed by the Oak Ridge National Laboratories, Phillips Petroleum Company and Hanford Laboratories) and radiant spray calciner process (developed by Hanford). The necessary space and auxiliaries are also being allowed for the possible demonstration of the rotary kiln calcination process (under development by Brookhaven National Laboratories). Major equipment pieces will include feed tanks, feed evaporator, radiant spray and pot calciners, condensers, acid fractionator, and final off-gas treatment.

Progress to date has included, in part, generation of a project proposal for the necessary funds, issuance of a comment copy of the engineering flow diagram, analysis for the critical path items involved in design, procurement and installation, and detailed analysis of the cost and expenditure pattern for the program.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

A Fourier Series solution for three-dimensional flow in a rectangular cube having a permeability distribution of the form $K = e^{ax + by + cz}$ was obtained. The series solution gives the potential, ϕ, as a function of the coordinate location (x, y, z) in the box-shaped flow system. Accordingly, the true potential can be calculated and the streamfunction defined. The solution will help test and check analog simulation methods rather than solve problems directly. Several specific applications are in determining desirable node spacing and optimum ratios of horizontal and vertical scaling in order to minimize the approximation error resulting from representation of a conducting continuum with discrete resistors.

The "Steady Darcian Flow in Soils" program was modified to include an internal means of stability control. The program now recognizes consistent instability and initiates controls that reduce the instability and cause the problem to converge. Until now recognition of instability was by visual inspection outside the machine and usually after considerable machine time was used.

About 100 pounds of clinoptilolite were obtained from a bed of the John Day Formation in central Oregon. Preliminary tests indicate that much of the rock evidently exceeds 95 percent clinoptilolite. University of California geologists who have studied the area report that the bed underlies a minimum of 500 square miles of area near Kimberly, Oregon.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Environmental Studies

Preliminary examination of trace element activation analysis of the Columbia River and its tributaries, river flow data, and reactor effluent water radioisotope concentrations indicate that the annual Spring increase in radioisotope concentrations is due to the Spokane River and lowland runoff. The Spokane River drains an area of agricultural and mining use in Washington and Idaho and generally achieves its highest flow some one to two months prior to the Columbia River peak flow. During the Spokane River peak period in 1962 the Cu-63 and Mn-55 concentrations showed increases by factors of 3 to 4. This increase corresponds (with the expected delay due to travel time) to the period of high effluent radioactivity.

Fallout Studies

Filters from air samplers located on the roof of the 329 Building have been collected almost daily since the resumption of nuclear testing this summer and have been analyzed for I-131, Zr-Nb-95, Ru-103, Ba-La-140, and occasionally for Ru-106, Ce-141 and Bi-212. (The Bi-212 is a daughter of natural thoron which is escaping from minerals in the earth.) By using a membrane type filter preceding an activated charcoal bed a separation of the particulate and gaseous material was obtained. Of the isotopes measured, all are essentially completely in the particulate form except for I-131 which appears to be largely gaseous immediately following a nuclear test but increasingly particulate in the following period. The

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gaseous I-131 appears to be in the reduced chemical forms (I^- and I_2) while the particulate I-131 appears to be about one-half in the reduced form and one-half as iodate and periodate. Some information regarding fallout rates and mechanisms in the Hanford area should be obtained on comparing these data with data obtained from samples taken at various altitudes by airplane.

Radiation Chemistry

The Electron Spin Resonance (ESR) machine was modified in order to hold two samples in the same microwave cavity. Separate magnetic field modulation is employed for these two portions of the new cavity, with separate modulation frequencies (400 cps and 100 Kc), and separate demodulation-recorder readout devices. Since one of the positions can be used for a standard it is now possible to make accurate comparative measurements on high loss samples. Although this dual sample cavity results in a reduction of the ESR signal by one-half, a full signal can be recovered by doubling the microwave power in cases where this causes no increase in noise level. In the case of samples which show saturation broadening (such as the seeds and films now under study) the power can be easily doubled.

Gamma irradiation of radish seeds produced an increased ESR signal (indicating a greater population of unpaired electrons), some of which persisted until the seeds had been heated for 30 minutes at 45 C.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition in Conduits

Several reproducibility tests and two deposition experiments were completed using the 1-1/4 inch diameter tubing. Satisfactory reproducibility for establishing the entering particle concentration was demonstrated. Because of relatively small deposition at the flow rates used initially, accurate determinations of ZnS entering and leaving the tube are necessary. Directly weighing the 4-inch molecular filters was found to yield uncertainties exceeding the allowable; hence, improvements in accuracy were sought. Phosphorescence and fluorescent measurements and more rigorous weighing techniques were explored.

For the two deposition measurements 9 percent and 14 percent deposition were measured, compared to 6 percent predicted. Improvement in accuracy is expected with the better techniques considered.

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Scouting experiments using very small tubes and the Royco size distribution counter suggest that this technique would be suitable for more rapidly developing deposition data for very small diameter tubes.

W. H. Reas
Manager
Chemical Research and Development

WH Reas:cf

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BIOLOGY OPERATION

A. ORGANIZATION AND PERSONNEL

David H.W. Liu joined the Aquatic Biology Operation as a Biological Scientist on October 25, 1962.

GENERAL

All of our experimental rats are now being obtained from Charles River Labs and are a specific pathogen-free strain of Sprague-Dawley animals. A preliminary study of the total body X-ray response of these animals suggests that the LD_{50} (30 days) will fall in the range of 750 to 850 r.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

Reactor effluent monitoring at 100-KE with young rainbow trout was terminated on October 24. The observations at the end of the test are summarized below:

Per cent effluent	Dead	Live	Total	Per cent mortality	Average wt. $\pm$ SE (g)	Average length $\pm$ SE (mm)
0	3	235	238	1.3	49.0 ± 1.1	149 ± 2.0
1	0	238	238	0	47.2 ± 0.8	147 ± 2.1
2	2	236	238	0.8	47.0 ± 1.2	147 ± 2.0
4	1	118	119	0.8	46.2 ± 1.4	144 ± 2.2

Analysis of variance on weight and length data gave no statistical significance of effluent effect on growth; however, a slight growth depression for both weight and length is suggested.

Preparatory work to test the biological effect of the presence of quachrome glucosate, a promising corrosion inhibitor, in place of dichromate on young chinook salmon is in progress for the next monitoring test of 5 to 6 months duration.

Columnaris

Field sampling of salmon and other river fish from the Spring Creek Hatchery, McNary Spawning Channel, and Burbank Area of the Columbia River, all gave mild incidents of columnaris infection, both as to number of fish and to density of columnaris infection on the affected fish.

Work to produce a synthetic medium for growing columnaris has been reactivated.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY AND TRANSFER OF RADIOACTIVE MATERIALS

Salmon

Aerial surveys were made of chinook salmon spawning in the vicinity of HAPO. In the section of river between Richland and Priest Rapids, 45 and 111 salmon nests were observed on October 15 and 22, respectively. Over half of these nests were upstream from the reactors, in the vicinity of Midway. Number of nests were about normal for this period in the spawning season.

Zinc-65

Exploratory tests to determine the amount of Zn^{65} excreted in the urine by trout after a single oral dose of the isotope were initiated. Under our experimental conditions tentative estimate of urine flow 24 hours after cannulation of the bladder is 80 ml/kg fish/day. This copious flow of urine of a freshwater fish is a contrast to the urine flow of the average normal man of about 15 ml/kg/day.

A larger fraction of Zn^{65} in reactor effluent water was removed by a 0.3 micron millipore filter than was removed from river water. A correspondingly higher exchangeable fraction was obtained from the Columbia River water.

The Zn^{65} deposited on the filter was not exchangeable with sodium acetate nor was it released by passing water with a pH of 4 through the filter.

Strontium

Processing continues on the Sr^{90} test fish killed by chlorine last April. These fish had been fed ad libitum a diet containing 0.5 μc Sr^{90}/g food for 14 weeks. The body burden determined in these groups of fish are summarized below:

	Trough 11	Trough 12	Pond 12	Total
Number of fish	25	29	62	116
Water temperature C	13	13	river temp.	
Average body burden, μc Sr^{90}-Y^{90}	69	58	36	
Standard error	4.0	3.1	3.6	

The apparent lower body burden of the pond fish compared to the trough fish is primarily due to the lower temperature of river water experienced during the experimental feeding period.

Dental tartar removed from the teeth of pigs fed Sr^{90} appears to have a high concentration of Sr^{90}, about the same as bone.

Iodine

A study was initiated to determine the estrus cycles and fertility rate of a group of 17 hypothyroid ewes which received a single 3 mc dose of I^{131} three years ago. Six controls are included in the study. A vasectomized ram equipped with a marking harness is used to determine estrus in the females. Twelve of the hypothyroid ewes and four of the control ewes have already manifested estrus. Tracer I^{131} uptake studies are now in progress.

A double tracer study employing both I^{125} and I^{131} is underway in order to determine thyroid uptake following various routes of administration.

The uptake from topical application was 5 to 10 per cent during the first week, while the thyroid uptake following subcutaneous, oral, and intravenous administration usually reached a peak by 48 hours and was in the range of 30 to 50 per cent.

Cesium

A study was initiated to determine the effects of age and exercise on Cs^{137} metabolism in male sheep. Preliminary data from whole-body monitoring of two animals restrained in metabolism cages revealed that in the older animal (3 year) 50 per cent of an intravenous dose was excreted at seven days, whereas the young animal (6 months) excreted 50 per cent during the first four days. Distinct differences in rate of loss continue to be apparent at two weeks post administration.

Neptunium

Neptunium-237 in the form of a nitrate appears to increase liver lipids to a greater degree after 24 and 48 hours than does the citrate. Liver, spleen, kidney, and brain sections from neptunium-237 poisoned animals have been examined histologically and autoradiographs are being processed. Preliminary data on lipid analyses suggests that there may be little change in the phospholipid and cholesterol fractions from livers of neptunium-237 poisoned rats.

Plutonium

Plutonium-238 and plutonium-239 citrate solutions have been tested for degree of polymerization in order to explain differences in toxicity and distribution. Filtration from Visking bags and autoradiographic study of distribution in rat liver have indicated no significant amount of polymerization. Additional studies are planned employing membrane filters.

Prompted by observations of elevated blood glucose in humans treated with DTPA, experiments were performed in rats which confirmed transient blood glucose increases within 1 to 2 hours after DTPA administration. By 24 hours post-treatment blood glucose levels had returned to normal even in animals treated with DTPA on several successive days. Insulin was effective in preventing the rise in blood glucose seen shortly after DTPA injection.

The relationship between plasma and urine concentrations of Pu^{239} following intramuscular administration of Pu^{239} was studied in two sheep in which the ureters had been surgically exteriorized to allow accurate collection of urine specimens. The intramuscular route of administration was used, since it would probably result in Pu^{239} entering the blood stream in a form most analogous to that of an industrial accident, yet still allow plasma levels that would be easily detected. Plasma concentrations were always higher than that of urine, the plasma/urine ratio increasing progressively from 4 to 5 at five hours post injection up to 60 at 48 hours post injection in one animal and 70 at 30 hours post injection in another. Although these changes could be related to the surgical treatment of the animals or other factors, it is suggested that the plasma Pu^{239} is becoming less available for kidney excretion with time following administration.

Six pigs were injected subcutaneously with 5 μc of a plutonium nitrate solution on each foreleg. Suffusing the site immediately after injection or four hours after injection with 0.017 g of Na_3Ca DTPA did not appear to effect any gross movement of Pu^{239} from the injection site or a reduction in the amount of plutonium found in the regional lymph nodes. There did appear to be slightly less Pu^{239} in the liver and skeleton than was present in an untreated pig. One pig that received 0.014 g Na_3Ca DTPA/kg body weight (IV) daily for four days following injection with plutonium had even less Pu^{239} deposited in the bone and liver than the animals that had the wound areas suffused with the DTPA salt (data on the urinary excretion of the Pu is not yet available).

Inhalation Studies

Forty days after depositing 2 mc $Ce^{144}O_2$, by inhalation, dogs appear normal, except for leukopenia.

Pluronics and DTPA caused a slight increase in the rate of excretion of inhaled plutonium in urine and feces. There was also some evidence that translocation and retention in other tissues were slightly increased.

A malignant lymphosarcoma (reticulo-endothelial type) occurred in a dog that inhaled $Pu^{239}O_2$. This dog was sacrificed 30 days post-exposure for tissue distribution analysis for a particle size study. The lymphosarcoma was probably not the result of $Pu^{239}O_2$ inhalation. This dog did not show any indication of lymphosarcoma in hematological examination or gross inspection at necropsy. This is especially significant since lymph node

The dog breeding program was discontinued due to congested quarters. This will seriously affect the availability of dogs for experiments beginning one year from now. Early completion of the new runs will permit continuation of our breeding program.

Radiation Protective Agents

LAF strain mice exposed to 950 r X-ray were "protected" by administration of rat bone marrow cells. These rat cells were either from normal rats or from rats "pre-sensitized" to LAF mice by neo-natal injection of LAF cells. Injection of 10^5 or 10^6 rat bone marrow cells afforded no protection. Injection of 10^7 rat bone marrow cells, either normal or "pre-sensitized", protected the X-irradiated mice for the 2-week period following irradiation. Six weeks after irradiation 2 our of 6 animals survive in the group treated with normal rat marrow cells. One of thes is showing a reversion to mouse type cells. Five out of 6 animals survive in the group treated with "pre-sensitized" rat bone marrow cells and all of these survivors show predominatly rat type cells.

Preparations of polymerized DNA for use in radiation protection studies were prepared from the spleen and thymus gland of LAF mice. The preparation was obtained in good yield and would appear to be comparable to polymerized DNA reported in the literature. A DNA preparation is also being obtained from an inbred strain of rats. Attempts to isolate a polymerized RNA from mouse livers has thus far proved unsuccessful.

Cellular Studies

Anaerobic glycolysis of glucose was inhibited 50 per cent when deuterated glucose was used as a substrate for cells grown on H_2O. No inhibition of glycolysis was observed when this deuterated glucose was used as a substrate for cells grown in D_2O.

Previous work with deuterated glucose in an oxidative system had shown that inhibition by this substrate was obtained only in a fully deuterated system. The nature of the difference between the anaerobic and aerobic system will be investigated.

A dose of 120 Kr of X rays produced marked increased in leakage of phosphorus and potassium from yeast cells able to carry on both oxidative and fermentative metabolism. The same dose caused almost no effect on leakage from cells having only oxidative capabilities and an intermediate effect was obtained in cells having only fermentative capabilities. There is a partial, but incomplete, correlation between the LD_{50} dose and these changes in permeability.

Plant Studies

A technique for obtaining secondary replicas of leaf surfaces is being used to determine the width of the stomatal openings. However, the stomata of the bean plant currently being used is not large enought to permit unequivocal measurement of the openings. The replication technique appears satisfactory in other respects since no plant injury is produced. Repetitive replication of the same leaf surface following different changes in the environmental conditions can be studied.

Columbia River Limnology

Counts of Columbia River plankton samples collected from the Vantage station during this year show Asterionella sp., the dominant diatom, was most abundant (approximately 900/ml) in mid-May. Melosira sp. had a pulse peak (265/ml) approximately one month later. Fragilaria sp. pulsed in late August. During the May-June period when phytoplankton increased, there was a marked decrease in concentrations of nitrate, chloride, zinc, calcium, and magnesium in the river water, while phosphate and silica decreased only slightly.

Rattlesnake Springs Limnology

Preliminary qualitative investigation of the zooplankton from Rattlesnake Springs reveals the presence of seven species of cladocerans, two or three copepods, at least three rotifers, and two ostracods. Specimens were sent to taxonomic authorities for specific determination.

Plant Ecology

The soil moisture status of the Hanford Reservation sagebrush community was determined from the critical time of seed germination and seedling establishment of cheatgrass. During the initial stages of soil moisture accumulation for the 1962-63 ecological growing season, rainfall provided 38 liters of water/m^2. Almost 60 per cent (22 liters/m^2) of it was stored in the upper 2 dm of soil profile. From October 16 to October 30, a period of no measurable rainfall, soil moisture depletion occurred at the rate of 0.45 liters/m^2/day.

Fallout

Iodine-131 concentrations in California, Maryland, and Washington deer thyroids increased markedly during the month. Weekly samples of 20 to 30 central Washington deer thyroid were obtained through cooperation with the Washington Department of Game.

Mean values were 5.0 ± 2.2 (S.D.) nc I^{131}/g wet thyroid tissue near Ellensburg and 3.8 ± 1.6 near Yakima. Comparable concentrations occurred in deer thyroids obtained in the Klickitat and Asotin regions.

Radiation Effects on Insects

An experiment was set up to study the X-ray effects on reproduction and development in populations of flour beetles cultured as single- and mixed-species populations in three different environments. The experiment was begun with 2700 adults previously sexed as pupae.

Groups of Ephestia larvae were exposed to different dosages of X ray, ranging from 1.3 to 19.5 kr, to study effects upon growth and development. At one month after irradiation, eclosions in groups that had received 1.3 and 3.9 kr did not differ from controls, however, there was a decrease in eclosions at 9.1 kr. Higher exposures inhibited development causing organisms to persist as larvae; but the larvae develcped to same size regardless of their size at time of exposure to X ray. These data indicated that at the higher X ray levels, larval systems apparently functioned properly but larval to pupal stage metamorphosis was disrupted.

HA Kornberg
Manager
BIOLOGY OPERATION

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C. Lectures

a. Papers Presented at Society Meetings and Symposiums

R. C. Thompson. October 15, 1962. Biological effects of internal ionizing radiation. Medical Symposium, Sandia Base, New Mexico.

W. J. Bair. October 17, 1962. Distribution and removal of inhaled Pu^{239} and Ce^{144}. Scientific Meeting on the Diagnosis and Treatment of Radioactive Poisoning, Vienna, Austria.

b. Off-Site and Local Seminars

None

c. Seminars (Biology)

H. A. Kornberg - Hanford Biology Program - October 23, 1962.

F. P. Hungate - I.A.E.A. Assignment in Greece - October 23, 1962.

d. Miscellaneous

None

D. Publications

a. Documents (HW)

None

b. Open Literature

Erdman, H. E. 1962. Comparative X-ray sensitivity of *Tribolium confusum* and *T. castaneum* (Coleoptera: Tenebrionidae) at different stages during their life-cycle. Nature 195: 1218.

Tombropoulos, E. G. and W. J. Bair. 1962. Treatment for removal of inhaled radioactive Ceria($^{144}CeO_2$) from the lungs of rats. Nature 196: 82.

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APPLIED MATHEMATICS OPERATION
MONTHLY REPORT - OCTOBER, 1962

ORGANIZATION AND PERSONNEL

Bonnie L. Pierce was granted a relocation leave of absence and Betty S. Hoerner has replaced her.

OPERATIONS RESEARCH ACTIVITIES

Work was begun updating an incomplete model of HAPO production-cost study. It is expected that the process of updating this work and putting it in final form will be completed over the next two months.

The study of HAPO relationship to the Tri-City Area continues as new data are acquired. Arrangements are being made to acquire data from local sources.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Fuels Preparation Department

An additional analysis was made of data from production test IP-310 to determine if a group of fuel elements, previously identified as having unusual grain structure, differed in performance during irradiation from other fuel elements in the test.

In a document co-authored with FPD personnel, and issued some time ago, a method was given for determining a reject setting for the UT-4 tester which is optimum in the sense that it minimizes total cost to the producer and the customer. This was based on an average quality level. In the event of a shift in quality, the optimum reject setting will change. Revised optima were found for several levels of quality.

Data were analyzed from a pilot plant test conducted to determine the effects on internal bond characteristics of lead plugging ingot and dingot cores, and of varying lead preheat time and time in the duplex furnace. In addition, the test design permitted a comparison of fuel elements canned in the left and right baskets.

Consulting assistance continues to be provided in interpreting output from the MERCY program which evaluates measurement error.

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Analyses were completed of data from the partially replicated, 4-factor, 3-level test previously designed as an initial step in obtaining an optimum set of operating conditions in the hot-die sizing process under development. The clear-cut results have provided much useful information to the development engineers. A second test, concerned with process variables affecting end bonding characteristics was designed and will be conducted shortly.

Irradiation Processing Department

During the first part of the month an oral progress report on the Reactor Simulation Study was presented. The analytical programs which delineate craft hours expended for designated reactor tasks by maintenance type, as well as assigned and nonassigned personnel, were presented. These reports illustrated that operations and radiation protection personnel do not use the same job or task breakdown method to describe their work performance as do the maintenance organizations. It was pointed out that until all craft personnel who are involved in reactor recovery activities indicate their activities through a system whereby standard craft sets per reactor recovery function can be established, the desired simulation model could only be based on guesstimated input data, and would, therefore, have limited use. The definition of present requirements which would permit the creation of a prototype model subject to later refinement as valid data are made available was discussed in a report as requested. Meanwhile, work continues on the development of the basic analytical tools needed for creation of the model.

Preliminary steps were taken to analyze badge and pencil data in order to evaluate the extent of agreement between the two types of measurements. Additional data are currently being processed.

A report was written presenting the results of an analysis of fuel element ledge corrosion data derived from Quality Certification fuel elements. This report will be embodied in a more complete report on this subject being prepared by IPD personnel. More extensive analyses are planned on these data, and on other data relating to different types of corrosion attack.

The complicated, theoretically determined equation which relates R values (temperature imbalance in a process tube) to tube variables presents difficulties in determining the importance of adjusting various tube variables because of its complexity. An attempt was made to simplify this using a quadratic surface. This attempt was very successful, thus permitting a far easier evaluation of the general effects on R values of changing tube variables.

Statistical consultation was given in connection with the optimum sampling of samarium poisoned balls from the population which includes balls made from different batches of material and different lots within batches.

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Data from the Tube Summary reports are being used to obtain an empirical model to predict uniform corrosion rates of fuel elements as a function of the fuel element power, and annular and hole surface temperature. The data are extensive and it is hoped that a comparison between the empirical model and other existing prediction models can be obtained.

A program was written to solve for probabilities of weld cracks in primary piping. Distributions of the crack length and frequency will be estimated from these. A new program is being written to use the raw data to enable separate analysis of different pipe sections.

Assistance was given in determining the sample size needed to estimate the condition of N-fittings with regard to failure.

Chemical Processing Department

Assistance was given in preparing appropriate tolerance statements demonstrating conformance to specifications for parts produced during the third quarter of the calendar year.

A 5-factor, 2-level experiment with 1 1/2 replicates was recommended for use in identifying the important factors influencing close tolerance machining of parts.

A comparison was made of bomb reducing agents supplied by different vendors. Button density was the dependent variable used in the comparison.

A review was made of the effect of part temperature on dimensional control of fabricated parts to evaluate the advisability of adjusting cutting tools for each part on the basis of its measured temperature.

A large number of relationships between railroad accident characteristics in the form of multiple classification frequency tables have been obtained from data supplied by the Interstate Commerce Commission. These relationships are to be used in increasing the understanding of the factors which should go into the safe packaging and shipment of radioactive materials. The results here should be combined with a complete theoretic-mechanical investigation to obtain satisfactory insights.

Relations and Occupational Health Operation

Considerable effort was devoted to preparing the General Managers' report on the 1962 HAPO Attitude Survey. In addition, a cover letter, which will explain to supervision the survey results presented in the form of IBM listings, was drafted.

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Contract and Accounting Operation

IBM machine utilization data on the business and technical routine and debug times, and on customer work, are being studied to see if the patterns from week-to-week, month-to-month, and year-to-year are sufficiently regular for prediction purposes.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HL

2000 Program

Pulse Column Facility

Work continued on the analysis of data from an experiment to estimate the organic zero shift of the gamma absorptiometer to be used for analyzing the feed stream concentrations. Several sets of data collected on the column during equilibrium conditions are being analyzed with a power spectrum estimation program in hopes of characterizing the mid-column uranium concentration variability under supposedly equilibrium conditions. To date, the power spectrum estimation has not been successful because of a single low frequency spike in the spectrum. Presence of a discrete component in the power spectrum invalidates the use of the spectral analysis estimation procedure. Current efforts are directed toward constructing a digital filter which will remove this discrete frequency from the data.

3000 Program

The first trial runs of shear-spinning theoretically designed metal blanks on a Floturn machine into certain preselected shapes having near uniform shear characteristics are in progress. Preliminary investigations show that the metal appears to be deforming as planned, but more specific results will be known when dimensional and radiographic tests have been completed.

A magnetic tape has been generated on the 7090 for the purpose of numerically controlling the experimental δ-ω lathe during an interior machining pass. This will supplement the already successfully completed exterior machining pass.

A series of meetings were held with representatives of CPD and offsite machine tool numerical controls manufacturers to discuss the feasibility of designing and controlling new lathes based on δ-ω geometry.

4000 Program

PRTR

Assistance was provided in determining how to group 32 fuel elements having differ[redacted]topic contents into four groups of eight each such that the range in the average isotopic contents for the four groups was less than a given amount.

5000 Program

Actinide Element Research

Further work was done on the problem of indexing hexagonal and orthorhombic crystals.

Division of Research

A number of Monte Carlo test cases were run on the GEM program to determine the precision with which a particular nuclide can be estimated as a function of its half life and the number and half lives of other interfering nuclides. The zero degrees of freedom case is also being programmed into GEM as a degenerate option.

Definition of the IRA II system continues. IRA 335, the calculation subroutine which uses the GEM program, was defined and is currently being programmed.

Graphing of program data to establish the sensitivity levels of various analytical procedures continues.

Further work was done on the preparation of tables and graphs for inclusion in the formal report, "Fixed Time Count Rate Estimation with Background Corrections".

6000 Program

Personnel Monitoring

Work continues on the problem of compositing urinalysis samples for the purpose of analysis. The technique requires that the individual urine samples be analyzed when the composite indicates a possibly high individual result. The optimum composite sample size depends on the distribution of d/m values and on the risk one is willing to take that an occasional sample above permissible limits will not be detected.

General

Further work was done on the analysis of mass spectrometer data on three gas standards. Previous analyses using a components of variance model resolved the total variation in the data into a between run component, individual peak component and experimental error component. The between run component was estimated from the whole series of approximately 50 runs for each standard and hence measured the long term variation over several months. Since gas standards are updated periodically the data are being reanalyzed to get an estimate of the short term between run component on a within-week basis.

OTHER ACTIVITIES

J. E. Schlosser and W. L. Nicholson attended a symposium on Application of Computers to Nuclear and Radiochemistry at Gatlinburg, Tennessee, on October 17 - 19, 1962. Their paper jointly authored with F. P. Brauer, "Quantitative Analysis of Sets of Multicomponent Timed Dependent Spectra From Decay of Radionuclides", was presented by Nicholson.

W. L. Nicholson visited the Department of Preventive Medicine of the University of California Medical School, Los Angeles, California, on October 15 to discuss the use and development of machine programs for performing statistical calculations.

R. Y. Dean attended the regularly scheduled meeting of the Numerical Systems Group of the IMOG which was held at the Dow Chemical Rocky Flats plant on October 23, 24, and 25.

A paper concerned with analytical control procedures within CPD was prepared and presented by D. F. Shepard at a combined meeting of the Seattle section and the Richland subsection of the American Society for Quality Control.

ORIGINALLY SIGNED BY
CARL A. BENNETT
Manager
Applied Mathematics

CA Bennett:dgl

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Economic Incentives in Fuel Element Development

Fuel element fabrication costs are responsible for a substantial portion of the total fuel cost in mills/kwh$_e$ for most power reactors. The fuel costs may be reduced by reducing the fabrication cost and by extending the fuel lifetime. Unlimited extension of the fuel element lifetime will ultimately increase fuel costs by virtue of the increased capital charges necessary to extend the lifetime, and an optimum exposure yielding minimum cost will usually exist. Many fuel cycle surveys examine the fuel costs at the minimum cost point, even though the associated optimum exposure may be beyond the fuel durability that currently can be achieved in practice. The practical situation frequently encountered is that the fuel fabrication costs are too high, and the attainable fuel exposure is too low. Furthermore, providing fuel durability to withstand higher exposures increases fuel fabrication costs. Fuel development programs are usually directed toward reducing fabrication costs and concurrently increasing fuel durability. The relative incentive to accomplish either or both of these has been calculated using the MINIMIZER code for a sample PWR.

The results listed in Table I show the fuel cost at optimum exposures, and at some arbitrarily assigned attainable exposures. In cases where the attainable exposure is less than the desired optimum exposure, the incentive for improving the durability of the fuel is derived by subtracting the fuel cycle cost at optimum exposure from that at the attainable exposure. (If the attainable exposure is higher than the desired optimum exposure, there is no incentive for improving fuel durability; this is indicated in Table I as a negative incentive.)

The incentive to increase the attainable exposure can be compared with the relative incentive to reduce the fabrication cost. For a fabrication cost of $80/lb. U and an attainable exposure of only 5,000 MWD/T, lowering the fabrication cost to $60/lb would effect savings of 1.15 mills/kwh, but increasing the attainable exposure with no reduction in fabrication cost would effect savings of up to 3.38 mills/kwh. For the same situation and an attainable exposure of 25,000 MWD/T, lowering the fabrication cost to $60/lb would still result in a significant cost reduction of 0.33 mills/kwh, but the incentive to strive for still higher exposure has shrunk to 0.06 mills/kwh.

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TABLE I

FUEL COSTS[1] AND INCENTIVES FOR INCREASING EXPOSURES[2]

Fabrication Cost $/lb U	Optimum Exposure (MWD/T)	Minimum Cost	Exp[3] = 5,000 Cost	Exp = 5,000 Incentive	Exp = 10,000 Cost	Exp = 10,000 Incentive	Exp = 15,000 Cost	Exp = 15,000 Incen.	Exp = 20,000 Cost	Exp = 20,000 Incen.	Exp = 25,000 Cost	Exp = 25,000 Incen.
10	15,900	1.95	2.47	0.52	2.02	0.07	1.95	0.00	1.97	-0.02	2.03	-0.08
20	19,100	2.16	3.05	0.88	2.35	0.18	2.19	0.03	2.16	0.00	2.20	-0.04
40	24,700	2.52	4.20	1.67	2.95	0.46	2.66	0.14	2.55	0.02	2.52	0.00
60	29,400	2.84	5.35	2.51	3.64	0.80	3.15	0.31	2.94	0.10	2.85	0.01
80	33,200	3.12	6.50	3.38	4.29	1.17	3.63	0.50	3.32	0.20	3.18	0.06

(1) Cost = Fuel cycle cost, mills/kwh_e, for a sample PWR type reactor.

(2) Incentive for increasing exposure = cost with currently attainable exposure minus cost at optimum exposure. (This quantity is expressed as negative when attainable exposure exceeds optimum exposure.)

(3) Exp = Postulated maximum exposure, MWD/T, currently attainable.

Generalized Nuclear Parameters Study

The results of this study have been reported previously as the investigation of key factors in plutonium value calculations (Programming Monthly Report - June 1962). In this work, the relative importance of plutonium's nuclear properties is to be determined by varying them and comparing the resulting plutonium value with that obtained with the standard properties. Previous work included varying σ and ν (of both Pu-239 and Pu-241 together) separately and simultaneously in three reactor types. Note that varying σ implies that σ_a, σ_c, and σ_f were varied in the same ratio so that alpha was preserved in all of these calculations.

The study was extended this month by calculating variations in σ_f in the APWR with σ_a and ν held constant. These results, together with some of the previous data, are shown in Figure 1 wherein the value ratio V/V_o is shown as a function of η for the fissile plutonium as supplied to the recycle irradiation step. η is calculated by the following formula:

$$\eta = \frac{\nu^{49} N^{49} \sigma_f^{49} + \nu^{41} N^{41} \sigma_f^{41}}{N^{49} \sigma_a^{49} + N^{41} \sigma_a^{41}}$$

where

ν = average number of neutrons released per fission (which is assumed to be independent of the incident neutron energy).

N = isotopic concentration, nuclei/barn-cm.

σ_f = effective fission cross section, barns.

σ_a = effective absorption cross section, barns.

The superscript 49 denotes Pu-239 while 41 denotes Pu-241. Note that these parameters are those of the plutonium initially in the first recycle step.

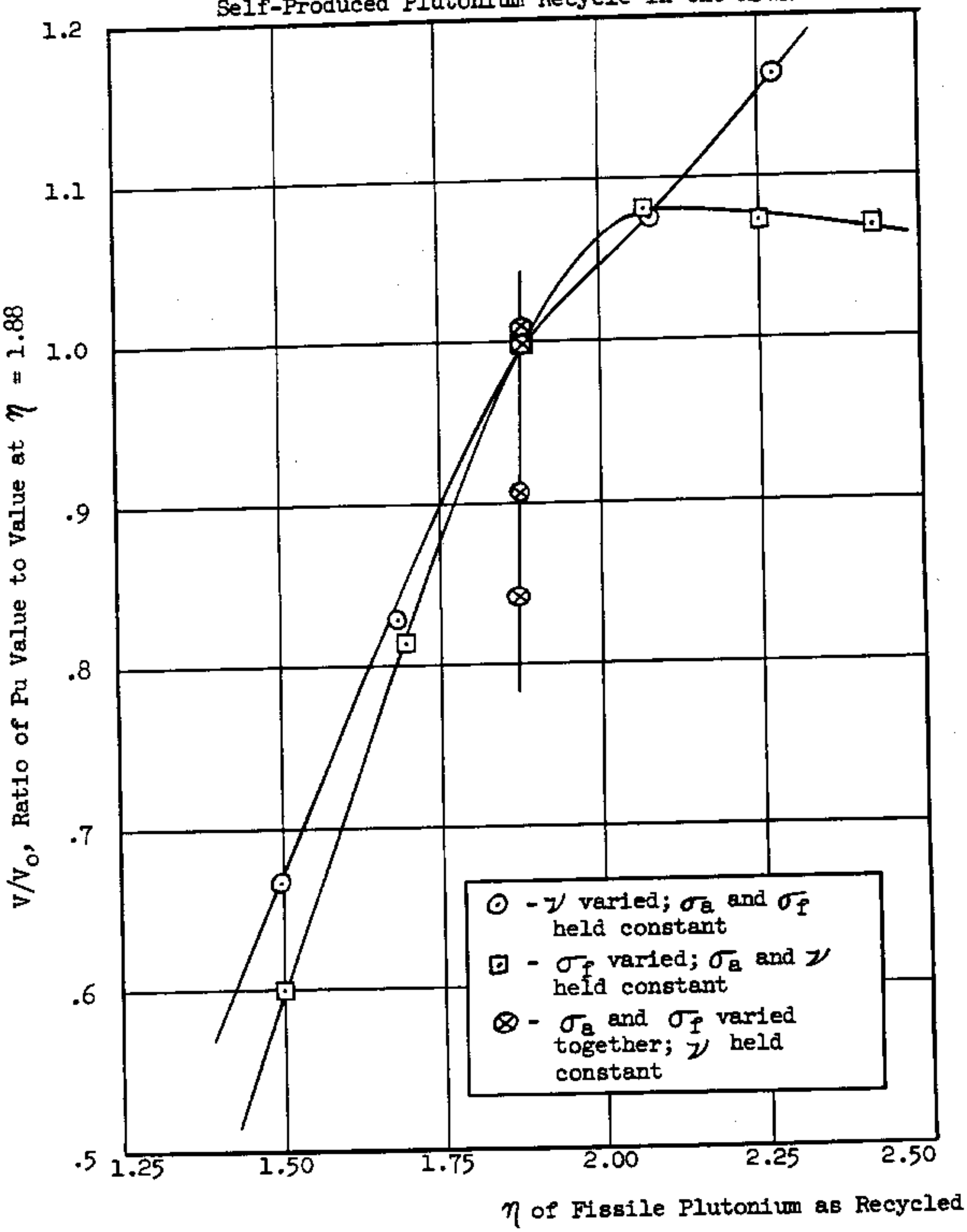
The plutonium value V is, in the previous work, the PUVE 3 step indifference solution, while at present it is a two step indifference solution.

It is evident from the figure that η is not adequate to correlate the data -- especially for variations in σ wherein α is held constant. Otherwise, the most striking result is that there appears to be an optimum nonzero value of alpha. The reason for this effect is undoubtedly that if Pu-239 had no capture cross section, the reactor would be denied the well-known benefits of Pu-240's fertility (i.e., phoenix-like action).

FIGURE 1

PRELIMINARY RESULTS OF THE
GENERALIZED NUCLEAR PARAMETERS STUDY

Self-Produced Plutonium Recycle in the APWR



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Reduction of U-238 Spatial Concentrations

Table II indicates areas that have recently been explored in the Reduced Density study. Areas checked are those for which fuel costs have been obtained.

TABLE II

		Batch		Graded	
SDPV	SCA	15 MW/T Specific Power	30 MW/T Specific Power	15 MW/T Specific Power	30 MW/T Specific Power
2	1	x	x	Being Computed	x
2	2	x	x	May Be Computed	x*
3	1	x			
3	2	x	x	May Be Computed	x
5	1	x			
5	2	x			

* Only a few economic cases have been prepared.

SDPV is the moderator index and SCA is the resonance shielding index. Eighteen sets of economic parameters have been used for each set of physics parameters.

In the September monthly report, representative fuel costs were reported for batch operation of a reactor fueled with plutonium "rich" in the Pu-240 isotope. The isotopic composition of the plutonium was 45 percent Pu-239, 40 percent Pu-240, 10 percent Pu-241, and 5 percent Pu-242. This was diluted with varying amounts of tails uranium to provide various fuel densities ranging from normal oxide density to one-tenth of normal density. Fuel costs reported were for a specific power equivalent to 15 MW/T at normal oxide density. During October, computations were made of fuel at twice the specific power (30 MW/T) which reduced the fuel costs about 0.3 mill/kwh. Furthermore, at this high specific power, the density at which optimum fuel cost occurred was shifted towards higher density in all cases studied. In the case of low self-shielding (very small diameter fuel elements of large surface to volume ratio) lowest fuel costs were still obtained at 50 percent density. Generally the results for the graded fuel cycle appear very favorable for reduced density fuels. Graded operation showed lowest fuel costs at ten percent density and maximum fuel costs at normal oxide density. Graded operation fuel costs were less sensitive to fuel density variations than were batch fuel costs. Graded fuel costs are given in Table III for 12½ percent AEC use charges. For the current AEC use charge of 4-3/4 percent, the fuel costs are much less.

TABLE III

Density Percent of Normal Oxide Density	FEFJ(1) ($/cc)	AEC Interest (%)	Minimized Fuel Cost(2) (mills/kwh)
10	1.22	12½	0.816
33	1.22	12½	0.849
50	1.22	12½	0.896
66	1.22	12½	0.927
85	1.22	12½	0.965
100	1.22	12½	---

(1) $1.22/cc is approximately $60/lb at normal oxide density.
(2) No startup or shutdown charges are included in these costs.

In some of the low density cases almost complete fuel burnup occurred. To achieve such complete fuel burnup in any single fuel element may entail large regional variations in the specific power.

This amendment is necessary because fuel cycle surveys cannot properly treat the many conditions that must be balanced to ensure proper performance of a reactor. Maintenance of reactivity is paramount, but once it is established other factors, in particular, proper heat distribution over the reactor must be attained. Furthermore, the heat distribution must be preserved as the irradiations proceed. Various fueling concepts, batch, graded, and zoning schemes, each has intrinsically different responses with respect to these factors.

In batch fueling all fuel is placed in the reactor at the same time and discharged at the same time, but it may be shuffled periodically within the reactor. In graded fueling the fuel is charged and discharged more or less continuously. The apparent degree of continuity need not be great if fuel is reshuffled on occasion and discharge of 1/5 to 1/3 of a reactor at a time with fuel reshuffle is sufficient to achieve graded rather than batch performance.

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The graded fueling system has special heat generation characteristics by virtue of the fact that the reactor flux level is kept constant to maintain constant reactor power. However, within a given fuel element the isotope concentrations vary from the initial to the final value, and there is thus a variation in specific fuel power through the reactor. In reality, the large power variations in the fuel associated with the computations may be impractical. However, the incentives for reduced U-238 concentration graded cycles are not lost if fuel exposures are terminated on the basis of specific fuel heat generation rather than reactivity.

Code Development

The ALTHAEA code described in September was written as a chain as an expedient to allow early testing. It was very slow because all data were saved on tape and reread each time a diffusion calculation was required. Tape manipulation is slow relative to computation speed. The code has now been rewritten and debugged as a subroutine of MELEAGER and all data are retained in machine memory, which eliminates tape reading during the calculations. Comparative cases have not been run but the new version is estimated to be four times faster than the old.

The REPORT GENERATOR code is being changed to (1) present a more concise and self-explanatory summary of the economics and value calculations, and (2) edit the chain output to obtain a more complete record of the physics calculations. When completed, the REPORT GENERATOR code will be able to prepare a complete but concise record of each chain calculation in a form that will constitute a report in itself.

Power Reactor Fuel Reprocessing Economic Studies

Work on the Reprocessing Economic codes was largely confined to refining input data. A few more "debugging" runs were made, and both codes appear to be in satisfactory operating condition.

This work has now reached the stage where future activity is more strongly dependent upon reliable information from the laboratory studies of the Salt Cycle process than upon consistent analysis of over-all fuel cycle economics. Accordingly, the work is being transferred to the Chemical Research and Development component of the Hanford Laboratories and subsequent reports will be issued by that component.

GENERAL

A report describing the major research and development accomplishments of the Hanford Laboratories during the past five years was prepared in support of the Laboratories participation in the Accent-on-Value program.

W. K. Woods

Manager,
Programming

WK Woods:jm

RADIATION PROTECTION OPERATION REPORT FOR THE MONTH OF OCTOBER 1962

A. ORGANIZATION AND PERSONNEL

Transfers within the Section during the month included Charles D. Pringle transferring from Environmental Studies and Evaluation to Radiation Monitoring, and Donald G. Morton transferring from Radiation Monitoring to Environmental Studies and Evaluation. Stanley D. Waggoner (new-hire) joined the Internal Dosimetry Operation.

B. ACTIVITIES

Occupational Exposure Experience

Eleven new cases of plutonium deposition were confirmed by bioassay analyses during October. Internal deposition of plutonium, estimated to be less than one percent of the permissible body burden for each of the ten CPD employees, occurred during the transfer of recycle material from a 234-5 Building hood to PR cans in late August and early September. Exposure to air-borne plutonium contamination without respiratory protection was the cause for the intake. The other new deposition case, also estimated to be less than one percent of the MPBB, resulted from a plutonium contaminated minor injury. The total number of plutonium deposition cases that have occurred at Hanford is 310, of which 227 are currently employed.

The total number of new deposition cases for 1962 through the month of October is 27. This may be compared with a total of 17 new cases for the same period in 1961.

Four CPD employees received minor injuries to the hands that were examined with the plutonium wound counter. Minor contamination was detected in three of the wounds with a maximum activity of about 9×10^{-4} μc plutonium (~2percent of the MPBB for bone). No medical action was required as the result of the plutonium contamination. Two of the contaminated injuries were sustained by employees at the 234-5 Building, a maintenance worker and a laboratorian, and the other contaminated injury was received by a laboratorian at the 222-S Building.

Spontaneous ignition of plutonium waste material stored in a vault at the 234-5 Building tripped the fire alarm circuit, but the exact location of the fire was at first unknown. In attempting to locate the fire, two employees opened the door to the vault without respiratory protection. The door was closed immediately when it was apparent that the fire was in the vault. Only minor floor contamination was detected outside of the vault. Special bioassay samples were requested from the two employees and from three other individuals who were in the room when the vault was opened.

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A spark apparently ignited a small amount of inflammable solvent that was being used to clean plutonium carbide specimens as they were broken from fiberglass mounts in a hood at the 308 Building. A hood glove was punctured to permit entry of a fire extinguisher. No spread of contamination was detected. Special bioassay samples were requested from the HLO employee involved.

An HLO employee dropped a plutonium carbide specimen contained in a plastic bag as he was transferring it from a shipping container to a hood in the 308 Building. The plastic bag ruptured, resulting in small areas of floor contamination to 5×10^6 d/m with general floor contamination of 1000 d/m in the adjoining room. An air sample in the immediate area indicated about 3×10^{-8} μc Pu/cc, using a one-minute sample collection time. Nasal smears from one of the two HLO employees were slightly above background levels.

During the month there were 13 incidents in CPD facilities and 2 incidents in HLO facilities that required special bioassay sampling for plutonium analysis for 24 employees.

Three IPD employees were exposed to high dose rates on the rear face of the 105-DR reactor during charging operations. A large number of tubes had been flush discharged for maintenance and tube inspection. A mechanical procedure used to verify that the tubes were empty had been conducted on all except 12 of these tubes. One of these tubes was not completely discharged. When a new charge of metal was introduced at the front face, irradiated metal was forced into the capped rear nozzle. Activation of the high dose rate alarm system on the rear face of this reactor resulted in immediate evacuation by the three employees from the rear face. As determined by the HM Chamber, the dose rate at 20 feet from the nozzle was about 30 r/hour. Evaluation of the film dosimeters worn by the employees showed a maximum dose of 0.4 r. Pocket ionization chamber readings and a review of previous work activities for the badge dosimeter period indicated that the dose received in the incident was about 0.2 r. Vacation absence of the shift supervisor for the work prevented the investigation committee from determining the reasons why the verification procedure was not used for 12 tubes.

Regular processing and evaluation of a finger ring film dosimeter worn by a Biology scientist at the 108-F Building indicated a hand dose of 9 rads, including 8.5 r gamma dose. This exposure exceeded the Hanford operational control which limits extremity dose to 8 rems in a four-week period. The ring dosimeter was worn while the employee was opening a plastic vial containing approximately 7 mc of Na^{24}. The vial was contained in a small lead shipping cask. Dose rate measurements obtained during the job did not support a dose of this magnitude, but calculations have indicated that the dose rate may have been higher than observed with the survey instrument because of the small beam area.

In an attempt to increase the efficiency of a vacuum system used for decontamination on the X-level at the 105-KE reactor building, the exhaust of a vacuum cleaner was connected to the line. Backpressure at the filter in the vacuum line actually resulted in pressurization and leakage of contaminated air to the room. Examination at the Whole Body Counter of seven IPD employees exposed to the air-borne contamination showed transient internal deposition of Co^{60}. The maximum quantity observed was 0.48 μc or approximately 5 percent of the permissible body burden. The intake for one other employee was approximately the same and five individuals received less than one percent of the permissible body burden.

An open valve permitted fission product washwater to spray from a funnel leg as the solution was being transferred from a cask to a storage tank in the 325-A Building. Floor contamination in the vicinity was 220 rads/hour. Nasal contamination of about 2000 d/m was detected for an HLO employee, but examination at the Whole Body Counter indicated that no internal deposition occurred.

Chemical decontamination of the PRTR primary system piping required extensive radiation monitoring as decontamination efficiency was measured and evaluated at thirty-four control points following each decontamination step. On three occasions backflow of contaminated liquid waste during pump-out operations spread fission product contamination to 15 mrads/hour over 100 square feet in the reactor hall and C cell. Thirty-seven cases of personnel contamination varying between 400 c/m to 10 mrads/hour were detected and successfully decontaminated. Draining of "dead legs" in primary piping involved whole body and extremity dose rates of 500 mr/hour and 4 rads/hour, respectively. Metallic fines reading 100 mrads/hour at one inch were removed from the system. Dose rates in A cell following the decontamination ranged from 1 mr/hour to 50 mr/hour while the lower access space ranged between 200 mr/hour and 1.5 r/hour.

Environmental Experience

The average concentrations of fallout materials in air at the ten Pacific Northwest locations was 5 μμc Beta/m^3 during the four-week period of September 22 to October 19, 1962. The average for the week ending October 26, 1962, was 15 μμc/m^3, the highest weekly average obtained since October 1958. The maximum weekly result was 23 μμc Beta/m^3 noted at both Walla Walla, Washington, and Lewiston, Idaho. A milk sample representing the Pomeroy area and collected on October 25, 1962, was found to contain 190 μμc I^{131}/liter. A sample collected southwest of Walla Walla the same date contained 45 μμc I^{131}/liter of milk.

Local milk samples for the calendar month of October averaged about 40 μμc I^{131}/liter for commercial milk and about 60 μμc/liter for milk sampled at local farms. Exclusion of the high average of 110 μμc I^{131}/liter for the Ringold farm would lower the average farm concentration

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to ~50 μμc/liter. This difference between farm and store is easily accounted for by radioactive decay during processing and shipping.

A total of 325 biological and produce samples were obtained for radiochemical analysis. They include:

Fish	114 samples	
Ducks	29 samples	
Geese	1 sample	
Oysters	2 samples	4 pounds
Ground round	2 samples	4 pounds
Vegetation	57 samples	114 pounds
Beef thyroids	50 samples	50 sets
Milk	83 samples	201 gallons

Fifty-six duck heads have been donated by local hunters in the duck head sampling program.

Two aerial surveys were made: one covered standard patterns in the vicinity of the Plant; the other was to the vicinity of Astoria, Oregon, to measure radiations from the Columbia River and its shores. Higher-than-average activity noted on previous surveys between McNary Dam and Arlington was determined to be on islands and amounted to about twice that observed over the water.

Studies and Improvements

Radiation Protection Operation participation in the design of the Fuels Recycle Pilot Plant continued during the month. The design is now essentially complete. Only 11 prints remain to be approved out of a total of 316 prints.

Two Hanford reactors were studied to determine the ratio of the gamma-to-neutron dose rate at the front face and in the front face work area. Gamma-to-neutron dose rate ratios from about 0.8 to about 10 were observed. In general, the lower ratios were observed near the top of the reactor front face and the higher ratios near the bottom.

The solid state neutron dosimeter fabrication contract with Battelle Memorial Institute was extended to December 1 because of a delay in the delivery of high purity silicon to Battelle. Additional energy response characteristics were plotted for 0.075" silicon diodes read with the constant voltage readout technique. The response of the individual diodes varied by about 25 percent over the neutron energy range studied, 0.2 to 16 Mev.

The linearity and sensitivity of a Li-6 sandwich neutron spectrometer was measured for neutron energies from 1 Mev to 5 Mev. The response of the spectrometer was linear within ±5 percent over this energy range. The sensitivity was on the order of 10^{-6} counts per incident neutron. Work is continuing to define the linearity and sensitivity of this spectrometer detector at other neutron energies.

Some 25 BF_3 tubes for the new neutron monitoring instruments (HFQ) were checked for sensitivity and compared to the original 10 BF_3 tubes. No major difficulty was encountered in obtaining tube reproducibility.

A re-examination of the neutron sensitivity of the standard CP survey instrument indicated that the CP neutron sensitivity for a plutonium fluoride spectrum was less than 2 percent of the beta-gamma sensitivity (neutron dose rate in rems/hour and gamma dose rate in r/hour). For slow neutrons, the CP sensitivity appears to be less than 5 percent of the beta-gamma sensitivity.

Seven Scintrans and 20 neutron monitoring instruments were added to the portable instrument pool during October. The prototype scintillation portable poppy was completed. Drawings detailing the fabrication of the instrument and procurement specifications for its purchase are being reviewed prior to issue in a bid package for off-site fabrication. The CP circuit and layout drawings were redrawn to provide improved blueprint reproduction. The CP procurement specifications were rewritten. The annual physical inventory of radium and uranium was completed and verified. Four Emergency Monitoring Kits were serviced. The 3745 Building stop watches were recalibrated and repaired as necessary. A total of 108 portable instruments received an audit calibration as part of the Calibrations quality control program.

Room 1 in the 3706 Building was equipped with counting equipment and glass rod dosimeter readers for emergency evaluation of the criticality section of the personnel film badge dosimeter and for evaluation of the foils contained in the Hanford Criticality Dosimeter. Revised procedures for the use of this equipment are being prepared.

Three circuit changes were designed to provide improved performance in the automatic densitometer. The first modification is expected to increase the periods between maintenance of the pay number display, which previously has failed to display all numbers after processing about 10,000 badges. The second change permits reading film exposed to doses to 800 mr without manual switching of the digital voltmeter in the automatic densitometer. The third circuit revision will increase the lifetime of the relay contacts in the payroll number matrix. Some pitting and damage of these contacts has occurred.

The Hanford rubber finger ring dosimeter design was modified to provide improved comfort to the wearer and still further improved reliability of finger ring dosimeter performance. Procurement of the newly-designed finger ring dosimeter is scheduled for December.

An air sample counter is being calibrated for use in standardizing air sample standards used throughout the Hanford Plant. The equipment is designed for use by Calibrations personnel, and will provide, on a routine basis, a recalibration of all air sample counting standards currently used at Hanford.

The characteristics of a new purchase of HV-70 filter paper were examined. Penetration tests for this paper were requested from Industrial Hygiene.

Additional air samples were obtained from the 234-5 Building for extensive analysis as part of the air contamination characterization studies. Several of the samples were submitted for electron microscope examination. Delivery of the Goetz aerosol spectrometer is now scheduled for November 5.

The PRTR stack monitor required maintenance for the I^{131} sampling components. A complete check of the particulate components by Maintenance forces was also completed. Written procedures for maintenance checks by instrument technicians on a shift basis were prepared.

The modifications to the Automatic Columbia River Monitoring Station (ACRMS) have essentially been completed and all systems are in routine operation.

Air sampling stations were established at Washtucna and Eltopia. Both have a particle filter and charcoal filter. With the improved network of air sampling and ground survey plots, the glass wool program for sampling of particles was discontinued.

Contracts with veterinarians in Moses Lake, Wenatchee, and Toppenish for the collection of beef thyroids were renewed for another year.

Low-level contamination in a vendor's tank truck used to transport liquid alum to HAPO was found to be due to natural thorium.

The program of examining children for I^{131} in their thyroids continued. In October, 11 examinations were conducted for eight children. Most of these examinations were made on October 27. The results varied from 54 to 97 pc I^{131}.

The feasibility of having a television receiver in the Whole Body counting cell during the examination of children was tested. Feasibility was established for I^{131} examinations by the almost rigid posture assumed by the children and an increase of about 5 percent in the apparent background counting rate for I^{131}. The use of the TV is not feasible for low-level counting with the Whole Body Counter due to the presence of relatively large amounts of K^{40} in the TV set. Channel-by-channel analysis of the data for the Whole Body Counter has not been completed.

Outfitting the mobile Whole Body Counter continued during the month. The heating and air circulation system was installed and made operational. The simulated Norwegian maple wall panelling was installed.

The selection, order, and format was provided to Data Processing for revision of the output reports from the film dosimeter processing program. Revision of the previous reports was required to make the reporting system compatible with the input data modifications resulting from the new film badge dosimeter and automatic densitometer; to increase the data and information retained in the radiation dose records system; and to respond to the requests for changes in the report formats made by radiation monitoring components. A program was also designed to provide for automatic inclusion of previous accumulated Hanford exposure for re-hires and repeat visitors.

The 300 Area Emergency Plan was completed and is being distributed.

C. VISITS AND VISITORS

Visitors consulting with members of the Radiation Protection Operation during the month included:

R. S. Landauer, Jr. - R. S. Landauer, Jr. and Company, Matteson, Illinois

C. Allday, T. Hughes, B. F. Warner, F. Butler - United Kingdom Atomic Energy Authority, Windscale, England

R. Sironen - Vallecitos Atomic Laboratory, Pleasanton, California

S. Fukuda - Tokai Laboratory, Tokai-mura, Ibaraki-ken, Japan

Members of the Radiation Protection Operation visiting off-site during the month included:

R. F. Foster - Participated in meeting of Special Awards Committee of the Pacific Northwest Pollution Control Association, Portland, Oregon.
- Consulted with C. M. Everts of the Oregon State Board of Health and J. Wilson of the U. S. Public Health Service, Portland, Oregon.
- Attended meeting concerned with implications of using nuclear powered devices in aerospace vehicles at the Naval Research Defense Laboratory, San Francisco, California.

A. R. Keene - Conducted National Committee on Radiation Protection and Measurements' Subcommittee 7 meeting in Boston, Massachusetts.

F. L. Rising - Discussed fabrication of instruments at the Instrument Laboratories, Seattle, Washington.

H. V. Larson - Attended information exchange on low energy X-rays at the Savannah River Plant, E.I. duPont de Nemours & Co., Inc., Aiken, South Carolina.
- Toured film dosimeter processing operation at Lexington Signal Depot, Lexington, Kentucky.

D. N. Brady - Attended Eighth Annual Bioassay and Analytical Chemistry Conference, Augusta, Georgia.

R. F. Foster)
J. K. Soldat) - Attended meeting of Pacific Northwest Pollution Control Association and presented papers; Salem, Oregon.
R. B. Hall)

D. RELATIONS

One suggestion was submitted by personnel of the Radiation Protection Operation during October. No suggestions were adopted; one was rejected. Five suggestions are pending evaluation.

Safety meetings were held throughout the Section during the month. Topics for these meetings included electrical hazards, fire and fire safety, and driver reaction time. Safety and housekeeping inspections were also conducted.

The new film badge dosimeter was explained to Biology Research Managers. Thirteen IPD personnel attended a session of the Disaster Radiation Monitoring course. To date a total of 316 persons have attended the course.

E. SIGNIFICANT REPORTS

HW-74307-9 - "Radiological Status of the Hanford Environs for September 1962" by R. F. Foster.

HW-75459 - - "Monthly Report for October 1962 - Radiation Monitoring Operation" by A. J. Stevens.

HW-SA-2609 - "Elements of Emergency Planning for Coping with a Serious Radiation Accident" by A. R. Keene, C. M. Unruh, G. E. Backman and L. A. Carter.

HW-SA-2629 - "Management of Radioactive Effluent Gases at Hanford Atomic Products Operation" by J. K. Soldat.

HW-SA-2748 - "Evaluation of Discharging Radioactive Wastes into Fresh Water Streams" by R. B. Hall and R. H. Wilson.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

External Exposure Above Permissible Limits	October	1962 to Date
Whole Body Penetrating	0	3
Whole Body Skin	0	3
Extremity	0	3
Hanford Pocket Dosimeters		
Dosimeters Processed	6,578	32,914
Lost Results	0	0
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	20,128	106,259
Results - 100-300 mrads	475	3,152
- 300-500 mrads	56	298
- Over 500 mrads	10	106
Lost Results	145	386
Average Dose Per Film Packet - mrad (ow)	10.23	12.88
- mr (s)	32.65	29.84
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	1,442	15,869
Results - 50-100 mrem	4	29
- 100-300 mrem	2	43
- Over 300 mrem	0	2
Lost Results	15	95
Fast Neutron		
Film Read	355	4,085
Results - 50-100 mrem	8	430
- 100-300 mrem	65	724
- Over 300 mrem	0	66
Lost Results	15	46
Hand Checks		
Checks Taken - Alpha	81,316	387,853
- Beta-Gamma	61,721	528,807
Skin Contamination		
Plutonium	34	257
Fission Products	89	524
Uranium	0	12
Tritium	0	0

Whole Body Counter	Male	Female	October	1962 to Date
GE Employees				
Routine	17	1	18	154
Special	16	0	16	185
Terminal	2	0	2	92
Non-Routine	19	4	23	230
Non-Employees	6	2	8	77
Pre-Employment	0	0	0	8
	60	7	67	746

Bioassay	October	1962 to Date
Confirmed Plutonium Deposition Cases	11	27*
Plutonium - Samples Assayed	226	3,294
- Results Above 2.2×10^{-8} μc/Sample	33	207
Fission Product - Samples Assayed	162	3,793
- Results Above 3.1×10^{-5} μc/Sample	0	15
Uranium - Samples Assayed	259	1,673
Biological - Samples Assayed	2	249
Strontium - Samples Assayed	0	299

Tritium Samples	Maximum	Count	October Total
Urine Samples > 5.0 μc/l	6.6 μc/l	4	
< 1.0 μc/l		46	
Samples Assayed			108
D_2O Samples			
Moderator	0	0	
Primary Coolant	0	0	
Reflector	0	0	
Samples Assayed			0
Other Water Samples			
3217-37-08	.1694 μc/ml		28
299-W-24-14			
(10-15-62)			
			136

Calibrations	Number of Units Calibrated	
	October	1962 to Date
Portable Instruments		
CP Meter	1,042	10,092
Juno	273	2,767
GM	576	5,552
Other	155	1,829
Audits	108	1,047
	2,154	21,287

*The total number of plutonium deposition cases which have occurred at Hanford is now 310, of which 227 are currently employed.

	Number of Units Calibrated October	1962 to Date
Personnel Meters		
Badge Film	972	15,154
Pencils	-	12,670
Other	490	4,276
	1,462	32,100
Miscellaneous Special Services	530	9,343
Total Number of Calibrations	4,146	62,730

A. R. Keene lw

Manager
RADIATION PROTECTION

AR Keene:ljw

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FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

Preparation of the FY 1963 Midyear Budget Review continued during the month. Personnel requirements were submitted to Contract Accounting with the following forecast by quarter-ending dates:

		Exempt	Salaried	Total
Actual	6-30-62	699	813	1 512
Actual	9-30-62	691	929	1 620*
Estimated	12-31-62	708	937	1 645
Estimated	3-31-63	700	944	1 644
Estimated	6-30-63	716	946	1 662

*Includes 140 maintenance employees transferred from FPD on September 1, 1962.

Estimated inventory balances at June 30 1963 for Spare Parts and Reactor and Other Special Materials were also compiled and submitted to Contract Accounting as a part of the Midyear Review. Preparation of the Midyear Review will be completed in November.

In connection with a reorganization within the Chemical Development Operation, effective November 1, 1962 the following organization titles and code changes were made:

New Codes and/or Titles

7624 Process Control and Equipment Development Operation
7625 Materials and Process Chemistry Operation
7626 Waste Calcination Demonstration Operation
7627 Process Engineering Development Operation

Cancelled Codes

7622 Chemical Engineering Development Operation
7623 Process Equipment Development Operation

New program codes established during the month were as follows:

Code	Title
.37	Waste Calcination Demonstration
.55	PRP - Other R & D Support

During the month, 02 Program Equipment funds in the amount of $50,000 were transferred from Fuels Preparation Department to Hanford Laboratories. This action will provide FY 1963 funds for the Laboratories' Maintenance function which was transferred from FPD on September 1, 1962.

General Accounting

Following is a summary of the status of letters or agreements covering specific actions requiring AEC concurrence:

AT-256	Participation in Standardizing Activities - (S. H. Bush)	Approved October 1, 1962
AT-262	Washington State Scientific and Advisory Committee - Selective Service System - (I. H. Dearnley)	Approved October 22, 1962

Travel activity continues to exceed the FY 1962 level, but is lower than in FY 1961, as shown below:

Number of Trips Started - FY to Date

FY 1961	463
FY 1962	369
FY 1963	442

Travel costs, however, are less in FY 1963 to date than in the like period last year (as shown below). During the first four months of FY 1962, costs of relocating new hires and transfers were unusually high. Such costs are substantially lower this year, more than offsetting higher costs for other types of travel expense.

Travel Expenses Charged to Cost - FY to Date

FY 1962	$113 543
FY 1963	110 168

Hanford Laboratories' completed Plant and Equipment investment at October 1, 1962, totaled $70.2 million. Investment by area is:

Area	Investment (in thousands)
100-B	$ 96
100-D	2 172
100-F	3 443
100-H	24
100-K	765
200-E	1 312
200-W	5 095
300	54 289
600	994
700	415
1100	3
White Bluffs	449
Off-Site	1 193
	$70 250

Hanford Laboratories' material investment at October 1, 1962 totaled $25.8 million as detailed below:

	(In thousands)
SS Material	$24 142
Reactor and Other Special Materials	1 344
Spare Parts	341-1)
	$25 827

(1- Includes a reserve of $79,000 established October 1, 1962.

The value of nuclear materials consumed in research this fiscal year to October 1, 1962 is $2.3 million, of which $2.2 million are applicable to Hanford Laboratories and $0.1 million to Fuels Preparation Department. The following is a detail by prpgram for Hanford Laboratories' portion:

	(In thousands)
2000 Program	$ 658
3000 Program	359
4000 Program	1 157
	$2 174

Survey 20 - Part 2, witnessed verification of HAPO inventories of enriched uranium, tritium and enriched lithium as of the end of November, 1962 will be made by a team consisting of HOO-AEC and Nuclear Materials Operation personnel.

A study was started during the month to mechanize the control of all items (including movable, uncataloged and expense items) and materials located at the Storage facility.

Responsibility for taking Hanford Laboratories' plant and equipment inventories was transferred to Contract and Accounting Operation effective October 15, 1962. This move also entailed the transfer of one nonexempt employee. The identification and tagging of equipment previously assigned to this clerk have been retained. In connection with reassignment of inventories to Contract and Accounting Operation, a revised schedule of physical inventories of movable cataloged equipment for FY 1963 was prepared and forwarded to Hanford Laboratories' section managers.

Responsibility for completing utilization of Projects AEC-167 PRTR and CAH-822 Pressurized Gas Cooled Loop Facility was transferred from Construction Engineering and Utilities Operation to the Hanford Laboratories during the month.

Laboratory Storage Pool activity for the month of October 1962 is summarized below:

Equipment	Current Month Qty	Current Month Value	FY-to-Date -1) Qty	FY-to-Date -1) Value
Beginning Balance	1 251	$703 107	1 081	$562 200
Items Received	173	119 035	418	301 831
Items Withdrawn by Custodian	18	9 448	46	28 623
Equipment Transfers	9	1 608	50	23 888
Items Disposed of by Excess			5	134
Items Disposed of by PDR			1	300
Equipment on hand at 10-31-62	1 397	$811 086	1 397	$811 086-2)

(1- FY-to-date totals include minor corrections on information reported in previous months.

(2- Includes 139 items valued at $64,559 which were on loan at 10-31-62.

During the month 47 items valued at $25,344 were loaned or transferred in lieu of purchase. A total of 99 items valued at $52,000 has been redirected to useful purposes this fiscal year in lieu of purchase. Estimated operating costs for the same period were $8,000, indicating a net saving of $44,000.

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One hundred fifty-four items valued at $101,040 located at the Laboratory Pool are in the process of being retired by means of a PDR or Declaration of Excess. These items will be retired from record during November business.

Material on hand at the Laboratory Storage Pool at month end included the following:

Material	Quantity	Total Value
Beryllium	1 035 grams	$ 631
Gold	2 211 grams	3 184
Silver	6 815 grams	477
Platinum	6 997 grams	19 942
Clean Scrap	544 grams	1 600
Contaminated Scrap	6 703 grams	15 551
Palladium	2 202 grams	2 686
Zirconium	4 846 pounds	79 319
		123 390
Additional material held for convenience of others		164 431
Total material held at the Pool		$287 821

Total investment of equipment and material in custody of the Laboratory Storage Pool at month end is $1,098,907.

Action during the month on projects is indicated below:

New Money Authorized Hanford Laboratories

CAH-916	Fuels Recycle Pilot Plant	$35 000
CAH-958	Plutonium Fuels Testing and Evaluation Laboratories	13 500

Physical Completion Notices Issued

CAH-927 Additions to Waste Demonstration Facility - 271 Building (AEM Service Only)

New and revised OPGs issued in October are listed below:

OPG No.	Title
22.3.1	Approval Authorizations
33.10.1	Health and Safety
55.6.3	Internal Committees

OPG No.	Title
11.2	Index
3.2.4	Overtime
7.8	Control of Documents Classified Secret and Confidential
22.3.2	Hanford Laboratories University Relations Council - (Cancellation)

The following contracts were processed during the month:

MRO-52	Philips Electronics and Pharmaceutical Industries Corp.
SA-248	W. E. Welsh
SA-249	D. R. Marble
SA-250	W. H. Harris
SA-252	L. M. Bodie
SA-247	Future Farmers of America
DIR-157	Battelle Memorial Institute
SA-255	Pennsalt Chemical Corporation

Personnel Accounting

The following employees received patent awards during the month of October:

Name	HWIR No.	Title
L. C. Amos	1497	A Composition of Matter
R. L. Moore	1406	(An Integrated Process for the
L. A. Bray	1406	(Recovery and Purification of
F. P. Roberts	1406	(Multi-Kilocurie Quantities of
E. J. Wheelwright	1406	(Fission Product Strontium from
		(a Purex Crude Strontium Con-
		(centrate
H. L. Libby	1346	A Multiple Parameter Eddy Current Nondestructive Testing Device

Number of Hanford Laboratories Employees

Changes During Month	Total	Exempt	Nonexempt
Employees on payroll at beginning of month	1 619	691	928
Additions and transfers in	22	10	12
Removals and transfers out	11	4	7
Employees on payroll at end of month	1 630	697	933

Overtime Payments During Month	October	September
Exempt	$ 4 696	$ 4 361
Nonexempt	23 884	24 173
Total	$ 28 580	$ 28 514

Gross Payroll Paid During Month		
Exempt	$ 655 917	$ 666 988
Nonexempt	507 562	612 321
Total	$1 163 479	$1 279 309

Participation in Employee Benefit Plans at Month End	October		September	
	Number	Percent	Number	Percent
Pension	1 466	99.3	1 464	99.1
Insurance Plan - Personal	395		397	
- Dependent	1 229	99.8	1 215	99.7
U. S. Savings Bonds				
Stock Bonus Plan	157	43.0	157	43.3
Savings Plan	74	4.5	77	4.8
Savings and Security Plan	1 124	88.9	1 126	89.6
Good Neighbor Fund	1 169	71.7	1 150	71.0

Insurance Claims				
Employee Benefits	Number	Amount	Number	Amount
Life Insurance	0	$ 0	0	$ 0
Weekly Sickness and Accident	10	849	7	599
Comprehensive Medical	53	3 876	46	3 585
Dependent Benefits				
Comprehensive Medical	110	10 733	93	9 261
Total	173	$15 458	146	$13 445

TECHNICAL ADMINISTRATION

Employee Relations

Ten non-exempt employment requisitions were filled during October with 14 remaining to be filled.

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Professional Placement

Advanced Degree - There were no Ph.D. applicant visits to HAPO during October, however, four offers were extended. No acceptance or rejection activity occurred. Five offers are currently open.

BS/MS - Five program offers and five direct placement offers were extended. Offers accepted: one program and three direct placement. Offers rejected: three program and two direct placement. Current open offers: five program and two direct placement.

Technical Graduate Program - Two Technical Graduates were placed on permanent assignment. Three new members were added to the rolls; there were no terminations. Current program members total 60.

Technical Information

Eighty pages of the Hanford Classification Guide (HW-37965) were revised to reflect changes which have been authorized by the AEC.

ECONOMIC EVALUATIONS

A draft of a fairly comprehensive report on nuclear power cost estimating and the influence of electric utility economics was completed and distributed for review.

The logic of the working capital cost calculation in the FEFC (Fuel Element Fabrication Cost) computer code was again reviewed and clarified for formal reporting of the computer program.

FACILITIES ENGINEERING

At month's end Facilities Engineering Operation was responsible for 11 active projects having total authorized funds in the amount of $2,191,100. The total estimated cost of these projects is $8,595,000. Expenditures on them through September 30, 1962 were $1,352,000.

The following summarizes project activity in October:

Number of authorized projects at month's end	11
Number of new projects authorized	0
Projects completed	2

CAH-927, Additions to the 271-CR Building Waste Treatment Demonstration Facility
CGH-951, A-C Column Facility - 321 Building

New projects submitted to the AEC ------------------------------ 2
CAH-985, Addition to the 222-U Building
CAH-986, 300 Area Retention Waste System Expansion

New projects awaiting AEC authorization ----------------------- 4
CGH-974, Analog Simulation Facility
CAH-982, Addition to the Radionuclide Facilities
CAH-985, Addition to the 222-U Building
CAH-986, 300 Area Retention Waste System Expansion
Project proposals complete or nearing completion -------------- 3
CGH-991, Waste Calcination Demonstration in the FRPP
Graphite Machine Shop
Neutron Calibration Facility

Pages appended to this report provide detailed project status information.

Services

Satisfactory progress was made in the engineering services provided on the following jobs:

Equipment procurement including issuance of 11 requisitions totaling $8,500. The material and equipment list for one project was revised
Biology Operation animal quarters air flow
Biology controlled environment facility study
General purpose analog simulation facility study
108-F Source handling facility electrical changes
108-F Intercommunication system proposal

Pressure system assistance was provided on:

The design and specifications of the Dynamic Materials Test Apparatus unit
Physical Testing Operation autoclaves
Quartz glass UO_2 test section
Third party inspections of nineteen pressure vessels and one test loop.

Plant Engineering effort was expended on:

231-Z alarm system standardization
231-Z proposed paging system
325 shielded analytical laboratory direct current power supply
3705 increased electrical service
329 lighting layout for proposed annex enclosure

1236335

309 audio alarm system design
329 switchgear testing
325 vacuum pump replacement
308 ventilation modifications
321 tank farm alarm system
3702 building ventilation
307 basin transfer pump
329 loading dock
309 critical mass air dryer
329 counting room air conditioner
306 fume exhaust problem

Facilities Operation

Landlord costs for September were $141,674 which represented 88 percent of the anticipated expenditure. Improvement maintenance costs were $15,068.

The following tabulation summarizes waste disposal operations:

	August	September
Concrete Barrels Disposed	8*	8*
Loadluggers of Waste Disposed	37**	49**
Crib Waste	270,000 gal.	230,000 gal.

* In the past the number of loadluggers of barrels rather than the actual number of barrels have been reported. In the future the actual number of barrels will be reported.

** Total loadluggers disposed. Previously reported figures were 325 waste disposal only.

The Wye burial ground was closed on October 3, 1962 pending AEC review. A new burial trench was dug at the 300 N burial ground. The trench was put into use on October 8, 1962. During the month two more trenches were staked out. One is a small trench to accommodate the burial of a decontamination cell from 327 Building, the other is to accommodate routine waste from the 300 Area. The 300 N burial ground was staked for extending the perimeter fence.

During the past month Waste Disposal and Decontamination supported the PRTR decontamination effort in three ways. 1) Thirty-one trailer loads (119,000 gallons) of decontaminating solutions were hauled from 309 Buiding to a special trench at 216 BC burial ground. This trench was especially built for this effort. 2) Twenty-four truck loads (120,000 gallons) of PRTR waste were processed through 340 Building. 3) Miscellaneous hardware such as caps was decontaminated in 325 Building.

A trailer was loaned to 321 Building for the removal of special solutions during the height of the PRTR decontamination runs.

Building operation during October included (a) continued shakedown of the filter plant (315 Building) with plant operation at an average flow of 600 GPM to 309 throughout the month, (b) a ventilation balance of the shielded analytical facility (325-B Building) during the month.

Drafting

The equivalent of 119 drawings was produced during the month for an average of 29 man-hours per drawing.

Major jobs in progress are - PRTR "As-Builts", Shim Rod Control (PRTR), Electrical Resistivity Sample Holder, Cladding Cutter Assembly for PRTR, Refractory Compounds Research Glove Box, Two Hi-Temp Vacuum Furnaces, Process Tube and Fuel Handling Carriage, PRTR Corrosion Test Facility, Inhalation Studies Hood, Scintillation Scanner, FRPP Rack Design, Fuel Element Identification System (PRTR).

Construction Supervision

Activity during the month on construction work (J. A. Jones Company) being performed for Hanford Laboratories components is given below:

	Unexpended Balance
Orders outstanding beginning of month	$129 723
Issued during the month (inc. suppl. & adj.)	43 777
J. A. Jones Expenditures during month (incl. C.O. Costs)	98 570
Balance at month's end	74 930
Orders closed during month	83 450

Maintenance work orders total seven with face value totaling $35,966.

Construction and maintenance activities completed during October included:

222-U room renovation and equipment installation
292-T replacement of small lead cave with larger one
308 dock and storage facility construction
308 room 208 and 212 decontamination
309 office construction
314 installation of roll-up doors
325 ceramic research room construction
325 second floor office construction

W Sale
Manager
Finance and Administration

W Sale:whm

SEMI – MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. – Hanford Laboratories

HW- 75376

DATE 10-31-62

PROJ. NO.	CAH-822
TITLE	Pressurized Gas Cooled Loop Facility
FUNDING	4141 Operating
AUTHORIZED FUNDS	$ 1,170,000
DESIGN $	43,000
CONST. $	1,127,000
AEC $	15,000
GE $	1,155,000
COST & COMM. TO 10-14-62	$ 1,147,639 (GE)
ESTIMATED TOTAL COST	$ 1,170,000
STARTING DATES – DESIGN	8-15-59
STARTING DATES – CONST.	10-17-60
DATE AUTHORIZED	7-25-62*
DIR. COMP. DATE	12-31-62
EST'D. COMPL. DATES – DESIGN	4-29-60
EST'D. COMPL. DATES – CONST.	12-31-62
ENGINEER	TRAO-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	98	93
PF	1.4	0	0
CPFF	22.3	100	95
FP	6.6	100	100
Gov. Eq.	69.7	100	93

SCOPE, PURPOSE, STATUS & PROGRESS

The new heater has been installed.

Loop pressure tests are scheduled for early in November.

Vendor is changing to Teflon impregnated bearings similar to those provided for Dragon Project.

* Initial authorization date was December 18, 1958.

PROJ. NO.	
TITLE	
FUNDING	
AUTHORIZED FUNDS	$
DESIGN $	
CONST. $	
AEC $	
GE $	
COST & COMM. TO	$
ESTIMATED TOTAL COST	$
STARTING DATES – DESIGN	
STARTING DATES – CONST.	
DATE AUTHORIZED	
DIR. COMP. DATE	
EST'D. COMPL. DATES – DESIGN	
EST'D. COMPL. DATES – 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

1236339

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75376

DATE 10-31-62

PROJ. NO.	CGH-857
TITLE	Physical & Mechanical Properties Testing Cell - 327 Bldg.
FUNDING	0290
AUTHORIZED FUNDS	$ 460,000
DESIGN $	45,000
CONST. $	415,000
AEC $	
GE $	460,000
COST & COMM. TO 10-14-62	$ 347,345
ESTIMATED TOTAL COST	$ 460,000
STARTING DATES — DESIGN	11-2-59
STARTING DATES — CONST.	2-12-62
DATE AUTHORIZED	9-22-61*
DIR. COMP. DATE	12-15-62
EST'D. COMPL. DATES — DESIGN	3-15-61
EST'D. COMPL. DATES — CONST.	12-15-62
ENGINEER	FEO - DL Ballard

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE	9	515
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING	.4	40

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	80	78
PF			
CPFF	18	80	70
FP			
Equip.	82	80	80

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facilities for determining physical and mechanical properties of irradiated materials, and involves the installation of a cell in the 327 Building.

Current estimate of Title I and II costs - $60,000. Detailed design started 4-1-60. Procurement and construction authorized 9-22-61.

Project estimate of engineered equipment. $253,000**
Value of orders placed. (Approx.) $210,000

Field installation of piping and electrical items is continuing.

Shop fabrication of Instron machine support and positioning equipment is nearing completion.

Shop fabrication of cell tray has been started.

Fabrication of ventilation exhaust filter holder has been completed. The cell equipment hydraulic lift system was received October 22, 1962, and is being installed.

The Impact Tester was received on October 22, 1962.
All four master slave manipulators have now been received.

* Original authorization for design was October 1, 1959.

** Includes delivery charges, inspection and contingency.

1236340

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-75376

DATE 10-31-62

PROJ. NO.	CAH-866
TITLE	Shielded Analytical Laboratory - 325-B Building
FUNDING	61-a-1
AUTHORIZED FUNDS	$ 700,000
DESIGN $	60,000
CONST. $	640,000
AEC $	546,500
GE $	153,500
COST & COMM. TO 10-14-62	$ 142,874(GE)
ESTIMATED TOTAL COST	$ 655,000
STARTING DATES — DESIGN	9-5-59
STARTING DATES — CONST.	6-15-61
DATE AUTHORIZED	5-31-60*
DIR. COMP. DATE	11-15-62
EST'D. COMPL. DATES — DESIGN	11-14-60
EST'D. COMPL. DATES — CONST.	10-21-62
ENGINEER	FEO - RW Dascenzo

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	0	2571
COST PLUS FIXED FEE	5	80
PLANT FORCES		8
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	10	100	100
AE-TIT. II	90	100	100
CONST.	100	100	99
PF	3	2	2
CPFF	2	1	1
FP	95	100	100

SCOPE, PURPOSE, STATUS & PROGRESS

This project will allow greater capacity for analytical work involving today's more highly radioactive solutions and consists of adding a shielded laboratory to the 325 Building.

Plant forces are building an area beta-gamma monitoring device and checking the ventilation system.

A Corning Glass Works representative has repaired the moisture stain on viewing window No. 3.

The CPFF Construction Services Contractor installed the rheostat dimming switch, the twelve master slave manipulators and six hinged cell-trays. They have also converted two tanks to shipping casks. They have a leak in the roof to repair yet.

* Original authorization for preliminary design was August 12, 1959.

1236341

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-75376

DATE 10-31-62

PROJ. NO.	TITLE	FUNDING
CAH-867	Fuel Element Rupture Test Loop	58-e-15

AUTHORIZED FUNDS			COST & COMM TO 10-14-62	$ 572,028(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	EST'D. COMPL. DATES	
DESIGN 8-1-60	CONST. 11-2-60	8-17-62*	DESIGN 3-15-61	CONST. 10-31-62
		DIR. COMP. DATE 10-31-62		

ENGINEER: TR&AO-MEEO - PC Walkup

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	0	2475
COST PLUS FIXED FEE	0	2400
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	91	100	100
AE-TIT. II			
CONST.	100	100	99
PF	2	100	50
CPFF	57	100	98
FP (1)	10	100	100
(2)	31	100	100

SCOPE, PURPOSE, STATUS & PROGRESS

(1) G. A. Grant Company
(2) Lewis Hopkins Construction Company

This facility is to be used for duel rupture behavior studies with respect to physical distortion and rate of fission product release.

Construction work has been suspended for Loop Design Tests.

Project will be closed out October 31, 1962 with accruals for completion of exceptions.

* 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	CONST.		DESIGN	CONST.
		DIR. COMP. DATE		

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

1236342

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-75376

DATE 10-31-62

PROJ. NO.	TITLE	FUNDING
CAH-922	Burst Test Facility for Irradiated Zirconium Tubes	62-k

AUTHORIZED FUNDS $ 29,600 — DESIGN $ 29,600 — CONST. $ — AEC $ — GE $ 29,600

COST & COMM. TO 10-14-62 $ 29,600 — ESTIMATED TOTAL COST $ 289,000

STARTING DATES: DESIGN 11-7-61, CONST. 12-15-62 — DATE AUTHORIZED 10-23-61 — DIR. COMP. DATE

EST'D. COMPL. DATES: DESIGN 5-31-62, CONST. 11-1-63

ENGINEER: FEO - DL Ballard

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER - Bovay Engineers		260
DESIGN ENGINEERING OPERATION		260
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	57	100	100
AE-TIT. II	43	100	100
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facilities to permit deliberate destructive testing of irradiated zirconium tubing. This will provide operating and tube life data not available because of the limited operating history of Zircaloy-2 pressure tubing in reactors.

The project proposal was submitted to the Commission on July 2, 1962 and is awaiting approval.

There has been no activity on this project during this report period.

PROJ. NO.	TITLE	FUNDING
CAH-927	Additions to the 271-CR Building Waste Treatment Demonstration Facility	61-j

AUTHORIZED FUNDS $ 92,000 — DESIGN $ 11,000 — CONST. $ 81,000 — AEC $ 76,300 — GE $ 15,700

COST & COMM. TO 10-14-62 $ 14,888(GE) — ESTIMATED TOTAL COST $ 92,000

STARTING DATES: DESIGN 6-5-61, CONST. 2-15-62 — DATE AUTHORIZED 5-15-61 — DIR. COMP. DATE 10-31-62

EST'D. COMPL. DATES: DESIGN 2-15-62, CONST. 10-31-62

ENGINEER: FEO - DL Ballard

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		370
COST PLUS FIXED FEE	2	290
PLANT FORCES		
ARCHITECT - ENGINEER		150
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING	.2	50

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II			
AE-TIT. II	100	100	100
CONST.	100	100	100
PF			
CPFF	33	100	100
FP	67	100	100

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.

Site grading, floor drain installation and miscellaneous clean-up has been completed by the CPFF Construction Services Contractor. A few remaining equipment adjustments and corrections are being made.

The project physical completion notice is being prepared and this project will not be reported further.

1236343

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75376

DATE 10-31-62

FUNDING 4-62-d-3

PROJ. NO.	TITLE
CAH-916	Fuels Recycle Pilot Plant

AUTHORIZED FUNDS $ 500,000 #	DESIGN $ 500,000 #	AEC $
	CONST. $	GE $ 500,000 #

COST & COMM. TO 10-14-62 $ 464,800

ESTIMATED TOTAL COST $ 5,460,000***

STARTING DATES	DESIGN 3-15-61	DATE AUTHORIZED 10-19-62**
	CONST. 11-15-62*	DIR. COMP. DATE
EST'D. COMPL. DATES	DESIGN 11-5-62	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	10	2200
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	99
TITLE I	11	100	100
GE-TIT. II	89	100	99
AE-TIT. II			
CONST.	100	0	0
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project is to provide a facility to perform a full scope of engineering tests and pilot plant studies associated with fuel reprocessing concepts.

All drawings have been issued for comment, and have been approved for construction. The specifications are 96% complete.

A new project proposal revision, to permit demonstration of the waste calcination program in FRPP has been submitted to the Commission for approval. It has been approved by HOO-AEC and transmitted to Washington D.C. - AEC for approval.

* Estimated construction starting date for removal of burial ground fill.

** Original authorization for initiation of design was February 9, 1961. October 19, 1962 is the authorization date for the last design supplement.

*** Including transferred capital property valued at $100,000.

\# Directive No. AEC-187, Mod. 5 authorized an additional $35,000 for design on this project. Work Authority No. CAH-916(6) authorized General Electric Company to incur costs in the amount of $500,000.

1236344

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-	75376
DATE	10-31-62
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 10-14-62	$ 14,488(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	12-31-62
EST'D. COMPL. DATES — DESIGN	1-1-62
EST'D. COMPL. DATES — CONST.	12-31-62
ENGINEER	FEO - DL Ballard

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	10	630
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		166
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING	.5	45

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	90	70
PF			
CPFF			
FP	100	90	70

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides facilities for 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 & II costs - $11,000.

Status of the fixed-price contractor's work is as follows:

Electrical work is approximately 85% complete.

Mechanical work is approximately 80% complete. Exhaust ductwork and service piping work is continuing.

The contractor's supplier of laboratory furniture, Metalab, was on strike from September 11 to October 8 and delivery of these items is now promised for December 15, 1962. This, plus the fact the contractor is currently behind schedule will probably necessitate extension of the project completion date. Several design changes have been required to eliminate interferences and other problems relating to ventilation and mechanical equipment.

Directive No. AEC-198, Mod. 3, extended the completion date to 12-31-62.

* Original authorization for design 8-9-61.

1236345

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75376

DATE 10-31-62

PROJ. NO.	TITLE	FUNDING
CGH-951	A-C Column Facility - 321 Building	0290

AUTHORIZED FUNDS			COST & COMM. TO 10-14-62	$ 32,801
$ 55,000	DESIGN $ 5,000	AEC $	ESTIMATED TOTAL COST	$ 55,000
	CONST. $ 50,000	GE $ 55,000		

STARTING DATES		DATE AUTHORIZED	1-2-62	EST'D. COMPL. DATES	
DESIGN	1-30-62			DESIGN	4-1-62
CONST.	3-15-62	DIR. COMP. DATE	10-31-62	CONST.	10-31-62

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		215
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	100*	100*
PF	100	100	100
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide a closely integrated "A" Column in series with the relocated "C" Column to permit the development of a mathematical model for the mass transfer of uranium, as well as the exploration of the possibilities of computer optimization of a combined "A-C" extraction battery.

Relocation of "C" Column is complete. Instrument line gutters are installed. "C" Column is in operation.

Process steam temperature control, flow control, pH control and variable speed drive control instrumentation systems are on order.

Remaining items include instrumentation hook-ups, minor piping runs, and delivery of remaining instrumentation.

* This project will be closed out with exceptions as of 10-31-62 and accruals will be itemized. This will be the last reporting period for this project.

1236346

HW-75376

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

DATE 10-31-62

PROJ. NO.	TITLE	FUNDING
CGH-955	Reactivation of the H-1 Loop - 105-H Building	0490

AUTHORIZED FUNDS			COST & COMM. TO	
$ 10,000	DESIGN $ 10,000	AEC $		$ *
	CONST. $	GE $ 10,000	ESTIMATED TOTAL COST	$

STARTING DATES		DATE AUTHORIZED 3-29-62	EST'D. COMPL. DATES	
	DESIGN 4-15-62			DESIGN
	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 was submitted to Contract and Accounting October 10, 1962.

* The $2614 in charges which were incurred for preliminary scoping have been transferred to operating costs.

PROJ. NO.	TITLE	FUNDING
CGH-957	Small Particle Technology Laboratory - 325 Building	62-k

AUTHORIZED FUNDS			COST & COMM. TO 10-14-62	$ 38,779
$ 40,000	DESIGN $ 2,000	AEC $		
	CONST. $ 38,000	GE $ 40,000	ESTIMATED TOTAL COST	$ 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	5.0	320
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION	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	99*	95
PF	100	99*	95
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 is essentially complete. The principal exception is installation of the two laboratory hoods and exhaust duct connections.

* Project Planning Schedule.

1236347

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 75376

DATE 10-31-62

PROJ. NO.	TITLE	FUNDING
CAH-958	Plutonium Fuels Testing & Evaluation Laboratory - 308 Bldg.	62-k

AUTHORIZED FUNDS			COST & COMM. TO 10-14-62	ESTIMATED TOTAL COST
$ 150,000	DESIGN $ 15,500 / CONST. $ 134,500	AEC $ 148,000 / GE $ 2,000	$ 2,000	$ 150,000

STARTING DATES	DATE AUTHORIZED / DIR. COMP. DATE	EST'D. COMPL. DATES
DESIGN 11-15-62	DATE AUTHORIZED 6-22-62	DESIGN 3-15-63
CONST. 2-1-63	DIR. COMP. DATE 5-15-63	CONST. 7-15-63

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.

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 completed the re-estimate which was submitted to the AEC 10-12-62.

PROJ. NO.	TITLE	FUNDING
CAH-962	Low Level Radiochemistry Building	05-1-63-H-001-23

AUTHORIZED FUNDS			COST & COMM. TO 10-14-62	ESTIMATED TOTAL COST
$ 113,000	DESIGN $ 113,000 / CONST. $	AEC $ 82,000 / GE $ 31,000	$ 16,500(GE)	$ 1,200,000

STARTING DATES	DATE AUTHORIZED / DIR. COMP. DATE	EST'D. COMPL. DATES
DESIGN 7-23-62	DATE AUTHORIZED 6-28-62	DESIGN 5-15-63
CONST. 8-1-63	DIR. COMP. DATE	CONST. 8-1-64

ENGINEER

FEO - DL Ballard

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION	1	35
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	0	0
TITLE I		100	90
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 Commission has requested Hanford Laboratories to prepare and submit a site selection report for this facility.

Preparation of the design criteria is essentially complete, in comment form, and is being prepared for submission to the Commission.

1236340

SEMI – MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-75376

DATE 10-31-62

PROJ. NO. CAH-963	TITLE Geological & Hydrological Wells - FY-1962			FUNDING 62-k
AUTHORIZED FUNDS $ 80,000	DESIGN $ 1,400 / CONST. $ 78,600	AEC $ 68,500 / GE $ 11,500	COST & COMM. TO 10-14-62 / ESTIMATED TOTAL COST	$ 10,317(GE) / $ 80,000
STARTING DATES	DESIGN 5-18-62 / CONST. 7-6-62	DATE AUTHORIZED 5-9-62 / DIR. COMP. DATE 4-1-63	EST'D. COMPL. DATES	DESIGN 6-1-62 / CONST. 4-1-63
ENGINEER	FEO - HE Ralph			

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	8	450
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 1			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	76	76
PF			
CPFF	2	2	2
FP	98	74	74

SCOPE, PURPOSE, STATUS & PROGRESS

This project involves the continued drilling of special hydrological research, test and monitoring wells.

The Contractor is operating 2 rigs on a double shift basis.

Fourteen of the nineteen wells have been completed.

3250 feet of drilling have been completed to date.

Contractor has recovered lost time and is now on schedule.

PROJ. NO. CGH-974	TITLE Analog Simulation Facility			FUNDING 62-a-1
AUTHORIZED FUNDS $	DESIGN $ / CONST. $	AEC $ / GE $	COST & COMM. TO / ESTIMATED TOTAL COST	$ / $ 1,600,000*
STARTING DATES	DESIGN / CONST.	DATE AUTHORIZED / DIR. COMP. DATE	EST'D. COMPL. DATES	DESIGN / CONST.
ENGINEER	FEO			

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT – ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide an appropriately sized and consolidated analog computer simulation facility for the Hanford complex. Initial application will be associated with startup programs for the NPR.

The preliminary project proposal submitted to the Washington AEC has been returned to HOO.

* Approximate estimate.

1236349

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-75376

DATE 10-31-62

PROJ. NO.	TITLE	FUNDING
CAH-977	Facilities for Radioactive Inhalation Studies	62-k

AUTHORIZED FUNDS			COST & COMM. TO	
$ 13,500	DESIGN $ 13,500	AEC $ 10,500	10-14-62	$ 3,000(GE)
	CONST. $	GE $ 3,000	ESTIMATED TOTAL COST	$ 140,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN	1-1-63	9-24-62	DESIGN	2-15-63
CONST.	5-1-63	DIR. COMP. DATE	CONST.	2-15-64

ENGINEER

FEO - JT Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide additional facilities essential to the conduct of Biology research programs involving the effects of inhaled radioactive particles. It will comprise an addition to the 144-F Building consisting of approximately 2000 square feet of indoor dog pens and supporting facilities and approximately 2200 square feet of outside dog runs.

Scoping is approximately 60% complete.

PROJ. NO.	TITLE	FUNDING
CAH-982	Addition to the Radionuclide Facilities - 141-C Bldg.	62-1

AUTHORIZED FUNDS			COST & COMM. TO	
$	DESIGN $	AEC $		$
	CONST. $	GE $	ESTIMATED TOTAL COST	$ Not Established

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN	12-15-62*		DESIGN	6-15-63*
CONST.		DIR. COMP. DATE	CONST.	

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 an addition to the 141-C Building in 100-F Area to supplement the present radionuclide study facilities. The building addition will comprise approximately 2500 square feet for laboratory facilities and controlled feeding pens for swine.

The project proposal requesting $14,000 for Title I and II design was submitted to the AEC September 14, 1962.

The AEC has not acted upon the proposal to date. GE has submitted an answer to an AEC telecon explaining the added complexity of the proposed structure over those previously constructed.

* Based upon AEC approval by October 15, 1962.

1236350

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-75376

DATE 10-31-62

PROJ. NO.	TITLE	FUNDING
CAH-985	Additions to the 222-U Building	63-1

AUTHORIZED FUNDS			COST & COMM. TO	$ -0-
$ -0-	DESIGN $	AEC $		
	CONST. $	GE $	ESTIMATED TOTAL COST	$ 150,000*

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
	DESIGN 12-1-62**			DESIGN 4-1-63**
	CONST. 6-1-63**	DIR. COMP. DATE		CONST. 2-1-64**

ENGINEER

FEO - DS Jackson

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 an addition to the 222-U Building in which to perform 1) geologic and hydrologic studies related to waste disposal practices and 2) studies on release of fission products from reactor fuels heated to high temperatures.

The project proposal, requesting design funds in the amount of $17,000, was submitted to the AEC-HOO October 8, 1962.

* Approximate estimate.

** Based on authorization by 11-15-62.

PROJ. NO.	TITLE	FUNDING
CAH-986	300 Area Retention Waste System Expansion	63-1

AUTHORIZED FUNDS			COST & COMM. TO	$
$	DESIGN $	AEC $		
	CONST. $	GE $	ESTIMATED TOTAL COST	$ 100,000*

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
	DESIGN 1-15-63**			DESIGN 5-31-63**
	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		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide two additional 50,000 gallon retention basins, automation of the basin influent valving and semi-automation of the effluent valving. It will provide required storage basin capacity and obtains maximum use of existing basins.

The project proposal, requesting design funds in the amount of $14,000, was submitted to the AEC-HOO October 8, 1962.

* Preliminary estimate.

** Based on AEC approval by 11-15-62.

1236351

TEST REACTOR AND AUXILIARIES OPERATION

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

The reactor remained shutdown the entire month for primary system decontamination. Work included system preparation for chemical cleaning, the chemical decontamination process, and system inspection and return to normal activities. Chemical flushes with appropriate intermediate demineralized water rinse flushes were conducted in the following sequence:

a. 10% oxalic acid
b. Caustic permanganate
c. 1% oxalic acid
d. Proprietary oxalic acid compound
e. Caustic permanganate
f. 1% oxalic acid
g. Proprietary oxalic acid compound
h. Caustic permanganate
i. Ammonium citrate

Overall decontamination factor was about 4 as measured by field and external system readings. Internal surfaces were cleaned very well and contact maintenance readings were actually less than those experienced prior to the rupture.

Fuel element decontamination by means of ultrasonic vibration continued with a total of 32 elements cleaned at month-end. Helium consumption was 97,400 scf. D_2O inventories showed a gain of 303 pounds as a result of D_2O recovery from ion exchangers and re-analysis of stored, used D_2O.

Equipment Experience

A planned wiring improvement program in the control room has been completed. The program involved installing a better quality wire marker on about 1500 wires, removing unused wires and rerouting others for easier and quicker trouble shooting.

The majority of maintenance effort following chemical decontamination involved activities directed at reactivation of isolated components and systems, regasketing of the primary system, overhaul of injection pumps, replacement of primary pump seals, and miscellaneous valve repairs.

A performance test on the deep well diesel driven pump resulted in insufficient discharge head. Corrective action was underway at month-end.

Preventive maintenance required 393 manhours or 7.5% of total effort.

Improvement Work Status (Significant Items)

Again, because of extensive efforts on decontamination activities, improvement activities were limited.

Work Completed:

Third exhaust air activity channel
Gas balance interlock circuit modification
Temperature alarm trip modification.

Work Partially Completed:

Safety circuit ground and low voltage detector
Fueling vehicle hoist modification
Primary oxygen analyzer installation
Flanges for safety relief valves in helium system - 85%
Position indicating lights for convection cooling assist valve
Decontamination facility

Design Work Completed:

Enlarge chemical feed system
Primary pump recording ammeters
High pressure helium compressor inter-after cooler relief
Outlet nozzle bracing
Interlock between charge-discharge machine, shroud seat and discharge hoist
Control room ventilation scope
Fuel transfer system modifications
Compressed air supply revisions

Design Work Partially Complete

Additional fuel storage and examination
Boiler feed pump seals
Core liquid level instrumentation

Plans for the oil storage building were not approved by HOO-AEC and this item was dropped from the list.

Process Engineering and Reactor Physics

A criterion has been developed for grouping LX Pu-Al fuel elements in batches for the first chemical processing campaign which will enable CPD to use normal alpha-counting methods of process control and thereby reduce costs.

A study of methods to predict reactivity burnup rates in PRTR core loadings continued. The current technique utilizes the Meleager code, the SWAP code and hand calculations. The results of these calculations are compared with observation in the following table:

Comparison of Observed and Calculated Reactivity Burnup Rates

	Date	Jan-Feb 62	June 62	Dec 62	Jan 63
$\frac{\Delta k \ (x10^5)}{\text{MWD-Reactor}}$	Observed	3.05	1.75	--	--
	Calculated	3.0	2.45	1.36	1.06

The rates calculated for December 1962, and January 1963, are those rates predicted for the first and third operating periods following decontamination. The steady reduction of the calculated burnup rates is quite encouraging.

Procedures

Revised Operating Procedures issued		2
Revised Operating Standards issued		9
Temporary Deviations to Operating Standards issued		1
Revised Process Specifications accepted for use		2
Maintenance Manuals issued		1
Drawing As-built status		
	October	Total
Approved for as-built	12	849
Ready for approval		30
In drafting		21
Voided		77
		977
Scheduled for review		432*
		1 409*

*Reflects addition of rupture loop drawings.

1236354

Personnel Training		
Qualification Subjects	326	manhours
Specifications, Standards, Procedures	76	
Fueling Vehicle	0	
Maintenance Procedures	145	
Decontamination	161	
	708	manhours

Status of Qualified Personnel at Month-End	
Qualified Reactor Engineers	7
Provisionally Qualified Reactor Engineers	2
Qualified Technicians	6
Qualified Technologists	17
Provisionally Qualified Technologists	2

Plutonium Recycle Critical Facility

Final preparation of process specifications was underway. Detailed description of startup physics tests were completed. Startup work included installation of cover blocks, partial completion of safety rod thimble modification, and preparation for the cell pressure test. Training consisted of 70 manhours plus the initiation of training of personnel from PIRDO to supplement operating personnel in the future.

Fuel Element Rupture Test Facility

Project Status (Project CAH-862)

Activities were directed at project closeout in conformance with the directive completion date of 10-31-62. Accrual items include the following:

1. B cell shielding
2. Installation of Δ P scram and automatic cooldown instrumentation
3. Emergency depressurizing valve installation
4. Flow limiting orifice installation
5. In-reactor test section installation
6. Fueling vehicle modifications

During PRTR decontamination activities, the rupture loop makeup ion exchange system was used to supply demineralized water. Capacity was found to be 140,000 gallons compared to a design rating of 115,000 gallons.

Operation

Operating procedure issuance was 65%. Ex-reactor startup tests have been prepared. Total training time consisted of 128 manhours.

1236355

GAS COOLED POWER REACTOR PROGRAM

Gas Cooled Loop

Project Status (Project CAH-822)

Construction is 93% complete. The replacement heater was installed, and several punch list items were completed. Bristol-Siddeley has advised that they are changing the journal bearing facings to Teflon impregnated bronze which has been used successfully in other designs. In-loop testing by the vendor is scheduled to begin October 30, 1962.

Operation

Distribution of operating procedures was completed except for one which awaits design information. Total training time was 40 manhours.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 22,885 hours. This includes 14,879 hours performed in the Technical Shops, 3,037 hours assigned to Minor Construction, 4,858 hours assigned to off-site vendors, and 111 hours to other project shops. Total shop backlog is 20,776 hours, of which 70 percent is required in the current month with the remainder distributed over a three-month period. Overtime hours worked during the month was 4.3 percent (846) of the total available hours.

Distribution of time was as follows:

	Manhours	% of Total
Fuels Preparation Department	8,082	35.31%
Irradiation Processing Department	3,151	13.77%
Chemical Processing Department	551	2.41%
Hanford Laboratories Operation	11,096	48.49%
Construction Engineering and Utilities	5	.02%

Requests for emergency service remained at a level which is considered normal for this operation.

LABORATORY MAINTENANCE OPERATION

Total productive time realized was 16,800 hours of a possible 18,720 hours theoretically available. Of the total productive time realized, 89% was expended for HLO components with the remaining 11% of effort directed toward providing service for other HAPO organizations. Overtime worked during the month was 2.8% of total available hours.

1236356

Manpower utilization for October is summarized as follows:

a.	Shop Work (Fabrication, Modification)		3,500
b.	Maintenance		9,600
	1. Preventive Maintenance	3,200	
	2. Emergency or Unscheduled Maintenance	2,200	
	3. Normal Scheduled Maintenance	4,200	
	4. Overtime	530	
c.	R&D Assistance		3,700

WD Richmond

Manager
Test Reactor and Auxiliaries

WD Richmond:bk

1236357

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
T. D. Chikalla, R. E. Skavdahl, and C. E. McNeilly	Production of Stable, High Resistance Electrical Devices, HW-74920, September 11, 1962
L. G. Merker and K. H. Hammill	Low Cost Remote Lighting Fixtures for Hot Application, HW-75117, October 2, 1962
D. P. Brown	HWIR-1564, A Telephone Answering Instrument
C. E. Fitch and N. E. Dixon	HWIR-1566, Description of a Method for Detecting Unbonds
C. E. Fitch, Jr.	HWIR-1567, Description of an Ultrasonic Method for Internal Stress Detection
C. E. Fitch, Jr, and N. E. Dixon	HWIR-1568, Description of a Gas Leak Detection System
C. E. Fitch, Jr. and N. E. Dixon	HWIR-1570, Ultrasonic Defect Detection
R. H. Moore	The Use of Fused Salt Cooling to Improve Metal Purification by Solid State Electromigration (HW-75462)
K. H. Hammill and L. G. Merker	Low Cost Remote Lighting Fixtures for Hot Cell Application
P. R. Rushbrook	The Incorporation of Zeolites Into Vitreous Bodies

H M Parker

Manager, Hanford Laboratories

1236358