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HANFORD LABORATORIES OPERATION
MONTHLY ACTIVITIES REPORT

DECEMBER, 1961

JANUARY 15, 1962

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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DECEMBER, 1961

Compiled by
Operation Managers

January 15, 1962

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RICHLAND, WASHINGTON

[redacted]

PRELIMINARY REPORT

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

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TABLE OF CONTENTS

TABLE I. HLO FORCE REPORT

DATE: December 31, 1961

	At beginning of month		At close of month		Total
	Exempt	Salaried	Exempt	Salaried	
Chemical R & D	128	118	129	117	246
Reactor & Fuels R & D	173	153	172	154	326
Physics & Instrument R & D	92	58	91	59	150
Biology	36	47	36	48	84
Operations Res. & Syn.	17	4	17	4	21
Radiation Protection	38	93	41	91	132
Laboratory Auxiliaries	32	132	31	132	163
Financial	19	16	19	16	35
Technical Administration	97	59	95	59	154
Programming	18	4	18	3	21
General	4	4	3	4	7
Test Reactor & Auxiliaries	35	71	36	72	108
TOTAL	689	759	688	759	1447

BUDGETS AND COSTS

December operating costs totaled $2,312,000; fiscal year-to-date costs are $13,743,000 or 51% of the $27,138,000 control budget. Hanford Laboratories' research and development costs for December, compared with last month and the control budget are as follows:

(Dollars in Thousands)	COST Current Month	Previous Month	FY To Date	Budget	% Spent
HLO Programs					
02 Programs	$ 47	$ 49	$ 266	$ 605	44%
04 Programs	881	853	5 616	10 855	52
05 Programs	82	85	432	993	44
06 Programs	182	198	1 209	2 720	44
	1 192	1 185	7 523	15 173	50
FPD Sponsored	120	119	720	1 400	51
IPD Sponsored	115	96	658	1 325	50
CPD Sponsored	147	140	820	1 570	52
Total	$ 1 574	$ 1 540	$ 9 721	$19 468	50

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

An additional number of favorable results pertinent to the NPR were obtained.

In a test of end closure integrity, two tubular metallic elements containing prototypic NPR brazed end closures have been irradiated for six MTR cycles (cold water) without incident.

The first irradiation of a prototypic assembled NPR tube-tube fuel element is proceeding in the 6x6 ETR loop and has currently achieved an average exposure of 500 MWD/T.

The resistance spike power welding process has been accepted as the method of attaching locking clips for N fuel elements. All required welding processes for attaching supports and locking clips to the N fuel elements have been put in place in the manufacturing plant.

Twenty-three boiling burnout points were obtained with an electrically heated, 46-inch long test section simulating the outer surface and the outside flow annulus of the NPR fuel element. The results obtained on this test section were found to compare favorably with previous results using a test section one-half as long, indicating that test section length is not a contributing factor in boiling burnout for the conditions considered.

Work on various aspects of Zircaloy continued:

a. The ultimate tensile stress for non-irradiated Zircaloy-2 was observed to follow the relation $\sigma = a + b \ln \dot{\epsilon}$, where σ is the ultimate tensile stress, $\dot{\epsilon}$ is the strain rate (varied from 0.0025 to 2.5 min^{-1}), and a and b are constants. The uniform strain decreases with increase in strain rate in the annealed condition but is insensitive to strain rate in the cold worked states.

b. Spacing members have been successfully attached to tubular Zircaloy fuel cladding, using 150 KV electron beam welders. The process appears to be superior, quality-wise, to any previously investigated for ceramic fuel elements.

c. A "brittle fracture" test has been developed for Zircaloy pressure tubing. While the tube is pressurized at the test temperature, a wedge-shaped projectile is fired through the tube wall; the test is intended to delineate conditions under which a brittle crack will propagate.

d. The electrical resistance across the ZrO_2 protective film on Zircaloy drops nearly a factor of 10^4 when exposed in either a dry He gas or a dry vacuum at 450 C. In the low resistance state admission of dry H_2 leads to eventual rapid hydriding, but admission of moisture restores the original electrical resistance and the normal protectiveness of the film.

The Phoenix capsule (GEH-21-1) containing plutonium with 6.25 w/o plutonium-240 has completed its second cycle of irradiation in the MTR, and reactivity measurements have been completed in the ARMF. Capsule number GEH-21-9, which contains plutonium with 27.17 w/o plutonium-240, has successfully completed its second irradiation cycle and reactivity measurements will be made after an appropriate cooling period. Reactivity transient measurements have been completed on the capsule containing plutonium with 16.13 w/o plutonium-240 and it is being irradiated for its second cycle in the MTR.

Post-irradiation studies revealed that in a fuel specimen containing pieces of highly enriched UO_2 heterogeneously distributed in natural UO_2 no "hot spot" effects, e.g., excessive corrosion or core-clad interaction, occurred. Apparently homogenization occurred rather quickly.

Uranium metal inclusions, presumably formed during irradiation of UO_2 at very high temperatures, have been detected in irradiated UO_2. If confirmed, this discovery will have important implications to many other fundamental UO_2 studies.

PuN was formed by arc melting alpha-plutonium under a static atmosphere of nitrogen.

After high temperature irradiation in the GETR to a peak dose ($E > 0.18$ Mev) of approximately 8.7×10^{21} nvt, or 58,000 EGCR equivalent MWD/AT, EGCR graphite samples show no diminution in the irradiation-induced contraction rate. Similar long term irradiations of NPR graphite in the GETR were initiated during the month.

The new shielding computer code has been used to correlate experimental data for the Hanford iron-masonite shield configuration. Relative thermal flux calculations agreed with experimental data within a factor of two through the entire thickness of the shield assembly.

Conceptual design studies included very preliminary work on two types of reactors: special purpose plutonium-fueled reactors and a chemonuclear reactor using fission fragment energies for fuel cell regeneration.

A detailed study of pressure piping code compliance and materials selection in the high pressure PRTR systems was completed. It was determined that the secondary and helium systems were designed in accordance with code. The primary system had several (previously documented) deviations, and one in which the Hanford Works Standard was given precedence over the American Standard Code For Pressure Piping.

2. Chemical Research and Development

Information of several types relating to waste disposal has been obtained:

a. The laboratory study of the addition of sodium silicate to reactor process water continues to show a significant reduction in the adsorption of As-76 and P-32 on aluminum surfaces, even those on which a corrosion film has been allowed to develop. These results have prompted an early in-reactor test.

b. Fission product tritium originating from Purex and Redox condensate wastes is detected in ground water at concentrations of 4×10^{-2} and 4×10^{-5} μc/cc at four and seven miles southeast, respectively, of the 200 East Area.

c. Reproducibility in sampling stack gases for I-131 and subsequent laboratory analysis was determined to be entirely adequate, ranging from one to nine per cent.

d. First evidence of a ground-water mound under a reactor was obtained from a leveling survey of the monitoring wells in the vicinity of 100-B.

Further developments in the molten salt Pu fuel recovery work included the following:

a. Addition of two weight per cent pre-formed SnO_2 to NaCl-2.6 LiCl containing 0.1 weight per cent plutonium precipitated 90 per cent of the plutonium. Recovery was not affected by the presence of 10 weight per cent uranium. Satisfactory decontamination from promethium and uranium was obtained.

b. UO_2 crystals weighing as much as 1.5 grams and with O/U ratios as low as 2.005 have been produced from the LiCl-KCl system.

c. A micro-scale deposition system has been developed to determine the "dryness" of a mixed chloride melt to be used in pilot plant studies. Using this control device, a large batch of $PbCl_2$-2.5 KCl was used to produce a 50-pound deposit of electrolytic UO_2 with an O/U ratio of 2.0036. Furthermore, the low density structure usually found near the electrode was absent. Current densities were lower than those obtained with less dry systems, indicating a possible limitation to production rates.

In the development of methods of preparing fission products for utilization:

a. The development of equipment stations for bulk fission product packaging is nearing completion. The stations provide for a sequence of operation as follows: (1) filling the canister and thermal stabilization of the product, (2) capping and sealing the canister, (3) remote welding, and (4) decontamination and leak detection.

b. The absorption capacity of Duolite C-3 for cesium was improved by a factor of 2.6 by the addition of excess sodium hydroxide to Purex waste-tank supernate. The excess caustic has no effect on the adsorption of technetium on Amberlite 401. Procedures for purifying

technetium eluted from resin columns were developed in the laboratory. At month's end a large scale run was in progress to purify about 15 grams.

c. Clinoptilolite columns were used effectively to remove cesium from synthetic formaldehyde-treated Purex wastes. Also, clinoptilolite is highly selective for trace cesium and strontium in solutions containing magnesium and aluminum.

d. Conditions for the extraction of strontium and rare earths from formaldehyde-treated Purex wastes have been further defined. Extraction of promethium is limiting.

e. Attempted use of the demonstration spray calciner to prepare strontium glasses was successful only in part.

3. Physics and Instrumentation Research and Development

In support of the NPR program:

a. Experiments were completed to select a satisfactory neutron detector for the forthcoming nuclear purity tests of the graphite as laid. Meanwhile a comparison standard value was obtained in measurements carried out in the exponential mockup of the NPR lattice. Fabrication of all the items required for the tests is progressing satisfactorily.

b. It was found that the strength of control rods in the NPR will increase if the annuli around the process tubes become flooded according to results of recent exponential pile tests. This increase in strength will reduce by two-thirds the gain in reactivity (2% in k) associated with such flooding. The investigation will be extended to measurement of the effects of steam vent flooding.

c. Technical evaluations of vendor drawings and plans for the NPR Integrated Data and Temperature Monitor System were compiled for the NPR Project Section and instrument specifications and costs estimates were provided IPD for their use in acquiring fuel rupture detection equipment for the PRTR Fuel Element Test Facility.

d. Major effort continued on nondestructive and destructive test evaluation of NPR primary loop piping. Attention also was given to possible methods for testing the piping for proper heat treatment; residual stress analysis by X-ray diffraction shows promise.

In the area of plutonium fueled reactors:

a. The first modified fuel elements to be irradiated in the PRTR primarily for obtaining physics data on fuel burnup were loaded into the reactor during the month. The first burnup data, however, will come from a standard Mark I Pu-Al element recently removed from the reactor because of a faulty end-spider; after some cooling samples will be obtained from burnup analyses.

b. The effect of fuel temperature changes on the reactivity of Pu-Al fueled reactors may be predicted with more certainty as a result of PCTR experiments completed this month. The reactivity of low exposure plutonium-aluminum fuel was insensitive to changes in fuel temperature. High exposure plutonium-aluminum fuel decreased in reactivity as the fuel temperature was raised.

c. The desirability of a fueling strategy involving a fast-thermal reactor complex was again emphasized by results of a study of the effectiveness of Pu-239 and Pu-241 in fast spectra. In a large fast oxide breeder it was found that Pu-239 was 60% more valuable than U-235 and P-241 was found to be more than twice as valuable as U-235.

d. To satisfy the immediate needs of preliminary hazards evaluation and planning of the initial program in the PRP-CF, a broad scope survey of the characteristics of a wide range of loading configurations has been completed for D_2O moderated, 19-rod clusters with low exposure plutonium and natural UO_2 fuel.

In connection with the older production reactors:

a. An improved version of a mechanism for measuring process tube distortion in the old reactors has been assembled which affords increased ease of reading and a computer program has been prepared to facilitate the reduction of the data.

b. In programs to improve the effectiveness and ease of control of the existing reactors, control rod position readout equipment was installed at 105-KW to obtain data relating rod positions to in-core neutron flux for studies on automatic control. Also, an analog simulation study was begun to determine the best locations in the reactors for scram circuit ion chambers.

Instrumentation research and development included the following:

a. The ultrasonic test for fuel sheath tubing has been improved so that small pits and pieces of impressed material are now readily detected.

b. Specifications were completed for use in soliciting commercial fabrication of modified pencil dosimeters which are to be incorporated in field prototypes of pocket dosimeters which will give an audible alarm when a preset dose is accumulated.

c. Solutions have been developed for most of the equipment difficulties experienced with the Atmospheric Physics radiotelemetry system. The necessary conversions and modifications have been planned and assigned to the maintenance group for completion.

Of interest also are the facts that

a. At the Critical Mass Laboratory the effectiveness of concrete as a neutron reflector proved to be unexpectedly high giving the lowest critical masses yet measured for plutonium solutions. Meanwhile calculational work proceeded with the exploration of the effects of nitric acid and Pu-240 content upon the critical mass.

b. I-131 in human thyroids was detected in measurements at the Whole Body Counter. It is thought to be due to fallout from recent Russian tests since its rise and fall coincides approximately with such fallout. The levels were such that if they had been due to Hanford operations they would have put Hanford in Federal Radiation Council Range II.

c. In the basic data area, gaps in the existing knowledge of neutron total cross sections should be reduced as a result of experiments conducted during the month. Time-of-flight techniques were used to measure the velocities of neutrons in the 3 to 15 Mev energy region, and the cross sections of Li, Na, K, Al, Fe and Cu in this region were determined.

4. Biology

Field and laboratory experience indicates increasingly that the basic cause of the columnaris disease is crowding of fish.

Gut perfusion experiments in rats show absorption of Sr^{85} increasing with lowered pH, but absorption of Ca^{45} unaffected.

12 mg (about 8 μc) of Np-237 injected into a young ewe caused death due to liver damage, presumed due to chemical toxicity.

Neither urethane nor ultrasonic treatment had a significant effect on plutonium deposition in a rat, either with or without DTPA, a well-known chelating agent.

Plutonium nitrate injected into pig skin moves slowly. Eighty per cent of that detected one hour after injection remained at the site eight days later.

Distribution of plutonium in dogs allowed to inhale plutonium nitrate was compared with that measured after intravenous injection. Important differences here and in urinary excretion occurred that continues to point to the need for changing the basis on which bioassay information is interpreted. The following summary Tables illustrate the point:

TABLE 1

Approximate distribution in per cent of body burden
30 days post administration

	Lung	Liver	Spleen	Muscle	Bone
I. V.	0.3	80	6	3	6
Inh.	70	10	0.2	2	20

TABLE 2

Approximate per cent of body burden excreted

		7 days	14 days	30 days
I. V.				
	Urine	0.01	0.009	0.005
	Feces	0.03	0.007	0.004
Inh.				
	Urine	0.04	0.02	0.006
	Feces	0.6	0.4	0.2

[redacted]

The water fowl sampling for the season was completed. About 3000 duck and goose heads have been received during the season for radiochemical analysis from various locations in Washington, Idaho, Oregon and California.

5. Programming

Fuel costs for central station power reactors pass through a minimum and then increase as fuel enrichment, and associated fuel exposure is increased. When plutonium is used as part of the fissile content of the fuel the enrichment associated with minimum fuel cost is altered because of changes in the reactor physics characteristics of the fuel. This enrichment for minimum fuel cost is first approximated by assuming a plutonium value, calculating the corresponding minimum cost enrichment, and from it the value of plutonium. If this computed value does not agree with the assumed value, the calculation is repeated, using the computed value. Substantial progress was made this month toward establishing the conditions under which such iteration is indicated.

At year end, three reports related to fuel cycle analysis and one report on hazards of shipping cerium-144 were in various stages of completion. Document HW-71793, "Estimated Costs for the Production of Pure Mercury-204 Isotope" was issued.

TECHNICAL AND OTHER SERVICES

Further work was done in connection with assessment of the accuracy and precision of measurements performed on pre-irradiated fuel elements, and a report was issued giving a general approach to the problem of routinely checking measurement stations within the Fuels Preparation Department. In connection with this work, a computer program was developed for the rapid determination of the precision and accuracy of multiple measurements.

The excellent agreement between theory and experimental results on the behavior of Lamb waves in slabs of material has prompted a request for mathematical development of a similar theory for cylindrical geometry.

The difference between calculated and chemically measured burnup values were studied for experimental metallic fuels irradiated over the past several years, and a report prepared giving an evaluation of the results obtained and a discussion of the sources of error present in the data.

The revised, multifeatured, computer program used in design studies for a proposed neutron flux monitor based on the Phoenix fuel concept was completed and turned over to the Instrument Research and Development Operation.

Programming is complete for the first stage of the statistical analysis of micrograph pore size distributions.

There were six cases of plutonium deposition confirmed by bioassay analyses during the month. The total number of deposition cases that have occurred at Hanford is 283 of which 208 are currently employed. Based on preliminary data the deposition in each of the six cases confirmed this month was estimated to be less than 10 per cent of the maximum permissible body burden.

The expanded beef thyroid gland sampling program was instituted. Arrangements were made with participating veterinarians in the Yakima Valley and Columbia Basin areas. This program will improve the sensitivity of I-131 monitoring.

The average concentration of air-borne radioactive materials in the Pacific Northwest diminished to about 1/2 that in November and was in the range of 3 to 10 $\mu\mu c/m^3$ at various collection sites.

A comprehensive detailed plan of action to be taken in the event of a radiation accident on the Columbia River was completed and put into effect.

There are 14 currently active projects having combined authorized funds in the amount of $5,703,000. The total estimated cost of these projects is $10,606,000. Total expenditures on them through November 30, 1961 were $2,736,000. In addition project proposals have been submitted to the Commission requesting $279,000 total project funds on three new projects.

Project CAH-902 - Uranium Scrap Burner, was completed 12-29-61. Estimated final cost is $36,000 compared to authorized funds of $36,000.

Project CAH-919 - Air Conditioning - 314 Building, was completed December 6, 1961. Estimated final cost is $34,000 compared to authorized funds of $35,000.

SUPPORTING FUNCTIONS

The PRTR was down throughout the month. During the outage a number of equipment repairs were made (mainly correction of leaks at seals and valves and replacement of a pump because of bearing damage). To improve reactor safety additional low flow indicating instrumentation was installed, and procedures to relieve vapor locking were developed. The repair work required discharging 15 process tubes. The work was accomplished more quickly than had been expected.

The plutonium recycle program critical facility is heading for completion in March, the fuel element rupture test facility is essentially on schedule, and the gas cooled in-pile loop for the PRTR is 92% complete. The latest estimate of the completion of the blowers for this loop is March 1962.

Educational Courses - Seventy-two Laboratories employees completed educational courses during the month. Courses involved were Data Processing, Fortran (Computing), PBM, BOCE and Understanding People. Forty employees are currently enrolled in Principles and Applications of Transistors, Creative Approach Seminar and Effective Presentation.

Advanced Degree - Three PhD applicants visited for an employment interview. Four offers were extended -- no acceptances occurred and two rejections were received. Current open offers total five. During December, six universities were visited for PhD recruiting purposes.

BS/MS - Thirty-six students home for the holidays were interviewed for summer employment consideration. Two Technical Graduate Program offers were made and are still open.

Technical Graduate Program - One Technical Graduate was placed on permanent assignment and two (E&SP members) transferred to other Company locations. Current program members total 73.

Heavy effort has continued to place exempt employees affected by reductions. We have knowledge of 16 offers resulting from the Boeing visit (8 accepted, 4 rejected, 4 still open). Additional offers are expected. Four offers have resulted from efforts with MSVD (3 acceptances and 1 rejection). Some excess personnel have been placed with APED, Kaiser Engineers, Aerojet General and the AEC.

M. C. Leverett

for Manager
Hanford Laboratories

HM Parker:MCL:st

REACTOR AND FUELS RESEARCH AND DEVELOPMENT OPERATION

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 2000 PROGRAM

1. METALLURGY PROGRAM

Corrosion Studies

Resistance Changes in ZrO_2 Films. Electrical resistance is currently being studied as a means for monitoring the protective condition of zirconium oxide films in hydrogen-containing atmospheres.

Resistance measurements at 450 C in a simulated NPR atmosphere (1% CO, 1% H_2, trace H_2O in He) have shown the same changes with H_2O content as were previously reported for measurements made in a vacuum. The resistance measured across the ZrO_2 film between the Zircaloy-2 metal and the external contact decreases from 100,000 ohms to less than 100 ohms when water vapor is removed from the atmosphere. The resistance rises again when moisture is added to the helium. Resistance decreases can also be produced if the flow rate of slightly moist helium past the Zircaloy sample is very low such that insufficient H_2O is available to maintain the ZrO_2 film in an oxygen-saturated condition. The observed resistance drop is predictable on this basis. Presence of hydrogen and carbon monoxide does not affect the film resistance. The final resistance values measured in oxidizing or non-oxidizing atmospheres are reproducible, but the rates of rise and fall have been erratic.

An experiment was set up to correlate electrical resistance and hydrogen permeability of the ZrO_2 film. After the resistance dropped below 100 ohms in dry He (containing H_2 + CO) at 450 C, pure H_2 was added to a partial pressure of 16 mm of Hg. The resistance continued to drop slowly but no H_2 was absorbed by the Zircaloy-2. About three days after the hydrogen was added, at a resistance of 70 ohms, hydrogen absorption was detected. Three days later at the next observation, most of the hydrogen had been absorbed and the resistance was 30 ohms. This experiment verifies a similar observation reported earlier, that hydrogen absorption occurs when the electrical resistance of the film is low.

Oxidation of Zircaloy-2 - Effect of Hydrogen and Oxygen Alloy Additions. Under conditions of localized overheating, high rates of oxygen and hydrogen solution in Zircaloy process tubes or fuel element jackets might occur. To simulate this condition a group of standard Zircaloy-2 specimens were charged with known quantities of oxygen and hydrogen and thoroughly homogenized at 900 C. The nominal oxygen concentrations added were (a) 0 ppm (as-received), (b) 1000 ppm, (c) 2500 ppm, (d) 4200 ppm. Samples at each oxygen concentration level were then hydrided to concentration levels of

(a) 0 ppm, (b) 100 ppm, and (c) 500 ppm. The corrosion behavior of this series of samples is being studied in 360 C de-ionized water, and corrosion data are available after the first 104 days of exposure.

In general, the higher the oxygen content, the higher is the corrosion rate. At 104 days, the Zr-2 with 0 ppm (added) oxygen shows an average weight gain of 38 mg/dm^2, while the Zr-2 with 4200 ppm (added) oxygen shows a weight gain of 67 mg/dm^2. In these two cases the hydrogen made no difference.

At 2500 ppm oxygen added, the specimens without hydrogen show an anomalous rapid weight gain of about 500 mg/dm^2 at about 60 days. The behavior is not a typical transition effect; it appears to be a discontinuity in the weight gain versus time curve. The oxide formed is a lustrous amber color with apparently little tendency to spall. The follow-up oxidation indicates a highly protective character. The specimens containing hydrogen behave in a normal fashion.

Zircaloy Components for NPR Fuel

Local thin spots on the inner Zircaloy cladding have caused the rejection of about two to three percent of the N outer coextruded fuel elements. A similar percentage has a slightly rough fibrous surface on the outer Zircaloy cladding. In cooperation with FPD the quality of the Zircaloy components is being examined as a possible cause of these defects. Large grains, which have been found in a few components, are suspected as a possible cause of the fibrous or wrinkled surfaces. Three coextrusions that had the fibrous surfaces used Zircaloy from the same vendor extrusion. A fourth component from the same material was not coextruded but was destructively tested and found to have grains ranging from 0.035 to 0.150 mm in diameter. With the different flow path of the inner clad during coextrusion, the large grains could also cause the local clad thinning. Here large grains combined with tensile forces created by local differences in extrusion rate could cause local necking and thinning of the clad. The forming processes are being reviewed with vendors to avoid large grains in as-received components, and the possibility of grain growth during the coextrusion preheat cycle.

Radiometallurgy Laboratory Studies

Examination of one NIN element exposed to 2000 MWD/T at simulated NPR conditions revealed that the braze layer at both end closures was in excellent condition, well bonded to the uranium (RM 587). Metallographic examination of a section from the midpoint of a three-foot long KER inner tube (GEH-12-8) shows that the intermetallic layer in the coextruded bond was considerably wider on both the internal and external cladding than is normally observed (RM 589). An NIE element which was irradiated to approximately 1000 MWD/T appeared to be in excellent condition when visually examined. Warp and dimensional changes as a result of the irradiation were insignificant (RM 592). The coextruded Zircaloy cladding of three swelling capsules had ruptured within the NaK-filled cans. An incipient rupture was

found where necking of the cladding had started (RM 565). Two defected KER single tube elements were examined after exposure in an ex-reactor loop. In both elements the reaction of the water on the uranium had progressed rapidly around the outer bonding causing the cladding to swell but not penetrating the core extensively (RM 436).

A total of 14 samples were dissolved for burnup analysis. Density was determined on seven samples and a series of room temperature tensile tests were completed on borided 304 SS.

Results and interpretations of these examinations will be reported in more detail in connection with the development programs served.

Basic Metallurgy Studies

Notch Sensitivity of Zircaloy-2. Recent failures of Zircaloy-2 clad test elements have shown localized failure with little uniform straining in the remainder of the clad. An element sectioned transversely through a potential failure area reveals a well advanced "necking" condition of the clad. This localized straining might result from a sensitivity of Zircaloy-2 to surface irregularities such as a notch or striation introduced during forming. Samples defected with notches of various shapes and depth are being tensile tested at 280 C to determine the effect of wall imperfections on strength and elongation.

Elevated temperature testing continued on tensile specimens with large radius notches. The large radius more closely simulates tube wall irregularities resulting from the coextrusion process. Although notches with large radius are more effective in reducing the uniform elongation than are sharp notches, no notch geometry thus far investigated has introduced a strain instability to the unirradiated material as pronounced as that seen in irradiated cladding on coextruded fuel elements. Tensile test data obtained from a proposed irradiation of rolled-notched material and specimens of cladding removed from coextrusions will be compared with data obtained from duplicate material tested in the unirradiated condition. Existing test data, particularly elongation values, are scattered; apparently from temperature variations during testing. An improved heating method will be used in future tests to stabilize the thermal testing environment.

Metallic Fuel Development

Fuel Irradiations. The production brazed irradiation test at the MTR, GEH-4-63 and 64, has completed its sixth cycle of irradiation and has been discharged. The samples are to be moved to the ETR canal for periscope measurements and pictures and then returned to Hanford for radiometallurgical examination.

Another test, GEH-4-68, 69, and 70, the variable braze thickness test, has been charged in the MTR and is operating satisfactorily. Five cycles of irradiation have been requested and should be completed about March 12, 1962.

The majority of the radiometallurgical examination is completed on GEH-10-39, 40, 41, and 42, the Zr-Be-Fe braze test irradiated in the ETR 6x9 loop. The samples consisted of two KER inner tubes and two KER outer tubes, or a total of eight end closures. Of the four closures which operated at the highest temperature (those at the reactor mid-plane), three showed severe cracking while the fourth showed no cracking but appeared to have had unbonded areas between the braze and the uranium before it went into the reactor. Further investigation of this closure is being made in Radiometallurgy. One of the above cracked three showed evidence of an incipient shear failure. Of the four remaining closures which operated at the lower temperatures, two showed no cracking, one showed a minor crack, and one showed a severe crack completely across the braze interface. All cracks were confined to the area of the braze between the Zircaloy cap and the uranium, and nearly all the cracks appeared to be through the braze rather than along either the Zircaloy to braze diffusion zone or the uranium to braze diffusion zone. This indicates that a good bond was formed in all cases except the one pointed out above and that the braze material was too brittle to support the strains encountered in the ETR irradiation.

Irradiation of the first full-sized assembled NPR tube-in-tube fuel element in the new 6x6 - M3 loop in the ETR has continued. Outlet temperature has been maintained at the requested temperature of 270 C. Maximum cladding surface temperature is approximately 300 C. Average total exposure as of December 21, 1961, is approximately 500 MWD/T. The irradiation is proceeding satisfactorily and nothing unusual has occurred.

Post-irradiation evaluation of five NIE's irradiated to 2100 MWD/T is nearing completion in Radiometallurgy. No inadequacies have been observed in the Be-Zr eutectic brazed closures or in the braze heat affected zones of the fuel, clad, or bonds. A few internal microcracks were observed in the uranium in the non-heat affected zone but these do not appear to have influenced the stability or integrity of the element. The outside diameter of these elements increased an average of 0.012", and there is no indication of buckling of the inner bore of the elements. The average change in single throw warp of the 24" long elements was 0.000" with the changes ranging from +0.007" to -0.006".

Five additional NIE elements have been weighed in the basin after an exposure of 750 MWD/T for the determination of swelling. The average fuel swelling of these five NIE's and the above mentioned NIE exposed to 900 MWD/T is 0.84 percent.

Single Tube Fuel Element. The fabrication of experimental single tube, dual enriched fuel elements was completed. Seven, 22.7" long elements were made from the coextrusion, with one inspection reject for voids in the cap-to-clad braze. Autoradiography indicated some clad thickness variation. The final yield of acceptable elements from the extrusion will depend on a quantitative determination of clad thickness from autoradiography. Two of the elements were fitted with buggy spring supports for centering in the ETR test loop and were given the 72-hour autoclave test. One

element is scheduled for irradiation testing in the ETR-M3 facility starting in February 1962. The fabrication of the basket and accessories for the irradiation was completed, and the assembly was shipped to the ETR site.

Fuel Deformation Studies. Four cladding studies capsules and all necessary equipment for the charge and discharge operation have been delivered to DR Reactor for charging at the next outage. All the samples in these capsules have a nominal cladding thickness of 0.025". The samples in three capsules will operate with a cladding surface temperature of 280 C, while the samples in the fourth capsule will have a cladding temperature of approximately 320 C.

A new test, to replace one which was prematurely discharged, has been designed and capsules parts and Zr-2-clad fuel rod samples are being fabricated. The basic parameters of this test will be the same as before except that the intentional thinning of the cladding will be accomplished by coextruding single striations of controlled depth and geometries on the interface surface of the cladding. The feasibility of producing the internal striations was demonstrated on a rod coextruded by the Fuels Fabrication Operation.

Coextrusion Studies. A scoping program has been initiated to determine the effect of various clad and core properties on the roughness of the interface between the clad and core of coextrusions. One pure, low-oxygen zirconium and one zirconium tin, niobium, iron high oxygen ingot were made in the arc furnace. These were extruded into cladding tubes which will be coextruded with cores of both cast and alpha-extruded uranium in the four possible combinations. The relative effects on interface roughness of the Zircaloy and uranium will be apparent in the degree of roughness of the coextrusion interface.

Component Fabrication. One composite Zircaloy-2 uranium billet was extruded into 0.625" diameter rod to determine the feasibility of coextruding fuel with longitudinal striations of desired geometry and size at the Zircaloy-2 uranium interface. Coextrusions of this type would be used by Fuels Design Operation in the study of failure mechanisms in the Zircaloy-2 cladding of the fuel during irradiation.

The grooves were machined in the Zircaloy-2 sleeve and the billet was constructed in such a way as to insure complete filling of the grooves with uranium during the billet upset prior to extrusion. The resulting striations in the extrusion were very close to the desired size and shape and the grooves did not appear to affect the OD surface of the extrusion.

Component Fabrication: Extrusion Behavior and Properties of Zirconium Alloys. A series of zirconium alloys, prepared by double vacuum arc melting, will be used to determine extrusion constant versus temperature data, resulting structure, mechanical properties, and corrosion behavior. The object of the work is to determine whether a corrosion resistant alloy more nearly matching the extrusion behavior of uranium and with greater ductility than

Zircaloy-2 can be developed by altering the tin and oxygen content of a nominal Zircaloy-2 composition. Two zirconium sponge blends are being used to permit high (1300 ppm)-low (560 ppm) initial oxygen contents and for each blend the tin content is being varied from 0.10 to 1.50 w/o. The iron, nickel, and chromium contents are being held at the nominal Zircaloy-2 composition.

All the alloy ingots were hot upset at 730 C in the four-inch container on the 700-ton press. Following upset, primary extrusion billets were machined, vapor blasted, copper immersion plated, and extruded at 750 C to 2-1/4" diameter bar. These primary extrusions are being machined to produce four secondary extrusion billets of each alloy.

Room temperature hardness and the 750 C primary extrusion constant of each of these alloys have been measured. The pronounced effect of oxygen on room temperature hardness is noted. This increase is 33-45 BHN points in each composition. The room temperature hardening effect of tin is 32 BHN points in the low oxygen series and only 20 BHN points in the high oxygen series.

The stiffening effect of oxygen and tin on the 750 C extrusion constant is somewhat reversed. At this temperature the 1-1/2% Sn has a greater effect than the additional 740 ppm of oxygen.

Electrode Fabrication. The first step in preparing special alloys in arc melting furnace is the fabrication of the consumable electrode since these electrodes must be prepared from available sponge as small metallic pieces, their fabrication is time consuming. Uranium electrodes for the arc melting furnace have now been fabricated on the 600 KVA resistance welder by joining overlapping eight-inch half-round bars. Two small uranium tabs are placed between the bars and act as projections and filler metal for two one-inch diameter spots in each joint. Fabrication by this technique is sixty times faster than the previous practice of electron beam welding each piece in place.

Closure and Joining. A mechanism for the evaporative depcsition of metallic coatings on uranium shot has been designed and will be built and tested within the next six weeks. If the coating experiments are successful, attempts will be made to coextrude the shot in a Zircaloy-2 jacket.

A device has been designed and is being built which will permit the study of distortions of a fuel element during the thermal relief of residual stresses. It should also indicate the amplitudes and directions of distortions due to asymmetry in orientation and grain size. The fuel element under test will be supported as a vertical cantilever and heated by connecting it in series with a high current transformer. Movements of the free and of the cantilever (fuel element) will be observed optically and will be plotted on a circulat coordinate chart. Amplitudes and directions of distortion will be plotted as functions of both time and temperature.

Two alternate approaches to the problem of forming completely sound bonds on the NPR self-brazing type closures were tried. For the first, a two-part constricting collet was placed over the end of an already-pressed fuel element with a poorly bonded closure. The collet was heated to 750 C by induction while pressure was applied axially to the element. Subsequent examination showed that the mechanical behavior of the device was satisfactory; the sidewalls had been swaged inward tightly against the cap. However, the 750 C temperature attained by the collet was insufficient to cause liquation and bonding of the Cu-Zr alloy at the interface between cap and clad. It is planned to pursue this approach further, employing increased temperatures and pressures.

For the second method, the already-pressed fuel element with poorly bonded closures was placed in a gripping chuck on the lower electrode holder in the heavy duty Sciacky spot-welder. An insulated stainless steel retaining ring was slipped around the upper end of the fuel element in the closure zone. The upper electrode was pressed with approximately 7 T load against the cap face, and several pulses of high amperage current were passed through the element. When the uranium became plastic, it was extruded radially between the restraining ring and the collet, forming an undesirable flange. However, the interfacial bond appears to have been greatly improved. This leads to the belief that a higher temperature than that ordinarily attained is required to form a sound bond. The principal problem here appears to be the prevention of distortion of the rest of the element. This problem may be solved by use of a ceramic die.

N Fuel Welding. The resistance spike power process has been accepted as the method of attaching locking clips for NPR fuel elements. The spike power welding machine used in the development work has been transferred from FPD Coextruded Products Engineering to FPD Facilities Engineering and moved to the 333 Building. It has been decided to attach the locking clips individually rather than simultaneously. Machine settings which produce acceptable welds have been furnished. The machine will be put into operation when commercially produced locking clips become available.

Beta Heat Treatment of Zircaloy Clad Fuels. Additional sections of ingot uranium containing iron and silicon addition have been heat treated in several conditions in efforts to better understand the mechanism of grain refinement. The treatments will show the effect of time at beta temperature in dissolving low alpha and high alpha formed precipitate and the effect of beta solution treatments prior to alpha precipitation treatment. These samples are currently being examined.

Charging Force Studies for Present Reactors. Continuation of charging force measurements with aluminum clad elements has led to the following observations:

a. 1.400" diameter I&E fuel elements (no self-supports) were scratched and galled, as were the ribs in the 1.520" minimum diameter ribbed aluminum process tube.

b. 1.990" diameter overbore I&E fuel elements (no self-supports) were scratched and galled, as were the ribs in the 2.157" minimum diameter, ribbed, aluminum, overbore process tube.

c. Self-supported overbore I&E fuel elements with the supports oriented in the vertical plane severely gall a ribless aluminum overbore process tube.

d. Little galling was seen in tests where an A-60 soluble oil was used as lubricant, and the supports were oriented in a 45° plane away from the vertical.

e. Calculated coefficients of friction, for self-supported overbore I&E fuel elements, differ very little from the non-lubricated to lubricated tests. This indicates that galling may be due to higher unit loading in the case of vertical support orientation rather than any lubricating effect by the oil.

f. Nickel plated self-supports on overbore I&E fuel elements show smeared aluminum on the contact surfaces after passing through a six-foot section of ribless aluminum process tubing.

g. Nickel plated self-supports on overbore I&E fuel elements show only polished areas, and no visible damage to a six-foot length of Zircaloy-2 overbore process tubing.

h. The work done to date is preliminary in nature, and above all, indicates the necessity for further examination of overbore fuel element and process tube compatibility.

Cleaning and Pickling. A program was initiated to compare the diffusion of metallic lubricants, copper versus low carbon steel, into Zircaloy cladding during coextrusion. Three Zircaloy billets, two canned in DLP copper jackets and the third canned in an ASTM A-83-60T steel jacket, have been extruded to provide the initial test material. The steel clad billet and one of the copper clad billets were extruded at 750 C. The other copper clad billet was extruded at 600 C to provide a comparison with the standard process material. In each case the extrusion ratio was 13.6 to 1. To obtain maximum diffusion resulting from a metallurgical bond between the metallic lubricant and the Zircaloy billet, the components were bright etched chemically and the assemblies closed under vacuum by using electron beam welding.

This was the first steel lubricated extrusion made by FFDO. From cursory visual examination of the as-extruded rod, the jacket appears to be smooth and sound.

2. REACTOR PROGRAM

Corrosion and Coolant Systems Development

Rupture Testing of NPR Fuel Elements in IRP. Data have been obtained on five additional rupture tests of NPR fuel elements. In all cases the release of rupture products from the slow cool-down is considerably higher than from the fast cool-down (5 minutes at 300 C, then to 160 C in 4 minutes). The fuel elements which had been irradiated for 2000 MWD/T appeared to rupture worse than those irradiated to 3500 MWD/T.

Corrosion Evaluation of AMS 5616 Steel. It is proposed to use AMS 5616 as material for the shafts of the main diversion valves of the NPR. Tests are being run to determine the effects of repeated decontaminations on this material. After eight cyclic decontaminations using alkaline permanganate and a bisulfate reagent, the samples were examined. Cracking was observed on the material heat-treated to 45 R_c. On material treated to 32 R_c, no cracking was observed; however, extensive intergranular corrosion (1 to 5 mils deep) was evident.

Decontamination of Loop After Rupture. An irradiated fuel element rupture test was conducted in IRP contaminating the loop and charged samples. The APACE process (modified) and a pretreatment of peracetic acid buffered oxalate-peroxide procedure was used. Liquid samples were taken from the uranium oxide dissolution step. The loop filter activity was reduced from 4,000 mr/hr to 100 mr/hr by the three-step process used.

Evaluation of New Decontaminants. Samples of two new improved reagents were received from Turco. One solution, an improved 4306 C proved safe and efficient for decontamination of present reactors at temperatures of 30 and 60 C. The efficiency was low when used as a single-step decontaminant for once-through decontamination of NPR. However, as a two-step procedure (modified APACE) for NPR, this material looks promising, giving high decontamination factors with low corrosion.

A modified bisulfate reagent from Turco was moderately effective at 60 C for decontamination of present reactors, but at lower temperatures it was ineffective. As a single-step procedure for NPR, it was ineffective. As a two-step procedure, it looked very promising (comparable to the modified 4306 C). Discussions were held with representatives from Wyandotte Chemical. Three sample materials are to be submitted for evaluation within the next two months. These materials are each for separate recommended applications: (1) a present reactor decontaminant, (2) a single-step carbon steel decontaminant, and (3) a single-step stainless steel decontaminant. The single-step decontaminant contains no phosphate, chlorides, or fluorides. By simple neutralization with alkali, practically all the activity will be precipitated with a very small amount of sludge, thus simplifying the waste disposal problem.

Analysis of Gas in KER. Coolant samples are now being monitored on a routine basis to determine the normal dissolved hydrogen (and total dissolved gas) concentration in the KER-1 loop coolant. The coolant samples are cooled and collected in an evacuated flask. The gas is first removed from the coolant sample by a vacuum pump and is then transferred to a gas burette where the volume is measured at atmospheric pressure. The gas is then transferred to the reaction vessel where the hydrogen concentration is measured. Additional data obtained this month confirm the previously reported information that the normal total gas concentration is four to five cubic centimeters (STP) per liter of coolant. The indicated hydrogen concentration was higher than reported previously, 1 to 1.5 cubic centimeters per liter compared to 0.5 reported earlier.

Technicon Autoanalyzer. Tests were resumed to evaluate the chloride analytical procedure. During these tests new reagents were prepared in an effort to determine the cause of the poor reproducibility of results reported previously. It was observed that when fresh ferric nitrate reagent (9 w/o ferric nitrate in 2 N nitric acid) was used in place of the reagent furnished by Technicon, the results obtained were quite stable and reproducible. The reagent can be used for two to three days before it becomes ineffective. Although evaluation of this procedure is by no means complete, data obtained during the recent tests indicate that the sensitivity for chloride is about 10 parts per billion. Further evaluation of the procedure will continue next month.

Caustic Cracking. Testing continued during the month studying the effect of 1 g/l LiOH on stressed samples of 304 SS and Zr-2 at 290 C. Examination of the samples after two months of testing showed no cracking or hydriding. The test is being continued with samples of A212 carbon steel included. Higher concentrations of LiOH will also be studied to determine at what concentration problems occur.

One test in TF-7 evaluating the cracking tendencies of 316 SS in the presence of a mixture of steam and concentrating LiOH solution was terminated. The outside surface of this tubing was contacted by a small amount of low temperature pH 10.0 water which flashed and deposited LiOH on the surface. The tubing was stressed to 5000 to 6000 psi by the internal loop pressure. No cracking was found after four months of operation at 260 C.

Identification of oxide Films on Aluminum Exposed to Process Water. An iodine-methanol solution has been used in the past to remove the film from aluminum pieces for work connected with various activity studies. However, low temperature films recovered by the iodine-methanol process appeared to be iron-free while the in-place films were a dark brown color and suspected of containing iron. A comparative test was run to determine whether the iodine-methanol solution was dissolving the iron oxides. The film removed with iodine-methanol and the film material removed with ALCOA Bright Etch were analyzed for iron. The iron contents were 0.56% and 8.0%, respectively. The 0.56% corresponds very closely with the iron content of the 1245 aluminum. It is concluded that the surface deposits of iron

oxide on low temperature aluminum films are dissolved and removed from the film in an iodine-methanol solution. (Iodine-methanol solutions used to remove film from aluminum are not water-free.) The method using the ALCOA Bright Etch appears to give valid results and will be used in similar studies of films formed in the reactor flux.

Nonmetallic Materials Development

Filler Bar Contraction. Post-irradiation length measurements were made on three graphite bars that had been charged in the G test hole at C Reactor before startup. They were discharged in April 1960, with an exposure of 18,640 MWD/AT. The temperature during the last several years of exposure was ~550 C.

The bars are 4 x 4 x 24 inches long, approximately half the length of a normal filler block. The three graphite grades represented are CSF, CSO, and CSGBF. Length determinations were made at the ends and center of each bar transverse to the extrusion direction and between the centers of the ends of each bar parallel to the extrusion direction. Surface condition of the CSF and CSGBF bars was good with machining marks still visible. The bar of CSO graphite was extensively pitted with the exception of regions on the bottom of the bar which were in contact with filler bars in the layer below the test channel. The contractions calculated from all measurements were averaged and the results are given in the following table.

Graphite Type	Length change, %	
	Transverse	Parallel
CSO	-0.450	-0.690
CSF	-0.529	-0.758
CSGBF	-0.566	-0.785

The transverse contraction was not constant along the two-foot bars. The ratios of the contraction measured at each end for individual bars were in the range of 1.1 to 1.2. Since the original four-foot bars were cut in half, the different contractions observed may be a result of structural differences at the ends of bars produced during manufacture.

The transverse contraction observed for the bar of CSGBF has been compared with data from C Reactor and with small samples of CSGBF irradiated in the Y test hole at C Reactor. The results are as follows:

Transverse filler bar contraction	Contraction of C Reactor based on traverse of 4675-C taken on 3-7-60	Contraction of small CSGBF samples extrapolated to filler bar exposure
0.566%	0.484%	0.21%

The ratio between filler bar and reactor data is ~1.2, whereas the ratio between filler bar and small sample data is 2.7. The reactor contraction value is lower than it would be if all graphite below the traverse tube were operating at the temperature and flux in the vicinity of the test filler bars.

The comparatively low value for small samples could be caused by the relief of stresses during machining or by the absence of stresses which may accumulate in large bars during irradiation. Evaluation of dimensional data will continue, and further information on weight and purity will be obtained.

Graphite Burnout Monitoring. A new type sample carrier which was recently put into use has provided some insight into the relationship between the burnout rates of small monitoring samples and full-size graphite bars. Two of the four samples in each carrier were shielded by a 3/8-inch wall of graphite through which the reactor gas had to diffuse; the other two samples were exposed directly to the ambient reactor gas. Five of these new carriers were placed in channel 3066 at KE Reactor from June 22 to November 21, 1961.

When the burnout rates from these two types of samples were plotted against their position within the channel, the curve for the bare samples showed a rather sharp peak with a maximum of ~1% weight loss per thousand operating days (KOD) 140 inches from the front of the stack. The curve for the protected samples revealed a very gradual peak with a maximum of 0.2 to 0.3% KOD at 175 to 195 inches from the front of the graphite stack. These results show that there is a time lag for the oxidant to diffuse through the 3/8-inch graphite wall. The burnout curves approached each other for samples near the end of the channel. The ratio of weight-loss for the two types of samples ranged from 3 to 6-fold in the front half of the stack.

Nitrogen was used in place of carbon dioxide in the KE Reactor gas atmosphere, and there were no samples during this test which exhibited a burnout greater than 2%/KOD.

Long Term Irradiation of NPR Graphite. The first capsule (H-4-1) of NPR core graphite (TSX) was installed in the GETR on November 27. The capsule was installed in the D-7 position and is operating satisfactorily with samples in the temperature range 450 to 800 C. The capsules employ gamma heating and the sample temperatures are monitored with thermocouples. This capsule, containing 18 NPR graphite samples and 6 CSF reference standards, will be irradiated for three GETR cycles.

Compatibility of Graphite with Helium Containing Water Vapor. One of the important considerations for the NPR atmosphere is the ability of water to diffuse through the graphite and reach the zirconium tubes. To test the possible effect that soot formation might have on the permeability, a small sample of graphite has been exposed to a microwave glow discharge of carbon monoxide. The relative permeability of the sample to flowing helium has been checked by a differential manometer. The sample was exposed to

~ 20 hours of the activated gas and no change in relative permeability was observed. The sample is now being checked to determine the extent of sooting which occurred.

Thermal Hydraulic Studies

Heat Transfer Characteristics of NPR Fuel Elements. The experimental studies to determine the boiling burnout conditions for the NPR tube-in-tube fuel elements were continued. Twenty-three additional boiling burnout conditions were obtained with a test section simulating the flow annulus between the outer tube of the fuel element and the process tube. This test section consisted of a 46-inch long electrically heated tube 2.387-inch OD by 2.095 inch ID placed concentrically in a 2.70-inch ID process tube. The inside tube was heated by electrical resistance heating, and all of the heat generated was transferred to water flowing through the annulus. Burnout conditions were detected by noting the temperature excursions indicated by thermocouples attached to the inside wall of the heated tube.

Coolant mass velocities were varied from 0.5 to 5×10^6 lb/hr/ft^2, and burnout was at local coolant conditions ranging from about 60 F subcooled to 40 percent steam quality. Corresponding burnout heat fluxes ranged from 0.4×10^6 to 1.4×10^6 B/hr/ft^2.

The results obtained here were compared with those of an earlier investigation which had a test section of identical cross-section geometry but only 23 inches long. This comparison tends to substantiate the assumption that test section length is not an important variable in boiling burnout and that the results obtained in experimental procedures involving short test sections are applicable to long columns of NPR fuel elements.

Heat Transfer Experiments for C Reactor Overbore Fuel Elements. Laboratory experiments were performed to determine the heat transfer conditions existing for a column of C overbore fuel elements during a power surge.

The test section consisted of a 1.991-inch OD by 0.339-inch ID electrically heated rod in a 2.147-inch ID Zircaloy tube. By varying the electrical resistance of the rod in the axial direction, a cosine variation of heat generation was simulated. This heater rod was equivalent to a 32-piece fuel charge and was centered in the process tube by 192 ceramic ribs of the same general shape as those being used on the reactor fuel elements. The rear nozzle assembly was that shown in Drawing SK-1-4774.

The method of experimentation consisted of first establishing steady state conditions of flow and bulk outlet conditions at the initial tube power and typical rear header pressure. A large power increase of desired magnitude was applied over a period of several seconds and was maintained at a steady value for a period of time. Initially, the temperature of the simulate fuel increased slowly since much of the heat generated in the rod was being lost to the coolant by nucleate boiling heat transfer. When the surface temperature of the rod became sufficient to establish film boiling

conditions, the temperature of the test section was observed to increase at a rate of almost 200 F per second. At this time a power decrease was effected in a manner to simulate a 1400 in-hour scram.

It was found that at an initial steady state tube power level of 1000 KW, a surge to 1930 KW could be tolerated for approximately 15 seconds before it was necessary to simulate the scram. At an initial power of 1800 KW, 3390 KW could be tolerated for about eight seconds before the scram was necessary. In both cases the recovery was immediate upon the decrease in power.

Heat transfer Conditions for Eccentric Annuli. Experiments were continued to determine the effect upon heat transfer of the non-coaxial placement of I&E fuel elements in the process tube.

The test section consisted of a 25-inch long electrically heated rod placed with 75% eccentricity within a simulated BDF process tube, 1.60" ID. (Percent eccentricity is defined as the percentage of the normal annulus thickness that the fuel element is displaced from the concentric position in the process tube.) All of the heat generated in the rod was transferred to water flowing through the annulus.

Twelve boiling burnout points were obtained at a constant flow rate of 23 gpm and a pressure of 121 psig and two were obtained at 34.5 gpm and 254 psig. These conditions correspond to the predicted values of annulus flow and local pressure at the point of burnout in the fringe zone and central zone, respectively.

These burnout conditions were determined experimentally by gradually increasing the heat generation rate within the test section in a stepwise fashion and noting temperature measurements of the heated surface as indicated by thermocouples within the test section. A large temperature excursion at some point on the surface of the rod defined the existence of burnout conditions, and this terminated the run. The conditions at boiling burnout were as follows:

Pressure psig	$T_{sat} - T_{water}$ °F	$G \times 10^{-6}$ lb/hr-ft^2	$(Q/A) \times 10^{-6}$ B/hr-ft^2
121	166	4.4	0.826
121	128	4.4	0.945
121	169	4.4	1.04
121	128	4.4	0.750
121	110	4.4	0.733
121	91	4.4	0.715
121	80	4.4	0.660
121	57	4.4	0.669
121	40	4.4	0.665
121	155	4.4	0.738
121	193	4.4	0.805
121	233	4.4	0.912
254	107	6.58	0.862
254	137	6.58	0.925

The data were very similar to those obtained previously for the K Reactors and indicate that a significant reduction in boiling burnout heat flux occurs as a result of eccentric placement of the fuel elements within the process tube.

Shielding Studies

The new shielding code has been reprogrammed to allow scattering from several preceding groups in the multigroup equations. Also, an improvement has been made in the length of time it takes to calculate the exponential functions E_1 (b), E_2 (b), and E_3 (b). This time has been reduced by a factor of two. The exponential function is defined as

$$E_n(b) = b^{n-1} \int_b^{\infty} \frac{e^{-t}}{t^n} \, dt.$$

With these improvements, flux calculations were made for the iron-masonite configuration in the bulk shielding facility at the DR Reactor. This configuration consists of a graphite reflector, iron thermal shield, and 13 alternate layers of iron and masonite. This experiment is a good test for the computing program because of the complexity of the multi-layer assembly and because the composition of each layer is known quite well. Relative thermal flux calculations agreed with foil data within a factor of two throughout the assembly (flux was normalized at the interface between the reactor core and the reflector). The error was much less than a factor of two in the graphite reflector.

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

Thermal Hydraulic Studies

The PRTR primary pumps were equipped with flywheels in order to maintain some coolant flow to carry away reactor heat following an electrical outage until natural convection is able to provide adequate cooling. It has been found that with certain types of impending pump seal failure, excessive friction is developed which causes a more rapid decay of pump speed and primary coolant flow. In order to avoid possible difficulties resulting from such an occurrence, a criterion for satisfactory seal operation was incorporated into the PRTR Process Specifications. This was based upon the time necessary for the pump to coast to a complete stop from full speed.

It was subsequently found that it was more desirable to base such a criterion on pump flow decay rather than on pump speed decay and, accordingly, calculations were undertaken to provide information necessary to modify the process specification.

These calculations indicated that a pump would provide adequate coolant to the reactor following a power outage if the pump flow will not decay to a value less than 20 percent of its initial flow within 30 seconds after an interruption of the electrical supply.

Component Testing and Equipment Development

Second Generation PRTR Shim Rods. One of the alternate designs being considered for PRTR shim control rods utilizes a liquid poison column. Physics considerations indicate that cadmium sulfate poison in an aluminum control rod assembly provides desirable burnup and operating characteristics. However, the possibility of potentially high corrosion rates for aluminum in this service is suggested in the literature. Additionally, cadmium sulfate has an inverse solubility characteristic which peaks at 43.6 C. The solubility characteristics could result in cadmium sulfate poison plating on the housing tube surfaces. A study is being made to determine whether special fabrication techniques (e.g., anodizing the aluminum tubes) would circumvent the potentially undesirable aspects of this condition.

PRTR Shim Rod Testing Facility. A facility is being designed for testing and evaluating shim rods in a radiation-less PRTR operating environment. Where possible, means for visually inspecting the various components in operation will be provided.

Plutonium Recycle Critical Facility Components. The new swivel joints in the PRCF irradiated fuel element transfer mechanism have been installed, and the mechanism is being prepared for transfer to the reactor. Some difficulty was experienced with leakage at threaded connections; these are being repaired.

The fabrication of the three control rods for the PRCF is 90% complete. Completion of the control rods awaits receipt of the remaining cadmium plated control element tubes. The vendor has shipped two of the inner tubes and the surface condition is satisfactory. The promised shipping date for the remaining tubes is December 22. Assuming receipt of cadmium plated tubes as promised, the control rods will be completed in January.

PRTR Gas Loop Components. Assembly of the gas loop pressure tube liner for use in the "B" Cell test section was started.

Testing of the CO_2 diaphragm compressor for the gas loop was completed. Consistent results on diaphragm life and mode of failure were obtained. The compressor was judged satisfactory for gas loop installation.

PRTR Rupture Loop Components. Preliminary testing has shown that the proposed method for discharging fuel elements from the rupture loop is not acceptable. In this method the pressure tube, with ruptured fuel element, is maintained full of water during discharge but without provision for forced circulation. Water is added through the top of the nozzle during discharge to replace the water lost because of boiling. The preliminary tests, however, have shown that the pressure tube acts like the tube in a percolator with the result that almost all of the water above the fuel element is ejected in one violent spurt. The time for this ejection to occur after flow has been stopped varies with the power level of the fuel element but may be as short as ten minutes.

Alternate solutions to fuel discharge by this technique are being sought.

Hazards Analyses

PRTR Process Specifications. Four PRTR process specifications were revised with two being approved during the reporting period. Special conditions to maintain reactor safety were written and approved for the conduct of PRTR Test No. 12, "Measurement of Effective Delayed Neutron Fraction," and for the repair of the primary coolant pump check valves.

Plutonium Recycle Critical Facility. The system of electrical interlocks and the mechanical "dump valve-cell crane" interlock was considered by the General Electric Technological Hazards Council to be an adequate safeguard against a fuel drop-in accident.

A report is being prepared, in response to the Council's request, describing the procedures that will be followed to limit excess reactivity and at the same time retain simplicity of loading procedures for irradiated fuel elements. The following is a summary of the preliminary conclusions.

Insofar as possible, the excess reactivity available for addition from the control console will be limited to 75 cents. It will be necessary to provide a mechanical stop to limit weir travel (and hence achievable moderator volume from the control console) to accomplish this. Most of the experiments planned for the Critical Facility can be performed within the 75 cents excess reactivity criterion by using additional components such as reactivity standards in the form of fuel pieces or by poisoning excess reactivity. Experiments requiring greater than 75 cents excess reactivity will be performed only if specific needs exist and will be thoroughly reviewed to assure that the experiment cannot be performed within the 75 cents criterion.

Miscellaneous Studies

PRTR Pressure Piping Code Compliance. A detailed study of pressure piping code compliance and materials selection in the high pressure PRTR systems was completed. It was determined that the secondary and helium systems were designed in accordance with code. The primary system has several (previously documented) deviations, and one in which the Hanford Works Standards was given precedence over the American Standard Code for Pressure Piping. These included:

1. Pressure Tubes: Zircaloy-2 was not (and still is not) a code approved material, and no code case was obtained for its use in PRTR. The design considerations are detailed in HW-50337. Allowable stresses were not based entirely on code practices, and design stresses were not determined by code formula.

2. Instrument Line Size: The code requirements for 3/4-inch pipe size instrument lines and root valves were not adhered to on-flow monitors and rupture monitor piping because of the relatively small main line (1-1/2" - 1-3/4"), space limitations on the top and bottom faces of the reactor, and safety analyses showed the smaller lines to be less hazardous to reactor operation upon failure than the larger sizes.

3. C Factor Allowance: No allowance of "C factor" was made in jumper piping wall thickness. This was interpreted as a corrosion allowance according to the working of the 1955 code and was considered unnecessary in an all stainless steel Zircaloy system containing high purity water. The C factor was subsequently redefined by code case (1959) as an "allowance for minimum structural stability". In the application on the PRTR, the additional jumper wall thickness was undesirable because of increased bending forces which would be placed on the pressure tubes.

4. The Hanford Works Standard 8064-S specifies that welded pipe without filler material shall be fabricated to ASTM A-312 specifications. This specification is approved by the ASME Boiler and Pressure Vessel Code but not by the ASA Piping Code. It is basically similar, however, to ASTM A376 which is an ASA Code accepted material.

Plutonium Fuels Development

Phoenix Experiment. The Phoenix capsule (GEH-21-1) containing plutonium with 6.25 w/o Pu-240 has completed its second cycle of irradiation and reactivity measurements have been completed in the ARMF. Capsule number GEH-21-9 which contains plutonium with 27.17 w/o Pu-240 has successfully completed its second irradiation cycle, and reactivity measurements will be made after an appropriate cooling period. Reactivity transient measurements have been completed on the capsule containing plutonium with 16.13 w/o Pu-240, and it is being irradiated for its second cycle.

The underwater guillotine has been successfully used for cutting the Al-Co flux monitors in the MTR canal. All of the irradiated monitors have been cut and are being counted.

A calibration capsule for determining what effect the presence of a Phoenix type sample has on the ARMF spectrum has been fabricated. The capsule consists of an Al-Pu alloy sample enclosed in Zircaloy cladding. The capsule parts at the center and is normally held together by screwing the two halves into Zircaloy sleeves. Aluminum and cadmium containers for copper, gold, and lutecium activation foils are positioned between the two halves of the sample. The zirconium cladding which covers the ends of the Al-Pu sample in the center of the capsule was made only 0.005-inch thick in an effort to place the samples as close to the activation foils as possible.

Irradiation Testing. The UO_2-PuO_2 PRTR prototype seven-rod cluster fuel element (GEH-11-7) is being irradiated in the ETR 3x3 high temperature loop. This element was fabricated by cold swaging a mixture of arc-fused depleted UO_2 and PuO_2 which contains 16 w/o Pu-240 to about 89 percent of theoretical density in 0.032-inch thick Zircaloy cladding. Swageable end caps were used which permits complete sealing of the tubes before swaging. The element has accumulated 16 full power days at a power generation of 154 KW which is only four percent lower than the calculated power generation. The maximum heat flux is 345,000 Btu/hr-ft^2 at a maximum calculated core temperature of 1850 C.

GEH-11-6, which is an injection-cast Al-Pu seven-rod cluster, will be inserted into the ETR 3x3 loop for irradiation during January. This element was fabricated by injection casting the Al-Pu alloy fuel directly into the Zircaloy cladding. A metallurgical bond is formed between the core and its cladding.

It is presently planned to irradiate the UO_2-PuO_2 cosine enriched element (GEH-11-8) after the irradiation of the injection cast element. The fabrication of this element is in progress. It is being fabricated by cold swaging with swageable end caps and the plutonium enrichment varies from 3.01 w/o PuO_2 at the top, 0.821 w/o PuO_2 at the center, and 2.40 w/o PuO_2 at the bottom. Plutonium with 16 w/o Pu-240 is being used. The enrichment is varied so that a uniform power of about 84 kw/ft will be generated along the full length of the element.

Radiometallurgical examination is continuing on the seven-rod UO_2-PuO_2 element which had sintered and ground pellets as the core material. Full length autoradiographs were made by exposing glass to the irradiated rods. The darkened glass was scanned with a recording densitometer and the relative exposure along the length of the fuel rods was determined. This technique worked satisfactorily for determining burnup along the length of any one rod; however, it is difficult to determine the relative burnup between the different rods in the cluster because of exposure problems and the annealing characteristics of the glass. Three of the fuel rods have been transversely sectioned at the mid-plane and the cores are being examined. General cracking of the UO_2-PuO_2 is observed. The grain size of the three cores that have been examined is small indicating that very little grain growth occurred during irradiation. From the actual power generation of the element, however, maximum central core temperatures in the range of 1600 to 1800 C are predicted. Micrographs of the transverse sections at 250X show some general porosity throughout the core with localized regions of higher porosity concentration. These sections will be compared with unirradiated samples.

The irradiated Zircaloy-clad capsule (GEH-14-27) containing an Al - 2.1 w/o Pu - 2.0 w/o Ni alloy core and fabricated by injection casting was recharged into the MTR. The specimen has completed two of the four requested additional reactor cycles. Goal exposure for the capsule is a minimum of 50 percent burnup of the plutonium atoms.

The eleven discharged high and low density UO_2-PuO_2 capsules which have exposures on the order of 10,000 MWD/ton of UO_2-PuO_2 arrived at HAPO and are being radiometallurgically examined.

The four specimens (GEH-21-13, 14, 15 and 16) each containing UO_2 - 0.154 a/o PuO_2 and UO_2 (1.00 a/o U-235) pellets, for the tests in the VH-4 Hydraulic Rabbit Facility at the MTR have not been charged into the reactor. The simulated specimen (GEH-21-17) was tested in the facility to determine the tube flow under various conditions. With an initial flow of 19.8 gpm and the specimen in the active zone of the reactor, it was found that by rapidly reducing the flow to 6.2 gpm the specimen would pass by the magnetic indicator, about six feet from the starting position, in 20 seconds. Of the six-foot distance, only the first 18 inches are in the high neutron flux field. By reducing to zero flow, the specimen required only 10 seconds to trip the indicator. From these data, it appears that a specimen could be charged and discharged without encountering a near-stagnant water period and the associated heat transfer problems with an element of high specific power. A modified test proposal, incorporating the corrected specific power values and the above flow data, is being prepared.

Design work on the ICARUS Experiment for high specific power rabbit experiments was completed, and two assemblies are under construction. The flow tube mockup for ex-reactor testing is being fabricated. Physics calculations were made for the case of UO_2 - 0.5 w/o PuO_2 and enriched UO_2

and from the physics data it appears that the specific power could be varied from 22 to 32 kw/ft for the high density (95 percent of theoretical value) materials by the use of neutron absorbers around the sample.

The design effort on the Helios Experiment for in-reactor temperature measurement is continuing. Experimental work on the fabrication of the UO_2 - 5 w/o PuO_2 fuel cores was initiated, and the results were very favorable.

The exploratory work on the ARGUS Experiment for in-reactor fuel element property measurements is proceeding. Replies were received from several vendors with respect to the proposed off-site development work.

Oxide Fuel Development. One of the problems associated with vibrationally compacted UO_2-PuO_2 fuels is maintaining the desired longitudinal PuO_2 distribution during compaction. Evaluation of the uniformity of PuO_2 in a UO_2 fuel rod can only be made by experimenting with PuO_2; however, some indication of the distribution can be determined from the ratio of fine to coarse particles of UO_2 after vibratory compaction since the PuO_2 will be blended with UO_2 fines before addition into the fuel tube.

Two tubes were compacted while being incrementally loaded with UO_2 in three-gram increments of fines (-200 mesh) and nine-gram increments of coarse plus medium (-6 +10 and -35 +60 mesh). After loading, the tubes were cut in six-inch lengths and the contained UO_2 was resieved. Longitudinal distribution of the uniformly added fines was found to vary from 18.5 to 26 percent in the first tube. The average was 22.9 percent which indicated some holdup of fines in the sieving operation since the amount of fines added was 25 percent.

In the second experiment the fines varied from 20.0 to 24.8 percent along the length of the rod; again, the average, 22.7 percent, was less than the amount added, indicating the doubtful sensitivity of the measurements. The longitudinal density variations for the two runs were:

Run No.	Maximum	Minimum	Average
1	87	81	85
2	88.5	83.2	85.4

A proportional feeding and blending device was developed for metering uniformly enriched UO_2-PuO_2 into vibratory compacted fuel elements. The equipment is being installed in the vibratory compaction glove box and will be tested with PuO_2 as soon as the hood is ready to operate. The present device will handle enough oxide for one 88-inch long fuel rod; however, the equipment could be modified for multiple tube loading.

UO_2 Fuels Development

High Voltage Electron Beam Welding. Spacer ribs were welded to Zircaloy tubes, using a 150 Kv electron beam welder. The electron beam weld appears to be superior in strength, corrosion resistance, and uniformity to that obtained by the high frequency resistance welding method, which also is being evaluated. Rectangular rib stock can be welded to the tube, whereas a pre-shaped rib, having two tapered sides, was required for the resistance weld. A full rib width weld is provided between the rib and the tube. Ribs of any height can be attached to tubing by this method.

Electrodeposited UO_2. Ninety-four percent removal of chloride impurities from unwashed (water), electrodeposited (from $KCl-PbCl_2$) UO_2 was demonstrated by heating ten hours in hydrogen at 1000 C. These and other data reported earlier are summarized below.

EFFECT OF 1000 C, H_2 TREATMENT ON IMPURITY CONTENT OF ELECTRODEPOSITED UO_2

	K,ppm	Pb,ppm	Cl,ppm	O/U
As electrodeposited (unwashed)	1400	12,100	1100	2.013
After H_2 Treatment	360	250	65	2.001

Moisture Content of Swaged UO_2 Fuel Rods. Moisture content of UO_2 in swaged PRTR fuel rods is approximately 40 ppm by weight. An electrolytic moisture monitor, which yields results reproducible to within 10 ppm, was used for the most recent analyses. Minus 20 mesh fused UO_2 absorbs an equilibrium concentration of 40 ppm water at room temperature in 40% relative humidity air. Water content of UO_2 removed from a swaged rod was also 40 ppm. Unexpectedly, water content of several UO_2 samples dried in air at 100 C before analysis was greater, 70 ppm in one instance. The water content of a fused UO_2 sample removed from a swaged rod was also higher, 120 ppm after drying at 100 C. This behavior suggests that heating in the presence of water vapor aids in the formation of a hydrated surface layer, possibly through oxidation of the UO_2 at the surface. It is concluded that attempts to dry UO_2 by heating in moist air as reported by other investigations may do more harm than good.

Fuel Evaluation. A "non-clad" irradiation test element consisting of high density sintered UO_2 pellets in a porous sintered stainless steel cladding was fabricated. This test will provide data on fission gas release from "non-clad" UO_2 during irradiation.

Remote Fabrication Studies. Future remote refabrication development studies will involve television-monitored vibrational fuel compaction equipment. A TV vidicoan tube was exposed to a sonic vibration field typical of those expected in the development program . An evaluation, performed by the tube manufacturer, revealed that although the useful tube life was subnormal, no damage could be attributed to the high energy sonic vibration. Design of a welding apparatus for the remote fabrication studies is nearly complete.

Fuel element end closure components are usually cleaned by acid etching and washing before welding in the Magnetic Force welder. However, cathodic etching of the tube end in the M-F welder, before welding, may be a more attractive approach. Preliminary work has shown that SAP and Zircaloy can be satisfactorily cleaned by this method.

Magnetic-Force SAP Welding. Twenty-six SAP specimens, welded at optimum welding parameters, were tested in tension. The tensile strength of each weld was within 95 pounds of the average T.S., which was 1440 pounds. Available ultrasonic testing methods do not reveal differences between a partially bonded joint and a completely bonded joint.

Hot Swaging. Twenty Zircaloy-clad fuel rods containing -20 mesh fused UO_2 were hot-swaged to approximately 93-94% T.D. Densities were uniform and surface finishes excellent. Ovality and diametral variation was less than 0.001". Total time required for loading, cold-swaging (two passes), and hot-swaging (final pass), was less than 10 minutes per rod.

Fabrication Studies. Ten vibrationally compacted UO_2 specimens were fabricated for joint HAPO-BMI high temperature isostatic pressing studies. The 16" long, stainless steel clad rods comprise mixtures of fused UO_2 and micronized UO_2, compacted to densities between 65 and 80% T.D.

Sintered and crushed UO_2 was vibrationally compacted to 88% T.D. (>92.5% compaction efficiency) in AISI-406 tubes (0.563" OD x 12" long x 0.009" wall). The tubes were coupled to a 9-foot long resonant beam supported horizontally between a fixed fulcrum and a 5000-pound-thrust vibrator.

The fuel material was prepared from sintered pellets (95% T.D.) of 3% enriched UO_2 by crushing, followed by "vibrational milling" in a container mounted on the large shaker.

PRTR Fuel Elements. The document, "Specifications for Vibrationally Compacted UO_2, Nested Tubular PRTR Fuel Element - Mark IIC," HW-70999, was written.

The thermocoupled, PRTR Mark I, cold-swaged UO_2 fuel element was discharged from the reactor after approximately 195 MWD/T_U exposure. Coding of the thermocouples was confirmed after discharge and disassembly of the element was initiated. Post-irradiation destructive examination will correlate fuel structure and heat flux with the measured fuel temperatures.

Mark I, cold-swaged PRTR fuel elements have accumulated exposure in excess of 1000 MWD/T_U.

Corrosion and Materials Studies

PRTR Water Quality Studies. The PRTR secondary system water treatment scheme is being reviewed with respect to present methods of operation and possible equipment modifications. The major consideration at present is the feasibility and desirability of using unfiltered Columbia River water

as the influent to the softeners (rather than the present 300 Area water supply) to decrease the ultimate residual alkalinity level in the boiler. Preliminary calculations indicate that this change would reduce the alkalinity by a factor of about two, but this is still not enough to bring the boiler water pH-phosphate relationship within the specified limits for moderate phosphate concentrations. The primary disadvantage of this scheme is that temporary or permanent "plugging" of the zeolite softeners might occur if unfiltered water is used.

Operating data indicated that one of the softeners was becoming exhausted at impurity loadings of only 40 to 60% of the initial softener capacity. These data imply that either the softener capacity had decreased, that regeneration was incomplete, or the unit was channeling. The unit was opened for inspection, and it was observed that channeling was occurring. The distribution system piping was corrected, and softener regeneration and exhaustion curves were prepared and compared with the other units. After this change, the unit returned to essentially normal capacity.

Carbon Steel Corrosion in 300 Area Water. The corrosion of carbon steel in 300 Area water is being investigated to determine the extent of corrosion which should be expected in the PRTR carbon steel piping exposed to this water. Coupon tests in refreshed 300 Area water for 600 hours at 45 C showed distinct anodic areas on uncoupled specimens with penetration rates in the anodic areas of about 20 to 30 mils/year. Galvanic coupling to large areas of stainless steel increased the penetration rate to 75 to 100 mils/year.

Preliminary examination of a section of carbon steel pipe in high flow service for about one year did not show any extensive pitting. The uniform corrosion is unknown because of the manufacturer's 12% allowable variation in wall thickness. Examination of a small carbon steel heat exchanger tube in service for about one year in the same water showed a penetration of greater than 50 mils/year in the form of several large pits. The flow velocity in this system was moderate. Another carbon steel heat exchanger is being examined for pitting rate after two years in service; however, results are not yet available.

Lithium Hydroxide Corrosion. Metallographic examination of a PRTR Type 316 stainless steel valve which failed after one month of service in pH 10 water (LiOH) at 300 C, showed the valve body to be severely cracked. Cracking occurred in a region where concentration of the lithium hydroxide would take place from leakage. The cracking was predominantly transgranular in nature and resembled chloride stress-corrosion cracking.

Corrosion of Stainless Steels and Nickel Base Alloys. Corrosion rates of various stainless steel and nickel base alloys in 550 C, 3000 psi steam are currently being investigated. Following 174 hours of exposure, the corrosion rates appear to fall into four separate groups. An experimental Fe-Cr-Al alloy (5.6% Al-24% Cr), Hastelloy "X", and 446 stainless steel corrode most slowly, rates ranging from 3.46 mg/dm^2 to 7.59 mg/dm^2 weight

gain. Hastelloy "N", Inconel "X", PDRL-102, and Incoloy are slightly higher than the first group and range from 10.5 mg/dm^2 to 12.1 mg/dm^2. 304 L stainless steel and Carpenter 406 stainless steel have similar weight gains of approximately 93 mg/dm^2. The remaining two heats of 406 stainless steel (Allegheny Ludlum and Weldrawn Tubing), 430 stainless steel, and experimental Ferral (7.6 Al-5% Cr) have the highest weight gains ranging from 148 mg/dm^2 to 186 mg/dm^2.

A comparison with previous results obtained for some of these alloys in 550 C, 1000 psi steam has shown no effect in raising the pressure from 1000 psi to 3000 psi.

Zircaloy Pressure Tube Evaluation. Susceptibility to brittle failure of pressure vessel and pressure tube materials is of grave concern in both conventional steam plants and nuclear reactors. In pressure-tube type nuclear reactors like the NPR and PRTR there is an opportunity to monitor the effect of total operating environment on the properties of the pressure-containing member of the system. A review of the many tests in use to predict safety from brittle fracture did not reveal any obvious method for obtaining results that would have direct meaning in establishing operating limits for pressure tubes. From an operating standpoint, the test should relate pressure and temperature to safety from brittle fracture. It was concluded that the closest approach to this ideal test would be to introduce a "brittle crack" into the tube wall while the tube is pressurized at operating temperature. A test has been devised wherein a crack is introduced by firing a shaped projectile into the wall of a tube-specimen held at a controlled temperature and pressure. The ratio of the critical pressure at which the crack does not propagate and the ultimate burst pressure is a measure of the degree of safety from brittle fracture.

An existing burst test facility consisting of an eight-inch diameter furnace with appropriate temperature and pressure controls was modified for this test. A standard "Ramset" powder actuated stud gun was fitted with a water cooled barrel extension and mounted on the furnace. Special projectiles were made with an elongated diamond-shaped cross section chisel, the major axis of which matched the bore of the gun barrel (22 cal.). Development is still proceeding, however, to establish a preferred size and shape of the projectile.

Tests for susceptibility to brittle fracture as described above have been made on KER process tube specimens containing 65% cold work. Projectiles were fired into specimens held at a given percentage of their bursting pressure and at temperatures of 20 C, 150 C, and 300 C. The results were then evaluated by measuring the length of the fracture as a percentage of the specimen length. In all tests performed thus far the projectile has either produced a mere puncture with no propagation or has caused the specimen to split full length. As predicted by the theory, it has been possible to determine at each of the three temperatures an approximate pressure below which a suddenly introduced brittle crack will not propagate:

Temperature	Critical Pressure, psi	Ratio of (Critical Pressure) / (Burst Pressure)
20 C	4800	0.35
150 C	7700	0.65
300 C	9800	1.00

With these encouraging results, work has started to isolate the role of the many variables that may affect the results in the absence of exposure to operating environment. An extensive series of tests will be performed on unirradiated KER, PRTR, and NPR tubes. These will cover the effects of such variables as uniform hydrogen content, localized hydrogen of high concentration, percent of cold work, wall thickness and initial crack length and shape.

An area 1-1/4 inch x 1/2 inch was hydrided on the thin-walled side of two lengths of Zircaloy-2, KER-size pressure tubes. Burst pressures at both 150 C and 300 C were virtually unaffected by the presence of up to 4000 ppm of hydrogen in a local area. Because the hydrogen does increase the strength of zirconium, most of the elongation occurred in the unhydrided regions surrounding the local areas of massive hydride which were virtually unchanged by the burst test.

A prototype burst chamber for testing irradiated pressure tubing at elevated temperatures is nearing completion. After debugging for normal testing, the chamber will be modified to permit firing "brittle cracks" into irradiated tubes. The "critical pressure" measured by the above test on tubes periodically removed from both the PRTR and the NPR is expected to provide a measure of the effect of total reactor environment on the susceptibility of the pressure tubes to brittle fracture.

04 Program Radiometallurgy Studies

Metallographic examination of the core from an 86 w/o ThO_2 - 14% w/o enriched UO_2 element revealed some recrystallization of the thorium oxide (RM 632). The maximum warp of three-foot long UO_2-PuO_2 rods (GEH-11-3) was 55 mils. Examination of the cores from three of the rods revealed that no significant recrystallization of the core material had occurred (RM 668).

X-ray analysis of the cladding from the rupture site in the inner tube of the nested tubular fuel element showed that the Zr-2 cladding had transformed to ZrH by hydrogen pickup. Localized recrystallization of the adjacent UO_2 fuel further substantiated the presence of a hot spot on the inner cladding surface (RM 637). X-ray diffraction of uranium oxide samples revealed a minor amount of beta uranium in the UO_2 (RM 634). Six UO_2-PuO_2 capsules were returned from the MTR after the second irradiation. Visual examination and dimensional data indicated that the capsules were in good condition (RM 653 & 654).

Six samples were dissolved for burnup analysis including UO_2, uranium, PuO_2 and stainless steel. Density was determined on two samples of PuAl. Hardness and microhardness were determined on Zircaloy-2 process tubing, PuAl (before and after annealing), UO_2, and mixed PuO_2-UO_2.

Results and interpretations of these examinations will be reported in more detail in connection with the development programs served.

2. PLUTONIUM UTILIZATION STUDIES

Plutonium Dioxide

Several experiments were run during the past month to obtain data on the equilibrium oxygen pressure over PuO_2 in hydrogen atmospheres. These experiments will provide additional data on PuO_2 stability and will eventually be run on UO_2-PuO_2 mixtures to determine the structural behavior of this "alloy" as a function of composition, temperature, and partial pressure of oxygen.

Some of the early experiments run in oxygen pressures of from 10^{-13} to 10^{-27} atm have shown that partial reduction of PuO_2 to Pu_2O_3 is quite temperature dependent. As noted previously, reduction to compositions near $PuO_{1.98}$ occurs at temperatures as low as 1000 C in nearly all atmospheres, while about 1250 C and very low oxygen pressures are needed for lower O/Pu ratios. Also, the densification-temperature relationships and the effect of oxalate calcination temperature on stability and densification are being investigated. In the experiments run to date, calcination temperature appears to have no effect on either of the above two variables. The two calcination temperatures used were 300 C and 550 C in an air atmosphere. The density data obtained has followed the parabolic relation normally seen on density versus temperature plots. Considerably more data will be obtained on this experiment.

Plutonium Carbides

A series of ten Pu-C alloys extending from 30 to 55 a/o C were arc melted in gettered argon for additional data on phase boundaries in this system. The carbon analyses, determined by combustion, agreed very well with the plot obtained from previous data on the percent C lost during melting. The precision in these analyses was good and duplicate and triplicate determinations frequently agreed to within 0.5 a/o C.

Back reflection x-ray patterns gave arc-melted lattice constants which were in agreement with those previously found with the exception of the region 36-41 a/o C. The discrepancies seen in this range again indicate some influence by the zeta phase on the PuC cell size. Samples containing less than 40 a/o C were annealed at 500 C (epsilon plus zeta region) for 83 hours and slowly cooled. Alpha plutonium plus zeta were the two phases observed. The lattice parameter of the zeta phase based on the highest angle planes observed and on cubic (NaCl) symmetry was 4.968 ± 0.001 A. Quenching these specimens from 530 C gave alpha plutonium plus PuC (a_o =

4.9702 ± 0.0003 A), with no trace of zeta. This temperature is slightly below the epsilon plus PuC field reported by Mulford and temperature measurement in the system is presently being recalibrated. A series of 15 Pu-C alloys has just been annealed at 8×10^{-7} mm Hg for 91 hours in the epsilon plus zeta region. Powder patterns should provide data on the equilibrium lattice constants of the PuC phase and should establish the room temperature composition limits of the zeta and PuC phases.

A 500-gram plutonium button was reacted with 20 mesh graphite in a tungsten crucible four hours at 1600 C. The temperature control on the resistance furnace was not working and the charge was brought up to temperature too rapidly. The exothermic reaction evidently splattered the charge up the sides of the crucible, for pieces of PuC were found high on the sides of the crucible while a mass of metal was in the bottom. Analysis showed this metal to contain about 2 w/o C, and it had reacted with the tungsten crucible. A number of melts using the arc melter were made in a water-cooled copper hearth. When eight mesh graphite was melted with a plutonium button, some graphite was invariably lost due to arc blow; however, when a solid piece of graphite was used, little loss was noted (average yield of 98.3 percent on material balance). In order to make a sound, apparently homogeneous, button in this matter it was necessary to remelt five times. The homogeneity of the buttons are currently being checked by x-ray diffraction and carbon analysis.

Plutonium Silicides

Synthesis of $PuSi_2$ was accomplished by heating compacts of PuO_2 and Si for 30 minutes at 1385 C in vacua. Lattice parameter values were calculated from the (312) and the (1, 1, 10) diffractometer peaks. Values for a_o and c_o were respectively 3.97 and 13.59 A. The c/a ratio is 3.42.

Prior to a density determination, the $PuSi_2$ pellet was arc melted in argon. A value of 8.47 g cm^{-3} was found by immersion in tetrabromoethane.

Plutonium Nitrides

Success was achieved in forming PuN by arc melting alpha-plutonium under a static atmosphere of nitrogen. Diffraction data agree with the ASTM data card for PuN except for a few extra lines, as yet unidentified.

An initial density determination yields a value of 15.47 g cm^{-3}. This exceeds the theoretical density, 14.20 g cm^{-3}, based on a lattice parameter of 4.909 A. The lattice parameter was calculated from the (600) line on the diffractometer chart. The high density may be due to the presence of residual alpha plutonium, or perhaps PuN exists over a range of compositions.

In another attempt, PuH_3 was reacted with anhydrous NH_3 at 550 C for 30 minutes. Argon was used as a furnace purge and carrier for the ammonia. PuN was not formed, and the hydride was converted to PuO_2. Either an air leak into the furnace or gas impurities could have caused the oxidation.

Gas Evolution

Data reported previously, showing an increase in evolved gases with an increase in particle size, have not been duplicated by experimentation at higher temperatures. The blank value for the Ta crucible does not appear to be reproducible at temperatures exceeding 1500 C, thereby making it difficult to achieve reliable high temperature data. Investigation is presently under way to remedy this difficulty.

ZrO_2-PuO_2 and MgO-PuO_2 Irradiation Capsules

An irradiation test proposal, HW-71713 RD, for these 16 capsules has been prepared, and the samples have been sent to the MTR for irradiation. It is planned to start irradiation of these capsules during MTR cycle 169 which commences January 29.

3. UO_2 FUELS RESEARCH

US/UK Newsletter

US/UK Newsletter No. 12, "Uranium Ceramic Fuels," was compiled and distributed.

Uranium Carbide-Cladding Compatibility

Niobium carbide was formed (< 3 minutes) when Nb was heated in contact with $UC_{1.05}$ at 2250 C. The hyperstoichiometric carbide dissociated at this temperature, and carbon reacted with niobium to yield a series of well-defined reaction zones. These were (from the outer surface of the cladding toward the UC-Nb interface): (1) niobium metal interspersed with Nb_2C platelets, (2) pure Nb_2C, and (3) NbC with Nb_2C inclusions. The $UC_{1.05}$ was decarburized to $UC_{1.00}$ to a depth of approximately 0.010".

Characterization of UO_2

A report, "High Density UO_2 for Nuclear Fuel Applications," HW-SA-2345, was presented at an AEC-sponsored symposium on "Characterization of UO_2," at Oak Ridge National Laboratory on December 12 and 13, 1961. The report describes the properties of fused, electrodeposited, and high-energy-impact-formed UO_2, and relates these properties to predicted and observed phenomena occurring in swaged and vibrationally compacted fuel elements during irradiation.

Hardness Measurements

The hardness of UO_2 single crystals decreased approximately 50 KHN after annealing for 12 hours in 1750 C hydrogen. No change of microstructure was detected. The hardness values measured along a diameter of the cross-section surface of an irradiated UO_2 fuel rod are scattered. However, these preliminary data suggest that the hardness of irradiated UO_2 is greater than that of non-irradiated UO_2.

Uranium Metal Formed in Irradiated UO_2

Inclusions in irradiated UO_2 (GEH-14-189) were tentatively identified as β -uranium by x-ray diffraction analysis. The inclusions appear in both the porous columnar grains and in the small columnar grains. The pore-free regions of the large columnar grains are relatively free of inclusions. These small inclusions are similar to those observed in non-irradiated, fused UO_2 crystals.

Irradiation of Heterogeneously Enriched UO_2

Preliminary examination of the four-rod cluster (GEH-4-62) that had severe inhomogeneity of enrichment in the UO_2 fuel revealed that highly enriched (93 w/o U-235 in U) UO_2 "pellets" caused no adverse affects during irradiation.

Visual examination of the element revealed cladding discolorations, indicative of hot spots, near the original position of highly enriched, approximately 1/8" diameter UO_2 particles. No evidence of catastrophic damage was observed on the cladding.

A cross-section of one rod near the original location of the subject particle revealed a large centrally located void surrounded by radially oriented, large columnar grains. A second cross-section, approximately 1/8" distant from the first, disclosed a smaller central cavity and associated columnar grains. A third section, approximately 5/8" from the first, revealed no evidence of extensive grain growth or in-reactor sintering. Examination is continuing.

Gross relocation of the highly enriched UO_2 apparently occurred rapidly during the early stages of irradiation. The visual evidence indicates that rapid homogenizing of the enriched UO_2 and the coarse, naturally enriched UO_2 particles occurred in a manner that distributed the heat generation load throughout a large, centrally located fuel volume, i.e., a self-regulating process occurred that relieved severe hot spots near the cladding.

Electron Microscopy

Specimens were selected from an irradiated UO_2 fuel core (previously used for fission product distribution studies) for reflection electron microscopy examination. UO_2 crystals were submitted to L.K.B. Instruments, Washington, D.C., for evaluation of their microtome equipment as a tool for preparing thin sections of UO_2 for transmission electron microscopy.

Fission Fragment Migration

Analyses of additional UO_2 samples removed from the cross-sectional surface of an irradiated UO_2 fuel rod (1.44" diameter, 300 MWD/T) confirmed previous observations of radially non-uniform fission fragment distributions.

UO2 Conductivity

Specimens for the continuing HAPO thermal conductivity program measurements at BMI will include: (1) irradiated, fused UO_2 single crystals (to 1200 C), (2) an irradiated (approximately 0.12 a/o burnup) sintered UO_2 rod (to 900 C), and (3) a non-irradiated UO_2 disc specimen (to 2500 C). This work will be a continuation of the HLO/BMI program of 1958-1961. A report of the latter program is in preparation.

4. BASIC SWELLING PROGRAM

Irradiation Program

Capsules numbered 13 and 14 are scheduled for charging in tandem at the next reactor shutdown. A new instrumentation facility at the reactor has been tested, and minor modifications and corrections are being made to the thermocouple circuits and capsule heater power supplies to accommodate these capsules and future capsules of this design. A previously assembled capsule (#9) was used at the reactor to test this instrumentation. Two additional capsules (11 & 12) are also at the reactor and will be charged in tandem at a subsequent shutdown.

Laboratory tests were successfully completed on capsule #10 after repairs were made to the heater and a small leak was eliminated on the inner chamber. Capsules # 7 and 8 are in the reactor discharge basin to permit radioactive decay before shipment to radiometallurgy for disassembly and post-irradiation examination of the specimens.

Two of the four unmonitored NaK-filled capsules under irradiation at the MTR were discharged after two cycles of exposure to a flux of about 5×10^{13} nv. The two remaining capsules will receive an additional two cycles. Each capsule contains six U-U diffusion couples for fission product mobility studies by post-irradiation annealing.

Post-Irradiation Examination

The radiometallurgy examination of the specimens from capsules 4, 5, and 6 is nearing completion. The density measurements have now been completed on all of the specimens that have been processed for metallography. These values are summarized in the following table:

Swelling of Specimens from Capsules 4, 5, and 6

Spec. No.	% Burnup	As-Irr. Condition °C	Density	% Swelling	As Annealed Condition (100 hrs) °C	Density	% Swelling
4-2	0.28	615	17.42	7.9*		Not yet annealed	
4-3	0.28	615	17.40	8.0*	615	17.03	10.4
4-4	0.28	575	17.61	6.8*	575	17.40	8.1
5-1	0.05	575	18.44	1.9*	575	18.35	2.4
5-2	0.16	800	14.85	34.7		Not yet annealed	
5-3	0.05	550	18.35	2.5*		Not yet annealed	
6-1	0.03	330	18.96	0*		Not yet annealed	
6-2	0.03	330	18.98	0*	575	18.80	0.9
6-3	0.03	260	18.97	0*	615	18.44	2.7

*previously reported.

All of the new density values reported in the table were obtained using carbon tetrachloride as the immersion media instead of tetrabromoethane. Hence, somewhat poorer accuracy can be expected. Annealing the specimens caused apparent density decrease. In the case of the specimens in capsules 4 and 5 the results are probably not significant due to the poor condition of the samples. In the case of specimens 6-2 and 6-3, however, these results probably are significant and the magnitude of the changes tend to corroborate the metallographic observations of porosity. Specimens 4-3, 4-4, 5-1, 5-2, and 6-1 were dissolved for burnup analyses. Specimen 4-2 is being annealed at 650 C for 100 hours to determine if the pores will grow when exposed to a temperature higher than the irradiation temperature.

Pore Size Distributions

Computer calculations of pore volume fraction and pore density, as a function of pore size have been obtained from measurements of 200 electron micrographs. The distribution curves pertain to a variety of post-irradiation annealing treatments of uranium which had been irradiated to two burnup levels, 0.29% and 0.41%, respectively. The shapes of the distribution curves, their moments, and similar characteristics are being intercompared.

The validity of pore volume fraction and pore density estimates by quantitative metallography depends upon proper specimen preparation techniques. If pores revealed by electron micrographs are altered during specimen processing, final results will be indicative only of the altered pores and not of the actual pores present in the uranium. The mathematical relationships being used include a correction term for an extrusion distortion of the replica, associated with replicating pores cut above their center. In order to evaluate how etching affects pores, an irradiated specimen was etched for 20 minutes, replicated, etched for an additional 70 minutes and replicated again. Electron micrographs were prepared of identical specimen areas from the two sets of replicas and pore size distributions were determined.

Statistical comparisons of the data indicate that the total pore volume fraction values are slightly higher (10%) for the prolonged etching time, but that the pore density values are the same and the shapes of the distribution curves are quite similar for both etching times. Thus it is concluded that prolonging the etching time does not introduce major errors in the metallographic estimates of swelling. Care must still be taken, of course, to assure that an adequate etch is employed to eliminate the worked metal from the specimen surface that arose as a result of the polishing operation.

Fission Product Mobility

One of the U-U diffusion couples, GEH-14-281, which had been annealed at 500 C for one hour was processed for light and electron microscopy. No change in structure was detected with either method. In view of the lack of structural change after annealing, a microhardness traverse across the U-U interface was not taken. The annealing treatment, if sufficiently high in temperature, should cause pore growth in the immediate vicinity of the enriched-depleted U-235 interface. If pores form in the depleted portion of the couple at considerable distances from the interface, they will permit study of fission gas mobility in uranium. Extensive study of the enriched and depleted U-235 zones showed no evidence whatsoever of fission gas pores. This is somewhat surprising in that the expected burnups in the U-235 enriched and depleted zones was to have been 0.3% and 0.015%, respectively. Uranium with such a high burnup (0.3%) should have yielded porosity during the post-irradiation annealing treatment. The specimen is now being re-annealed at 650 C for 100 hours for development of porosity.

Restrained Irradiations

In order to gain insight into the influence of Zircaloy-2 cladding on the swelling of uranium, Zr-2 clad uranium rods with selected cladding thicknesses, are being irradiated in NaK-filled, temperature monitored capsules at selected uranium temperatures to a series of goal exposures.

After the cladding was removed, density determinations were made on four of 22 such rods recently irradiated in the MTR and ETR. The irradiation conditions and the measured volume changes are summarized for the four rods in the following table:

Swelling in Zr-2 Clad Uranium

Rod No.	Irradiation Temp., °C	Burnup %	Swelling % Δ P/Pf
98-1	525	0.61	7.42
104-1	505	0.65	6.85
105-1	270	0.65	0.34
105-2	290	0.65	0.22

The swelling per atom percent burnup in the uranium irradiated at the higher temperatures corresponds very closely with values determined previously from clad uranium irradiations.

Second Phase Distribution in Dilute Alloys of Uranium

It is important to the fundamental study of swelling that the role of second phase particles in the uranium matrix be understood. It has been reported that the "delta phase" in uranium alloyed with small amounts of silicon and iron is quite effective in controlling the grain size. It has also been reported that alloying uranium with small amounts of iron and aluminum substantially reduces the swelling that occurs in uranium during irradiation. It is possible that iron and aluminum enhance the uranium by virtue of a finely divided precipitate that acts to promote the formation of many small gas pores.

A study has been initiated using the electron microscope to examine this "delta phase" with regard to the influence of structure, composition, and thermal treatments. Three materials are now on hand and samples are in process of being heat treated. These materials are high purity dingot uranium, dingot containing 150 ppm Fe and 100 ppm Si, and dingot containing 80 ppm Fe and 69 ppm Si. Should an appropriate precipitate be observed that is stable at normal swelling temperatures, samples will be irradiated in temperature controlled capsules to determine the effect of the precipitate on swelling.

5. IN-REACTOR MEASUREMENT OF MECHANICAL PROPERTIES

In-Reactor Creep Measurements

An in-reactor creep test is in progress at 310 C and 30,000 psi stress on a 20 percent cold worked Zircaloy-2 specimen in a capsule modified with water cooling jackets. The capsule, modified by installing water cooling jackets immediately over the heaters, has provided the necessary cooling to maintain a 310 C test temperature previously unobtainable in helium cooled creep capsules. Over 1000 hours of continuous testing have now been completed with the capsule. All measuring and control systems in the capsule are still functioning properly. During this period of testing two reactor shutdowns occurred. The strain measuring systems provided continuous readings during the reactor outages, but the temperature on the second shutdown was not uniformly maintained. Temperature control at the first shutdown was maintained at exactly 310 C with only a minor five-degree depression for a few minutes as the controllers responded to the new power demand condition. The second shutdown, however, recorded a twenty-degree depression which did not recover until the reactor resumed operation. Minor adjustments of the controllers corrected the low temperature condition initially, but additional changes in cooling water temperature and flow rates resulted in further temperature depressions which, if observed in time, could have been corrected with existing control equipment. Temperature control for the test thus far, with the exception of the second reactor shutdown, has held the end-to-end gradient on the specimen to less than 1 C.

Results of the creep test show that the creep rate appears to be less under neutron irradiation than ex-reactor for the 20 percent cold worked Zircaloy-2 specimen. When irradiation ceases during reactor shutdowns, marked increases in rate are observed. The resulting strain after reactor cycles is greater than that found in an unirradiated test. The total plastic creep strain on the specimen at 1000 hours was 0.742 percent as compared to a total predicted strain of 0.38 percent for the ex-reactor test. Of this total 0.742 percent, 0.43 percent occurred during the first shutdown and 0.14 percent occurred during the second shutdown. The creep rate just before the first shutdown was 1.66×10^{-6} in/in/hr which went to 1.35×10^{-4}, a 100-fold increase, immediately after the shutdown. The rates and total strain during the second shutdown were lower than the first due to the temperature depression but still accounted for a significant contribution to the total plastic strain. These results, while internally consistent, required corroboration by additional capsule tests.

Reviewing the results from our previous capsules and those reported at other sites, this increased creep occurring immediately after reactor shutdown may well have occurred but was either not detected or properly interpreted. Insensitivity of measuring systems and the immediate upset of temperature conditions and system calibrations at shutdown left too much doubt as to what was being observed. The accumulated strain, from our own capsules, did show an increase in the total strain over the control specimens leading to a calculated rate that was greater than the ex-reactor rate. This calculated rate accounted for the observed increase in total strain occurring over several reactor cycles; however, much of the total strain may well have occurred during the shutdown.

Capsule and Instrument Development

The next capsule scheduled for testing is also being modified with water cooling jackets and is being tailored to operate at a temperature between 250 and 300 C. The capsule also contains a 20 percent cold-worked Zircaloy-2 specimen, with the test scheduled to obtain the variation in rates and activation energies at the lower temperature.

6. GAS-GRAPHITE STUDIES

Graphite-Water Vapor Reaction Under Gamma Irradiation

Measurements in the cobalt-60 gamma facility of the rate of oxidation of TSX graphite by He containing a small partial pressure of water vapor have continued. Two additional series of rate determinations at water vapor concentrations of about 90 and 300 ppm respectively were completed. Both employed a solid cylinder of TSX graphite weighing approximately 8.3 grams. The net oxidation rates obtained are compared below with those found previously using this same graphite sample but with a H_2O vapor concentration of about 155 ppm.

Temp., °C	Oxidation rate, gm/gm/hr x 10^7		
	H_2O conc.=90 ppm	H_2O conc.=155 ppm	H_2O conc.=330 ppm
600	--	5.29	5.42
700	7.12	8.90	10.13

A plot of these rates versus the water vapor concentration exhibits an entirely different behavior than a similar plot using data for the thermal reaction of TSX graphite. The latter shows an essentially linear relation, whereas in the former case the oxidation rate at a given temperature tends toward a saturation value at the higher water vapor concentrations. It thus appears that at the higher concentrations the radiation-induced reaction is somehow limited by the number of active species reaching the graphite surface. Further measurements are planned to attempt to clarify this point.

Irradiation of EGCR Graphite

The H-3-3 capsule was discharged from the E-7 position of the GETR on November 27. The capsule was disassembled and the samples were returned to Hanford for measurement. The capsule and all samples were in excellent condition. After measurement of dimensional changes, some of the samples were re-installed in the H-3-4 capsule which will be returned to the GETR for five cycles of irradiation.

The EGCR graphites (NC-7 and NC-8) and the reference graphite (CSF) in the H-3-3 irradiation continued to contract approximately linearly with exposure. The maximum exposure on the samples is now approximately 8.7 x 10^{21} nvt, or 58,000 EGCR equivalent MWD/AT ($E \geq 0.18$ Mev). The minimum contraction rate was observed at about 800 C. At this temperature the contraction at the peak exposure was 0.55% for CSF (transverse), 1.2% for CSF (parallel), 0.30% for EGCR graphite (transverse), and 1.2% for EGCR graphite (parallel).

Carbon Dioxide - Hastelloy-X Compatibility

The temperature of the Hastelloy-X tubes in the gas heater of the PRTR gas loop will approach 1800 F when the outlet gas temperature is 1500 F. The rate of oxidation of Hastelloy-X in CO_2 is being measured by periodically weighing a tubular sample maintained at 1850 F by induction heating. A linear weight gain of 0.003%/hr was observed during the first 300 hours. The oxidation test will be continued for an additional 200 hours.

7. GRAPHITE IRRADIATION DAMAGE STUDIES

Irradiation of Lampblack Carbon

To date no evidence of saturation of radiation-induced contraction in graphite has been observed. Several materials that are expected to contract at an accelerated rate are being irradiated to identify factors that affect the contraction rate and to search for evidence of saturation. Carbon samples manufactured from a lampblack-binder mixture and heat-treated

to 1400 C were irradiated in the ETR at a temperature above 600 C. The foil monitors included with the samples have not yet been measured, but the exposure is estimated to be equivalent to approximately 6000 MWD/AT. The samples contracted 2.2% transverse to the grain and 2.7% parallel to the grain, which is equivalent to a volume contraction of 7.1%. This is approximately one-fourth of the volume contraction necessary to attain theoretical density. The samples will be recharged for another irradiation to about the same exposure.

8. ALUMINUM CORROSION AND ALLOY DEVELOPMENT

Aluminum Corrosion Test Program in H-1 Loop

A Zircaloy-2 liner tube has been cut to length and slotted and is ready to be installed in the H-1 Loop. The loop was operated on out-of-reactor recirculation to check out the new pumps and instrumentation. Coupon holders and spacers for the first charge were completed. On being nickel-plated some aluminum coupons containing silver foils swelled during baking at 300 or 400 C. Although attempts to duplicate this swelling in other coupons were not successful, this has introduced concern that some coupons containing silver foils may have leaks in the welds which will admit water and possibly release Ag-110 activity during loop operation. Alternate materials to produce high beta activity were investigated; rhodium and vanadium were selected as having no long-lived activity.

A description of fuel elements to be used in corrosion tests of active elements was submitted to Plutonium Metallurgy Operation. The elements will be 0.94-inch diameter by 7.5-inch long, and employ cores of 20 w/o Pu in Al coextrusion-clad with either 0.050-inch or 0.130-inch aluminum. Orders were placed for bar stock of three aluminum alloys to be tested as cladding: X-8001, A-288 (1% Ni, 0.5% Fe, 0.1% Ti and low Si, and an alloy with 1.8% Fe, 1.2% Ni, and low Si. A preliminary design was made of casks required for the discharge of these elements.

Corrosion in Water at pH 4.5 Inhibited with Chromic Acid

A test evaluating the corrosion of X-8001, A-288, 304 SS, A212 carbon steel, and Zr-2 in pH 4.5 chromic acid was initiated in ex-reactor loop TF-6 during the month. The first discharge of samples was made after two weeks of testing at 300 C, and although corrosion data are not yet available, visual examination indicates fairly good protection. The carbon steel shows no pitting and is covered with a thin film of dark brown oxide. The aluminum corrosion film appears adherent and the underlying metal is free of pitting. Loop water analyses have been low in Fe and Ni indicating low steel corrosion. The test is continuing.

Heat Transfer Testing

Corrosion testing of X-8001 cladding continued during the month. The test is being operated at a bulk water temperature of 600 F using high purity water. Testing has now reached about 1500 hours. The temperature drop

across the water, oxide, and crud films continues to rise gradually between coupon discharges and decrease to a lower value following startups. Evidently thermal shocks occur which flake off corrosion product oxide at these times. The over-all increase in the temperature drop is now about 50 F.

9. USAEC-AECL COOPERATIVE PROGRAM

Ultrasonic Test for Sheath Tubes

After a series of meetings with visitors from Canada and other experts in the field of ultrasonic testing, the sensitivity of the ultrasonic test for Zircaloy sheath tubes was increased to detect defects in the range of 1 mil deep and 30 mils long. Addition of a peaking coil tuned the crystal to give a greater response for a given input energy. A new crystal holder, on order for several months, permits precise adjustments of the angular relationship between the crystal and the tube. Utilizing published ultrasonic theory, angles of incidence were calculated. When applied to a small number of tubes, the refined test detected pits, gouges, impressed particles, and scratches on the inner surface of the tubes not detected by earlier ultrasonic tests.

CANDU Heat Transfer Studies

Fabrication of the second electrically heated 19-rod bundle test section for boiling burnout experiments under the USAEC-AECL Cooperative Program was completed. Each rod in this test section is 0.629-inch OD and has a heated length 19.5 inches long. Spacing between adjacent rods is 0.015 inch and is maintained at this value by 0.015-inch diameter wire wrapping on a 10-inch pitch around 12 of the 19 rods. The bundle will be placed in a horizontal process tube 3.25 inches ID. Data from laboratory tests with this test section are to be compared with previous results obtained on a test section with 0.075-inch spacing between rods to determine the heat transfer penalty, if any, of very close rod spacing.

10. IRRADIATION EFFECTS IN STRUCTURAL MATERIALS

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

Six specimen quadrants containing 36 tensile specimens and 48 bend test specimens of Zircaloy-2 were removed from the ETR 6x9 hot water loop after cycle 42. These quadrants were concurrently replaced with quadrants containing 72 bend test specimens of Zircaloy-2. The total number of both Zircaloy-2 and stainless steel specimens discharged from the loop to date is 372. Of these, 288 were removed from their quadrants and placed in special containers for shipment to HAPO. Shipments will commence before the month's end.

The operating efficiency of the 6x9 loop has greatly improved during the first three cycles of operations and is now operating at design conditions. During cycle 42 the loop operated 86.0 percent of the time at the desired temperature of 540 F.

The tensile properties of Zircaloy-2 as a function of strain rate in the range 0.0025 to 2.5 min^{-1} were studied during the month. Within this range, the ultimate tensile stress is raised approximately 10,000 psi for each of the cold-work levels (from 0 to 70 percent), and follows the relation $\sigma = a + b \ln \dot{\epsilon}$; where σ is the ultimate tensile stress, $\dot{\epsilon}$ is the strain rate, and a and b are constants. The uniform strain decreases with increase in strain rate in the annealed condition but is insensitive to strain rate in the cold-worked states. The same general conclusions also apply to the necking strain. These results indicate that significant stress relaxation through creep occurs in Zircaloy-2 at room temperature. Effects of strain aging on strain-rate sensitivity at room temperature, as reported by others, have not been observed.

During the month bend tests were started on Zircaloy-2 and tensile tests were started on unirradiated control specimens of types 304, 348, and 410 stainless steel. A fixture for conducting tensile tests at 300 C was completed and satisfactorily used to test two Zircaloy-2 specimens at this temperature. However, a few improvements are required before extensive testing is initiated.

11. REACTOR STUDIES PROGRAM

Advanced Reactor Concepts

Two general categories of reactor types were considered during the report period: high temperature plutonium-fueled reactors and a chemo-nuclear reactor using fission fragment energies for fuel cell regeneration. The objectives in one of the plutonium-fueled reactor concepts were to attain very high working fluid temperatures (several thousand degrees F) in a compact system which is (1) self-controlling and failsafe, (2) capable of high power density (extended heat transfer surface), and (3) within the realm of materials feasibility. The materials problems at elevated temperatures and the heat transfer surface requirement suggest that a major portion of the reactor heat be generated with the fuel and coolant intimately mixed, and not in contact with any intermediate material (such as a heat exchanger). One method may be through mixing the molten fuel with a liquid metal (sodium) and accomplishing the heat transfer while "spraying" the mixture through an inert atmosphere. A possible configuration for the reactor core might be concentric cylinders or spheres (of heavy walled tantalum) in which the inner portion contains sodium at temperatures of about 1200 F and the outer portion molten plutonium. In this configuration the reactor would be subcritical. By pumping the sodium through nozzle type devices in a radially outward direction, the plutonium is aspirated with the sodium into an inert atmosphere in a larger, surrounding container. It is intended that the greater core "diameter" which results will bring the reactor critical. The heat generated in the plutonium

will be transferred to the sodium causing it to vaporize (~1600 F). Sodium vapor will be extracted from one end of the reactor facility for power generation; molten plutonium at comparable temperatures (for preheating the condensed sodium) from the other. The reactor is controlled by varying the flow rate (core diameter). Over-power results in dispersing the intimate fuel-coolant mixture, vaporizing additional sodium or superheating the sodium vapor, and tending to limit reactor power. Problems of core size and configuration, fuel and coolant solubilities and mixing, control dynamics, among others, remain to be investigated.

A second reactor type was considered in which fuel and sea water could be intimately mixed in a somewhat analogous manner to the previous reactor concept to result in steam and salts. No satisfactory configuration has been conceived which gives a usable end product (water and chemicals) free of contamination and permits the fuel to be easily returned to the reactor.

A chemo-nuclear reactor was considered for application in direct regeneration of fuel cell reagents via fission fragment energies. The lack of experimentally determined "G" valves for promising fuel cell reagents in a flowing system prevented drawing significant conclusions.

D. RADIATION EFFECTS ON METALS - 5000 PROGRAM

Work associated with establishing how neutron irradiation and carbon impurity levels affect the metal molybdenum has been primarily in the area of characterizing and preparing specimens for irradiation, and assembly of necessary laboratory vacuum and heat treatment facilities.

Several single crystal rod specimens having 10-20 ppm C and 400-500 ppm C are being ground into bar shapes having a central reduced square cross-section for use in tensile-deformation studies. These crystals have (110) surfaces parallel to their longitudinal axes.

Single crystals with polished faces as well as cold-worked polycrystalline foil specimens were annealed at 1250 C for four hours. Essentially no change in microstructure occurred. Moreover, Laue x-ray diffraction patterns from the polished surfaces of the single crystal specimens showed no sharpening of spots or rotation of the crystal. In order to reduce imperfections in the single crystals and to grow large, recrystallized grains with homogeneous carbon distributions in polycrystalline foil and rod specimens, higher annealing temperatures are required. Two types of high temperature furnaces are, therefore, being assembled: (1) an evacuable furnace heated by induction, and (2) an inert gas atmosphere furnace utilizing hydrogen protected molybdenum wire heaters.

Orientation determinations of a number of 1-1/2" long single crystal specimens for future study and irradiation have continued. Foils containing 100-200 ppm carbon and 400-500 ppm carbon in the as-rolled state, have been electrolytically thinned for transmission electron microscope studies. All specimens examined show preferential attack of the matrix metal around second phase carbide

precipitates. The carbide phase is not retained in the thinned regions of the foil. It is hoped that during proper thermal treatments many of the carbides will go into solution. Attempts to observe dislocation movement in these foils due to thermally induced stresses associated with electron bombardment in a double condenser electron microscope have now been successful.

Polycrystalline 1/2" diameter molybdenum rod containing 100 ppm carbon was successfully swaged to a 25% reduction of cross-sectional area. This was accomplished by sealing the molybdenum in an evacuated cold-drawn copper tube and swaging at approximately 600 C. Induction heating was used, and a marked temperature gradient existed across the diameter of the rod; since copper is a good suscepter and a good conducter, the heating was primarily in the copper and the center of the molybdenum rod did not reach 600 C. This resulted in a pronounced variation in grain size across the rod.

E. CUSTOMER WORK

Hydriding of Zircaloy-2

Seven Zr-2 specimens containing known quantities of hydrogen (as the hydride phase) were supplied to the Physical Measurements group for use as standards in their non-destructive hydrogen determination program. Seven Zr-2 test-loop tube sections have been hydrided for use as test specimens by the Structural Materials Development group.

Radiometallurgy Service

The point of initial water entry of a side-other rupture from 1581 H was determined to be in the immediate rupture area, but because cladding from that area was gone, the exact cause of failure could not be discovered (RM 434). An overbore element that was irradiated to more than 900 MWD/T is being examined to determine the cause of damage to three of the self-supported rails. A preliminary examination indicates that the rails were defected during the discharge operation or subsequent handling (RM 435).

Penetrations to the AlSi in two spots adjacent to a support tab were observed in an area of accelerated corrosion on an overbore element which has been irradiated to 950 MWD/T (RM 436). Metallographic examination of the transversely cracked process tube from 3077 H has shown that fatigue was the cause of failure. The appearance of this crack was similar to previously examined tube failures (RM 438). Two transversely cracked dingot uranium production fuel elements were examined. Initial sectioning indicated that an oxide layer ranging from 10 to 50 mils thick was present around the spire over the total length of each element (RM 437). Two samples of dingot uranium were dissolved for iron analysis.

Metallography Laboratories

An improved etch was developed for outlining the 5% beryllium-zirconium braze layer between Zircaloy components. The procedure uses 3% (by volume) HNO_3, 2% HF, and 95% glycerine to stain the brazed area after the specimen has been

polished. Previously, alcohol had been used instead of the glycerine, and the immersion time for proper stain etching was less than one second. That time has now been extended to four seconds which is far more controllable and produces more uniform results.

Carbon steel pipe for the NPR primary piping system is still being examined. A series of sections is being planned which will show the geometric nature of several small defects by alternately photographing a surface, grinding off 0.010 inch of metal, photographing the new surface, etc.

Graphite specimens which had been impregnated with lead and bismuth were polished for optical and electron microscopy. Impregnation with bismuth produced greater clarity and definition between the specimen and the mounting material than did either the lead or the usual plastic impregnant. Visual examinations as well as optical micrographs were enhanced. There appears to be no immediately observable advantage to replication techniques from this preparation, however. Both lead and bismuth cathodically etch much more rapidly than the graphite, and large shadows result from the "steps" formed in the replica at the metal-graphite interface.

Other work during the month will be reported in connection with the respective research and development programs served.

Samples Processed During the Month	
Total samples	1345
Replicas	28
Photographs	
Micrographs	594
Macrographs	239
Electron Micrographs	99
	932

NPR Charging Machine

All components of the N Reactor charging machine are operable. Check-out of the electrical system was completed. A number of deficiencies in the as-designed electrical system have been corrected. Installation of the 440-volt feed to the hydraulic power package was started. Modification of the end thrust stabilizer assembly was completed, and testing was started. Testing of the machine positioning capabilities for loading and unloading magazines from the wall racks was started also.

Servomanipulator Evaluation

Evaluation of an Italian-built servomanipulator for reactor application was completed for the AEC. The servomanipulator featured force reflection in the manipulator motions together with a high degree of mobility. It is somewhat comparable to the GEL II servomanipulator (the "Handyman") with respect to force reflection, and to General Mills "Little Ranger" or IPD's mobile servo-manipulator with respect to mobility. The Italian servomanipulator has been

proposed for general reactor service and will be used in a thorium fuels reprocessing plant in that country.

CVTR Startup Consultation

A visit was made to the CVTR site near Columbia, S.C., to furnish consultation to the Carolinas Virginia Nuclear Power Associates on reactor startup problems. The CVTR is a 60 Mw(t) pressure tube reactor having many similarities to the PRTR. Construction is scheduled to be completed in April 1962; startup procedures corresponding the the PRTR Design, Critical, and Power Tests are now being organized for preparation. In anticipation of a later completion date, there is a desire to minimize the startup test period. Discussions were centered on PRTR Design Tests, cleaning, flushing, and drying of systems, unexpected problems, and manpower requirements.

Nuclear Health and Safety Review

A review and compilation of current AEC requirements pertaining to nuclear health and safety at HAPO have been completed. Those AEC guides, regulations, and requirements which HAPO may anticipate in the future were noted also. A letter outlining the current and possible future AEC requirements has been drafted.

Fuel Pellet Fabrication for GE-APED

The local areas of high gamma activity in the sintered 1.5 w/o PuO_2-UO_2 Euratom pellets were analyzed in detail during the month. Four pellets showing high Am-241 (60 kv) gamma activity on the spectrometer were polished longitudinally. Auto x-ray and auto alpha plates were made of these sections. The analyses showed spots from 5-12 mils in diameter randomly distributed through the volume of the pellets.

Reruns of the pellets after very careful ultrasonic cleaning showed the spots to be caused by internal activity rather than surface contamination. Since the high exposure material used in this job had been chemically separated six months before its use, a significant amount of Am-241 had grown in before the sintering operation. It was remotely possible that the americium, which probably forms Am_2O_3, would not go into solid solution with the PuO_2-UO_2 during sintering. In order to check this possibility, the ratios of the Am-241 (60 kv) to U-237 (100 kv) activities at a large number of "hot" spots were compared to a number of "normal" areas. There was no significant difference between the two areas. This indicated that the spots were due to incomplete solid solution in the four-hour 1550 C sinter. From some recent British data and an assumption of a 70,000 cal/mol activation energy for solid solution formation it was calculated that it would be necessary to sinter 12 hours at this temperature in order to achieve complete solid solution. An experiment was run to check this. The four sectioned pellets were loaded into individual molybdenum boats each one of which could be individually withdrawn during the sintering cycle. Unfortunately, the sintering furnace burned out during the test. This is to be repeated when the furnace becomes available again.

[redacted] [redacted]

Fabrication of Fission Product Transient Capsules

Development work for the fabrication of 72 fission product transient capsules for Phillips Petroleum Company is continuing. Preliminary extrusions of the 0.8-inch OD, 0.060-inch wall coextruded aluminum tubes were made at a ten to one reduction in area starting from a 1.5-inch diameter hollow billet extruded over a floating mandrel. Problems with wall thickness and tooling are apparent. The first extrusions had wall thickness variations up to 0.010-inch and occasionally produced a bent mandrel. This was attributed to misalignment and non-uniform starting of the extrusions. To correct this misalignment problem, an extension was machined into the front of the billets to insure centering of the mandrel in the die. Also, through extended use, the liner in the extrusion container had assumed a barrel-like shape which would contribute to misalignment. A new liner is being fabricated.

Coextrusion billets of aluminum containing uranium-aluminum cores as stand-in for the aluminum-plutonium cores have been fabricated for continued development.

Critical Mass Fuels

Electroless nickel coated UO_2-polystyrene matrix solid fuels were thermal cycled from room temperature to 72 C repeatedly with no apparent rupturing of the coating. The cycled cube still contained all of the alpha radiation.

The open-faced hoods in Room 6-C of the 231-Z Building have been set up for plating the PuO_2-polystrene matrix fuel cubes.

Breeder-Type Flux Monitoring Foils

Initial pyrolytic carbon depositions were made by the decomposition of methane gas. The coatings were of non-uniform thickness and inconsistent between experimental runs.

Substitution of acetylene gas eliminated a majority of the problems. The tendency of the deposited carbon to break away from the base material during cooling is still a problem to be solved.

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PHYSICS AND INSTRUMENT RESEARCH AND DEVELOPMENT OPERATION

MONTHLY REPORT

DECEMBER 1961

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

N-Reactor Startup

A measurement of the diffusion length in the N-Reactor mockup exponential has been made with the process tubes inserted to provide a comparison to startup tests. The result is L = 54.6 cm. This result is to be compared with the diffusion length without process tubes of 83.0 cm. Both results are quoted for a graphite density of 1.70 gm/cm^3.

N-Reactor Exponential Experiments

The NPR mockup exponential pile has been modified to measure the effect of flooding from a ruptured process tube. Four stages of flooding will be mocked up as follows: 1) polyethylene wrapped around process tubes to simulate half of the annular gap filled with water, 2) process tubes replaced by thinner tubes of the same outside diameter to simulate all of the annular gap filled with water, 3) thin process tubes wrapped with polyethylene to simulate flooding of the gap and impregnation of nearby graphite with water (also to add confidence to interpolation between points), 4) masonite added to steam vents to simulate water in the steam vents. Measurements of buckling and change in relaxation length due to control rod insertion are planned for each stage. Wet enriched fuel is used for all measurements. The results obtained to date are listed in Table I. Measurements with thin process tubes have been corrected for the absence of part of the aluminum by using the PCTR flux measurements to determine Δf and Δk_∞. The corrections are 0.014 in f, 0.017 in k_∞, and 26 μB in B_m^2.

TABLE I

Flooding Condition	$B_m^2(\mu B)$	ΔB^2 (6 Rods)
None	116	-218
Half (polyethylene)	135	---
Full (thin tubes)	150	-241

The extrapolation distances used are 1.7 inches perpendicular to the fuel, and 1.4 inches parallel to the tubes. The results are not corrected for leakage from the exponential as would be required to estimate the total control strength directly. The value of ΔB^2 is subject to an absolute uncertainty of up to 10 μB due to the curvature in the fit of the data. The curvature results from using harmonic corrections for the pile without rods. The

comparison of one ΔB^2 to another is not so bad, since the same data points are used for both and the curvature should be the same. Unfortunately the error is not conservative--the quoted ΔB^2 overestimates the rod strength.

The rod strength increases with flooding and thus compensates for about two-thirds of the increase in buckling with flooding.

Optimization of C-Pile Retubed Lattice

An exponential pile has been built to mock up C-Reactor with either C-II-N fuel (standard I and E) or C-VI-N fuel (overbore I and E). Measurements of buckling and safety rod strength are in progress to support the design studies and hazards analysis of the overbore program. The buckling measured with wet C-II-N fuel is 82 μB and both measured extrapolation distances are 0.9 inches. Comparison to the dry measurement (114 μB) gives a wet-to-dry difference of 32 μB, which is much lower than the earlier measured differences of 53 μB for C and 40 μB for the old reactors. Consideration of the low C/U ratio in this pile (safety rod voids, slots around the tube block, etc.) could raise the new wet-to-dry difference to 38 μB which is still somewhat unreasonable. It is possible that the central safety rod void disturbed the vertical traverse which is taken close to the void. Work is in progress to eliminate the effect of this central void.

A special vertical traverse was taken to look for oscillations in the flux similar to the oscillations found earlier in horizontal traverses. No significant effect was found. This traverse is of interest to establish a limit to the magnitude of the oscillation since the small source theory method used to analyze horizontal traverse has not yet been successfully applied to vertical traverses.

Resonance Capture in Multisurface Fuel Elements

The lattice parameters have been measured for a concentric tube fuel in a 14.5-inch graphite lattice. The fuel assemblies are the same ones used for earlier measurements at different lattice spacings (HW-67904). The graphite used gave an over-all C/U atom ratio of 116.9. The results are recorded in Table I. Flux matching results were reported last month.

TABLE I

Coolant	k_∞	f	p	ϵ	η	R. I.
Water	0.924	0.768	0.877	1.055	1.300	16.4 barns
Air	1.034	0.858	0.870	1.064	1.302	13.0

The resonance integrals (derived from $-\ln p = N_U V_U$ R. I./$(\xi\Sigma_S)_M$) are larger than the values for the 10.5-inch lattice, but the resonance integrals implied by exponential data plots would be considerably smaller. This is a confirmation of earlier suspicions that p is not a simple exponential function for internally-moderated fuel.

The data available from resonance capture measurements (cadmium ratios, activity ratios for lattice and thermal spectra, activity ratios for resonance and 1/v detectors, etc.) were studied to see if information on the effective resonance integrals could be obtained without assuming the shape of the spectrum. This derivation would be analogous to lutetium studies where the thermal non-1/v behavior (the g-factor) is obtained from cadmium ratio and thermal ratio measurements. However, the resonance integral derivation fails because no epithermal calibration is available (for lutetium a known thermal calibration--a Maxwellian--is used). This result emphasizes the need for a standard facility of some kind with a known epithermal spectrum for resonance capture studies. The spectrum of the test lattice could be approximately determined by comparison with the standard--at least for energies important to the absorbers of interest. Resonance capture measurements in unknown spectra are not entirely useless, since the results are often applied directly to lattice parameters (especially p) without deriving the resonance integral directly. However, for faster spectra and higher leakage than are typical for Hanford reactors, the 4-factor formula becomes less useful and either more factors or a direct multigroup analysis would be needed. Spectral information is more important for the more complex systems.

A document (HW-72137) has been submitted describing an IBM program (PULP) for reduction of data in PCTR measurements of lattice parameters.

Green's Function Treatment of Exponential Piles

The analysis of the cadmium shutter method reported on last month has been extended to include a small source representation of the fuel region. As expected, the resulting expressions for the flux difference shows the same simplifications observed in the two-region case. Hence, small source effects appear as oscillations superimposed on the cadmium shutter difference of the two-region assembly. This completes the current series of theoretical studies of exponential assemblies, which has included studies of homogeneous assemblies, two-region assemblies, lattices, and the cadmium shutter method.

Comparison of Hurwitz Spectra with Experiment

A comparison of Hurwitz type spectra with experiments performed by the British is being undertaken with personnel from Reactor Lattice Physics. The British made measurements with a fast chopper in a graphite uranium lattice at various temperatures. The original data were fit by using a Maxwellian, an E^{-1} tail, and a connecting function. The connecting function is empirical, but appears to be suitable for a wide range of temperatures. While the British fits were quite good, the temperature of the Maxwellian was chosen in order to give a best fit and, hence, is not predetermined on the basis of moderator temperature. On the other hand, the Hurwitz spectrum consists of a Maxwellian whose temperature is that of the moderator and an E^{-1} tail with an associated theoretically determined joining function. Hence, the Hurwitz spectrum is easy to use in a practical situation. Preliminary comparison with experiments indicate some discrepancies, which are probably due to crystal binding, since the Hurwitz spectrum is derived for the heavy gas model.

Computational Programming Service

APDAC

An addendum(1) to the APDAC descriptive document(2) has been issued. The addendum describes several changes and improvements which have been incorporated in the production (SPL) version of APDAC. The changes have been designed to increase the program's flexibility, and decrease machine reruns due to input mistakes.

TRIP

TRIP003, a temporary version of the reactor kinetics code which permits the additional of as many as ten non-standard equations, is in operating condition. A more refined version, TRIP004, incorporating all the features of previous versions, is in debug now. The code has been extensively rearranged to provide neater, more descriptive output, simpler input, and greater speed through more logical division into subroutines.

New fast source corrections for cadmium shutter experiments are being incorporated into VTOCL, the exponential pile data reduction program. The new corrections are being debugged.

Lattice Parameter Calculational Methods

The 18-group calculation of the NPR lattice continued. The latest series of calculations constituted a test of one method of account for resonance self-shielding. The results of the test are inconclusive.

The original 9-ZOOM cross section data yielded a k_∞ which was 30 percent below the measured value. The U-238 capture cross section was then reduced in magnitude so that the excess resonance integral calculated from the revised 18-group constants and a 1/E flux in the fuel, agreed with that inferred from Hellstrand's experiments. The reduction in capture cross section for each group was assumed to be proportional to the "non 1/v" part of the capture cross section in that group. Using these reduced capture cross sections, the calculated k_∞ was 30 percent high. Next, the transport cross sections were corrected by assuming that the fractional decrease in effective scattering cross section affected only the difference between total and potential scattering, and was the same fraction of this difference as the required difference in the "non 1/v" part of the capture cross section. This correction yielded a k_∞ which was 20 percent high.

Since the relatively "dirty" geometry of the NPR cell introduces additional complications, it has been decided that, as a test of multi-group diffusion theory and a determination of suitable cross sections, a better approach will be to correlate first with earlier data on solid fuel lattices. This approach is now being tried.

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(1) Lilley, J. R., "AFDAC-I, A PCTR Data Analysis Code for the IBM 709. Input Data Changes." HW-63592 Add I, December 21, 1961.
(2) Lilley, J. R., "AFDAC-I, A PCTR Data Analysis Code for the IBM 709," HW-63592, September 7, 1960.

Instrumentation

A number of tests of various BF_3 tubes for use in the NPR graphite stack purity measurements showed that an N. Wood Counter Laboratory tube had the best drift, pulse-height distribution, and gain characteristics. The count rate of the best tube decreased one percent after 40 hours use under constant conditions. All BF_3 tubes tested exhibited a rapid counting-rate increase for the first few minutes following high voltage application. Seven of the N. Wood tubes were ordered for the reactor experiments; evaluation-exchange privileges were included in the order. Fabrication continues on the probe assemblies and the source holder to be used in the tests. Two preset counters were borrowed for the tests. Modifications were made on the developed transistorized charge-sensitive preamplifier to be used with the aforestated BF_3 tubes with tests indicating gain stability of ± 2% from 70°F to 110°F, gain linearity within 2% for an input charge change from 2×10^{-13} coulombs to 1×10^{-12} coulombs, and a signal-to-noise ratio of 5:1 at the 2×10^{-13} coulomb level. Fabrication of all items required for the tests is progressing satisfactorily.

It has been determined that about half the background counting rate of the recently developed uranium surface contamination monitor for NPR fuel is due to cosmic ray induced scintillations in the large lucite light pipes of the alpha detector. New hollow aluminum light pipes have been designed to eliminate this background. Fabrication is nearly completed.

New instrument specifications and cost estimates were provided IPD for a revised Appropriation Request for experimental fuel rupture detection equipment for the PRTR Fuel Element Test Facility. Total engineering, design, equipment, and installation costs are estimated to be about $90,000.

Assistance was provided Instrumentation Design, CEUO, regarding GE-APED questions about the scintillation probes for the slow-scan, gamma energy analysis portion of the NPR fuel failure monitor. The problem involved the resolution capabilities of the packaged integral NaI crystal and phototube assembly. Because of gain and stability characteristics, the RCA 6342-A was determined to be the better phototube for the probes although the guaranteed resolution (for Cs^{137}) is no better than about 9%. GE-APED will ship the prototype electronic portion of the slow-scan system to HAPO on 12-28-61. Nucleonic Instrumentation will test all system components.

Meetings were held December 11 and 12 in Los Angeles with engineers of the company fabricating the NPR Integrated Data and Temperature Monitoring System. Methods were presented by which they propose to meet HAPO specifications for the system. Evaluations of these methods in the light of NPR requirements were compiled and forwarded to NPR Porject Section along with analyses of thirty engineering drawings submitted since the Los Angeles conferences.

Drawings submitted by the manufacturer of the proposed NPR Zone Temperature Monitor were reviewed, and comments were transmitted to the NPR Project Section.

The Mark III of 12-inch HV process tube distortion traversing mechanism for the old reactors was fabricated and assembled. Both the Mark II (6-inch HV) and the Mark III were tested in the Optical Shop and in a 40-foot-long process tube at 189F Area. A vertical traverse using a mercury manometer was also run on the 189F tube. The mercury manometer data serves as the standard to which the HV traverse mechanism results are compared. A considerable amount of calculation is required to obtain a process tube contour from HV traverse data. A Fortran program has been prepared so that these calculations can be done on the IBM-7090. Results from the first computations are not yet available; however, an analysis of the data for the HV traverse mechanism by use of a desk calculator gave quite promising results. Both the 6-inch and the 12-inch HV traverse mechanisms are much easier to read than was the Mark I model.

Systems Studies

Work continued on the task of describing a reactor in control system terms. With a distributive reactor model on the analog computers, the flux-to-flux transfer function was measured for various distances and flux distributions. Preliminary analysis indicates the flux-to-flux transfer function (neglecting metal temperature effects) to be simple, non-integrating, and containing only real roots. However, the transfer function varies with the distance and flux distribution but does not vary with reactor power.

The measurement was made on the analog computers by solution of the reactor diffusion equation with a one-dimensional distributive reactor model. In the simulation, six equally spaced nodes were used with parameters for KW Reactor. The next step is to determine the reactivity-to-flux transfer function and include metal effects. By determining the basic flux-to-flux transfer function first, the effects of the metal and reactivity can be separated.

Rod position readout equipment was installed at the 105-KW reactor during the month in accordance with Production Test IP-439-C, "Rod Position Change Versus Neutron Flux". Tests made after installation indicate that the paralleled selsyn receivers have no readable effect on the existing rod position indicators. A series of recordings will be made, using normal operating rod movements and the recently installed in-core flux chambers, in an attempt to determine the reactivity-to-flux and flux-to-flux dynamics of the reactor under a variety of operating conditions. Information from this test is expected to complement the information from the tests with the analog computer simulation of the KW Reactor.

The six-node reactor speed-of-control model was used to study the old reactor. The effects of various rod insertion times and sticking rods were investigated. The main purpose was to determine the effects of assumptions made in previous speed-of-control studies.

The reactor instrumentation simulation was programmed on the EASE and GEDA computers. The simulation uses a slightly modified version of the six-node, top-to-bottom speed-of-control analog circuit. The reactor undergoes a nuclear excursion due to a reactivity ramp. It is desired to know, for various reactor conditions, the peak power level in a single reactor region

when the pile is scrammed by an ion chamber in another region. This study is being conducted to determine a suitable placement pattern for the ion chambers which activate the Beckman trips to scram the reactor.

Another investigation was designed to determine the effect of time delays introduced by the instrumentation on the reactor outlet temperature as a function of reactivity. All analog runs made previously were based on a "scram" tripped by a percentage of power level or by the outlet temperature. New runs have been requested with the "scram" tripped by the power level rate-of-rise (dn/dt) and by the period. These trips require added circuitry which has been designed. Some automatic ranging circuits to make the reactivity additions automatically have also been worked out. These circuits should make it possible to operate this simulation repetitively, thus reducing the computer time necessary by a considerable amount.

A more detailed estimate is being prepared on the computer equipment requirements to simulate the total NPR plant.

SEPARATIONS

Criticality Experiments with Plutonium Solutions

Further criticality studies were made of plutonium nitrate solutions in a fourteen-inch-diameter stainless steel sphere. The concentration of the plutonium in the solutions being studied ranged from ~35 g Pu/ℓ to ~238 g Pu/ℓ with acid molarities in the range of ~2 to 6.3. Types of reflectors used were a one-inch-thick paraffin reflector, and a four-inch-thick concrete reflector.

The initial experiments were begun for determining the effect of a concrete reflector on criticality. In these experiments a four-inch-thick spherical shell of concrete was positioned closely about the sphere. In subsequent experiments an additional layer of concrete will be used to increase the total thickness of concrete to ten inches.

Some efforts were directed toward evaluating the effect of the stainless steel shell of the critical assembly vessel on the critical mass, as well as to determine the correction for the sphere neck.

Several experiments were run to check the consistency of the results or the reproducibility of the criticality measurements. In three successive experiments in which the plutonium solution was drained from the vessel and then made critical, the measured critical volumes varied between 22.72 ℓ and 22.77 ℓ, indicating a maximum spread of ~0.05 ℓ or ~0.2 percent in volume.

The data from the criticality experiments completed in December are listed in the following table.

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

(Measured Sphere Volume 23.22 Liter; Wall Thickness, 0.044-inch)

Date	Exp. No.	Reflector Condition	Pu(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)
11-29-61	1141033	1" Paraffin + Neck Mockups	126.4	6.27	1.40	743	512	167.8	22.6	2.86
12- 4-61	1141034	1" Paraffin	79.8	3.55	1.241	851	306	293.7	18.3	1.46
12- 5-61	1141035	1" Paraffin	85.1	6.18	1.322	766	459	256.3	22.8	1.94
12- 8-61	1141036	1" Paraffin	238.0	5.21	1.517	745	501*	88.3	22.5	5.36
12-12-61	1141037	1" Paraffin	228.9	5.31	1.516	742	523*	91.6	22.6	5.17
12-13-61	1141038	0.036" S.S. + 1" Paraffin	233	5.55	1.514	738	-	89.8	22.7	5.29
12-14-61	1141039	0.036" S.S. + 1" Paraffin	233	5.55	1.514	738	-	89.8	22.7	5.29
12-15-61	1141040	0.036" S.S. + 1" Paraffin	230	5.62	1.515	735	-	90.7	22.8	5.24
12-19-61	1143041	4" Concrete Shell	57.9	3.82	1.194	868	-	413.7	15.5	.90
12-20-61	1143042	4" Concrete Shell	41.4	3.57	1.158	873	-	580.4	20.1	.83
12-21-61	1143043	4" Concrete Shell	~35	~2	-	-	-	-	22.6	
12-26-61	1143044	0.036" S.S. + 4" Concrete Shell	~35	~2	-	-	-	-	22.9	

* Subject to change

\- Final results of chemical analyses for plutonium solutions were not available at time of this report.

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The results obtained with the concrete reflector are preliminary (with the exception of the measured critical volumes) pending the completion of the chemical analyses of the solutions used in these experiments. The criticality data for the concrete reflected sphere indicate the four-inch-thick layer of concrete to be the equivalent of an infinite water reflector. This conclusion is predicated on the basis of a comparison between the critical mass for the concrete reflected sphere with the calculated and experimental results for the critical mass when an effectively infinite water reflector is used.

Data Correlation - Development of Nuclear Codes - Criticality Calculations and Nuclear Safety

1. Criticality of Plutonium Nitrate Systems

The calculations of the critical parameters of plutonium nitrate systems reported in part last month have now been completed. The final calculations give the critical parameters for plutonium-nitrate systems containing plutonium (30 to 700 gm/ℓ), nitric acid (two to six molar) and Pu^{240} (2.5 and 6.5 atom percent). The values for the parameters are given for bare and H_2O-reflected sphere in Tables I-IV.

The values for bare and reflected spheres at 5%-Pu^{240} are interpolated from the directly calculated data and are given in Tables V and VI.

TABLE I

CRITICAL PARAMETERS FOR $Pu(NO_3)_4 \cdot H_2O + HNO_3 \cdot H_2O$ BARE SPHERES - 2.5% Pu^{240}

H/Pu	Pu(g/ℓ)	H^+(Mol)	Spec.Grav.	H_2O(g/ℓ)	NO_3(g/ℓ)	Rc(cm)	Vc(ℓ)	Mc(Kg Pu^{239})
857	30	2.00	1.112	926.5	153.0	21.72	42.92	1.197
512	50	2.00	1.146	921.4	173.0	19.40	30.58	1.491
363	70	2.00	1.182	916.7	193.0	18.57	26.82	1.831
281	90	2.00	1.216	911.4	212.8	18.15	25.04	2.198
165	150	2.00	1.316	891.8	271.9	17.65	23.03	3.367
95.0	250	2.00	1.477	855.7	369.9	17.44	22.22	5.416
55.1	400	2.00	1.712	794.7	516.1	17.33	21.80	8.500
32.3	600	2.00	2.009	698.8	708.8	17.42	22.14	12.954
25.8	700	2.00	2.160	650.9	807.8	17.47	22.33	15.243
812	30	4.00	1.167	859.4	273.2	22.45	47.39	1.386
486	50	4.00	1.203	856.5	292.8	20.09	33.96	1.655
344	70	4.00	1.235	850.1	311.4	19.25	29.88	2.040
266	90	4.00	1.268	844.1	330.0	18.80	27.83	2.442
156	150	4.00	1.367	827.2	386.2	18.23	25.38	3.711
89.4	250	4.00	1.520	789.0	477.0	18.02	24.51	5.974
51.5	400	4.00	1.743	727.2	612.1	17.96	24.27	9.463
30.0	600	4.00	2.028	635.7	789.3	18.07	24.71	14.458
23.8	700	4.00	2.172	589.1	880.4	18.14	25.00	17.064
771	30	6.00	1.227	796.6	394.3	23.18	52.17	1.526
456	50	6.00	1.250	785.4	409.3	20.93	38.41	1.872
323	70	6.00	1.283	780.5	427.2	20.02	33.61	2.294
250	90	6.00	1.317	776.3	445.4	19.55	31.30	2.746
146	150	6.00	1.410	757.5	496.9	18.98	28.64	4.189
83.8	250	6.00	1.559	722.4	581.2	18.70	27.39	6.677
48.0	400	6.00	1.772	662.5	704.4	18.61	27.00	10.529
27.7	600	6.00	2.044	575.5	865.0	18.75	27.61	16.153
21.9	700	6.00	2.182	531.1	947.6	18.85	28.06	19.492

TABLE II

CRITICAL PARAMETERS FOR $Pu(NO_3)_4 \cdot H_2O + HNO_3 \cdot H_2O$ REFLECTED SPHERES - 2.5% Pu^{240}

H/Pu	Pu(g/ℓ)	H^+(Mol)	Spec.Grav.	H_2O(g/ℓ)	NO_3(g/ℓ)	Rc(cm)	Vc(ℓ)	Mc(Kg Pu^{239})
857	30	2.00	1.112	926.5	153.0	17.65	23.05	0.674
512	50	2.00	1.146	921.4	173.0	15.29	14.97	0.730
363	70	2.00	1.182	916.7	193.0	14.41	12.53	0.855
281	90	2.00	1.216	911.4	212.8	13.99	11.47	1.006
165	150	2.00	1.316	891.8	271.9	13.40	10.08	1.474
95.0	250	2.00	1.477	855.7	369.9	13.15	9.52	2.321
55.1	400	2.00	1.712	794.7	516.1	13.07	9.35	3.647
32.3	600	2.00	2.009	698.8	708.8	13.17	9.57	5.597
25.8	700	2.00	2.160	650.9	807.8	13.23	9.70	6.620
812	30	4.00	1.167	859.4	273.2	18.27	25.54	0.747
486	50	4.00	1.203	856.5	292.8	15.85	16.68	0.813
344	70	4.00	1.235	850.1	311.4	14.98	14.08	0.967
266	90	4.00	1.268	844.1	330.0	14.52	12.82	1.125
156	150	4.00	1.367	827.2	386.2	13.89	11.22	1.642
89.4	250	4.00	1.520	789.0	477.0	13.64	10.63	2.590
51.5	400	4.00	1.743	727.2	612.1	13.55	10.42	4.064
30.0	600	4.00	2.028	635.7	789.3	13.64	10.63	6.218
23.8	700	4.00	2.172	589.1	880.4	13.70	10.77	7.351
771	30	6.00	1.227	796.6	394.3	18.83	27.97	0.818
456	50	6.00	1.250	785.4	409.3	16.50	18.82	0.917
323	70	6.00	1.283	780.5	427.2	15.57	15.81	1.079
250	90	6.00	1.317	776.3	445.4	15.07	14.34	1.258
146	150	6.00	1.410	757.5	496.9	14.43	12.59	1.841
83.8	250	6.00	1.559	722.4	581.2	14.11	11.77	2.868
48.0	400	6.00	1.772	662.5	704.4	14.02	11.54	4.501
27.7	600	6.00	2.044	575.5	865.0	14.12	11.79	6.898
21.9	700	6.00	2.182	531.1	947.6	14.19	11.97	8.168

TABLE III

CRITICAL PARAMETERS FOR $Pu(NO_3)_4 \cdot H_2O + HNO_3 \cdot H_2O$ REFLECTED SPHERES - 6.5% Pu-240

H/Pu	Pu(g/ℓ)	H^+(Mol)	Spec.Grav.	H_2O(g/ℓ)	NO_3(g/ℓ)	Rc(cm)	Vc(ℓ)	Mc(Kg Pu^{239})
894	30	2.00	1.112	926.5	153.0	18.50	26.52	0.744
533	50	2.00	1.146	921.4	173.0	15.99	17.12	0.800
379	70	2.00	1.182	916.7	193.0	15.08	14.36	0.940
293	90	2.00	1.216	911.4	212.8	14.65	13.17	1.108
172	150	2.00	1.316	891.8	271.9	14.11	11.77	1.650
99.0	250	2.00	1.477	855.7	369.9	13.93	11.32	2.646
57.5	400	2.00	1.712	794.7	516.1	13.90	11.25	4.207
48.4	600	2.00	2.009	698.8	708.8	14.00	11.49	6.448
38.6	700	2.00	2.160	650.9	807.8	14.05	11.62	7.604
847	30	4.00	1.167	859.4	273.2	19.12	29.28	0.821
506	50	4.00	1.203	856.5	292.8	16.57	19.06	0.891
359	70	4.00	1.235	850.1	311.4	15.67	16.12	1.055
277	90	4.00	1.268	844.1	330.0	15.20	14.71	1.238
163	150	4.00	1.367	827.2	386.2	14.61	13.06	1.832
93.2	250	4.00	1.520	789.0	477.0	14.42	12.56	2.936
53.7	400	4.00	1.743	727.2	612.1	14.39	12.48	4.668
31.2	600	4.00	2.028	635.7	789.3	14.50	12.77	7.164
24.8	700	4.00	2.172	589.1	880.4	14.59	13.01	8.514
804	30	6.00	1.227	796.6	394.3	19.75	32.27	0.905
476	50	6.00	1.250	785.4	409.3	17.26	21.54	1.007
337	70	6.00	1.283	780.5	427.2	16.30	18.14	1.187
261	90	6.00	1.317	776.3	445.4	15.81	16.55	1.393
153	150	6.00	1.410	757.5	496.9	15.22	14.77	2.071
87.4	250	6.00	1.559	722.4	581.2	14.97	14.05	3.285
50.0	400	6.00	1.772	662.5	704.4	14.91	13.88	5.193
28.9	600	6.00	2.044	575.5	865.0	15.02	14.19	7.962
22.8	700	6.00	2.182	531.1	947.6	15.10	14.42	9.439

TABLE IV

CRITICAL PARAMETERS FOR $Pu(NO_3)_4 \cdot H_2O + HNO_3 \cdot H_2O$ BARE SPHERES - 6.5% Pu^{240}

H/Pu	Pu(g/ℓ)	H^+(Mol)	Spec. Grav.	H_2O(g/ℓ)	NO_3(g/ℓ)	Rc(cm)	Vc(ℓ)	Mc(Kg Pu^{239}
894	30	2.00	1.112	926.5	153.0	22.60	48.35	1.356
533	50	2.00	1.146	921.4	173.0	20.17	34.37	1.607
379	70	2.00	1.182	916.7	193.0	19.28	30.02	1.964
293	90	2.00	1.216	911.4	212.8	18.90	28.28	2.379
172	150	2.00	1.316	891.8	271.9	18.47	26.39	3.702
99.0	250	2.00	1.477	855.7	369.9	18.32	25.75	6.020
57.5	400	2.00	1.712	794.7	516.1	18.26	25.50	9.538
48.4	600	2.00	2.009	698.8	708.8	18.37	25.97	14.567
38.6	700	2.00	2.160	650.9	807.8	18.40	26.09	17.079
847	30	4.00	1.167	859.4	273.2	23.39	53.60	1.503
506	50	4.00	1.203	856.5	292.8	20.87	38.08	1.780
359	70	4.00	1.235	850.1	311.4	20.02	33.61	2.200
277	90	4.00	1.268	844.1	330.0	19.60	31.54	2.653
163	150	4.00	1.367	827.2	386.2	19.08	29.09	4.080
93.2	250	4.00	1.520	789.0	477.0	18.95	28.50	6.663
53.7	400	4.00	1.743	727.2	612.1	18.97	28.59	10.694
31.2	600	4.00	2.028	635.7	789.3	19.05	28.96	16.246
24.8	700	4.00	2.172	589.1	880.4	19.10	29.19	19.103
804	30	6.00	1.227	796.6	394.3	24.15	59.00	1.655
476	50	6.00	1.250	785.4	409.3	21.81	43.46	2.032
337	70	6.00	1.283	780.5	427.2	20.83	38.76	2.537
261	90	6.00	1.317	776.3	445.4	20.39	35.51	2.988
153	150	6.00	1.410	757.5	496.9	19.87	32.86	4.609
87.4	250	6.00	1.559	722.4	581.2	19.64	31.73	7.417
50.0	400	6.00	1.772	662.5	704.4	19.65	31.78	11.887
28.9	600	6.00	2.044	575.5	865.0	19.76	32.32	18.130
22.8	700	6.00	2.182	531.1	947.6	19.84	32.71	21.410

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TABLE V

CRITICAL PARAMETERS FOR $Pu(NO_3)_4 \cdot H_2O + HNO_3 \cdot H_2O$ BARE SPHERES - 5% Pu^{240}

H/Pu	Pu(g/ℓ)	H^+(Mol.)	Spec.Grav.	H_2O(g/ℓ)	NO_3(g/ℓ)	Rc(cm)	Vc(ℓ)	Mc(Kg Pu^{2}
880	30	2.00	1.112	926.5	153.0	22.26	46.20	1.317
525	50	2.00	1.146	921.4	173.0	19.72	32.12	1.526
373	70	2.00	1.182	916.7	193.0	19.03	28.87	1.920
288	90	2.00	1.216	911.4	212.8	18.63	27.08	2.316
169	150	2.00	1.316	891.8	271.9	18.20	25.25	3.599
97.5	250	2.00	1.477	855.7	369.9	18.06	24.67	5.860
56.6	400	2.00	1.712	794.7	516.1	17.90	24.02	9.129
33.1	600	2.00	2.009	698.8	708.8	18.40	24.59	14.017
26.4	700	2.00	2.160	650.9	807.8	18.06	24.67	16.408
833	30	4.00	1.167	859.4	273.2	23.04	51.25	1.460
498	50	4.00	1.203	856.5	292.8	20.58	36.51	1.734
353	70	4.00	1.235	850.1	311.4	19.73	32.17	2.139
273	90	4.00	1.268	844.1	330.0	19.34	30.30	2.591
160	150	4.00	1.367	827.2	386.2	18.82	27.92	3.979
91.7	250	4.00	1.520	789.0	477.0	18.70	27.39	6.506
52.8	400	4.00	1.743	727.2	612.1	18.60	26.95	10.243
30.7	600	4.00	2.028	635.7	789.3	18.71	27.43	15.6
24.4	700	4.00	2.172	589.1	880.4	18.78	27.74	18.45
791	30	6.00	1.227	796.6	394.3	23.79	56.40	1.607
468	50	6.00	1.250	785.4	409.3	21.44	41.28	1.961
332	70	6.00	1.283	780.5	427.2	20.56	36.40	2.421
257	90	6.00	1.317	776.3	445.4	20.08	33.91	2.899
150	150	6.00	1.410	757.5	496.9	19.54	31.25	4.454
86.0	250	6.00	1.559	722.4	581.2	19.35	30.35	7.208
49.2	400	6.00	1.772	662.5	704.4	19.28	30.02	11.408
28.5	600	6.00	2.044	575.5	865.0	19.40	30.58	17.432
22.5	700	6.00	2.182	531.1	947.6	19.49	31.01	20.623

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TABLE VI

CRITICAL PARAMETERS FOR $Pu(NO_3)_4 \cdot H_2O + HNO_3 \cdot H_2O$ REFLECTED SPHERES - 5% Pu^{240}

H/Pu	Pu(g/ℓ)	H^+(Mol.)	Spec.Grav.	H_2O(g/ℓ)	NO_3(g/ℓ)	Rc(cm)	Vc(ℓ)	Mc(Kg Pu^{239})
880	30	2.00	1.112	926.5	153.0	18.13	24.96	0.711
525	50	2.00	1.146	921.4	173.0	15.71	16.24	0.771
373	70	2.00	1.182	916.7	193.0	14.73	13.39	0.890
288	90	2.00	1.216	911.4	212.8	14.36	12.40	1.060
169	150	2.00	1.316	891.8	271.9	13.79	10.98	1.566
97.5	250	2.00	1.477	855.7	369.9	13.72	10.82	2.569
56.6	400	2.00	1.712	794.7	516.1	13.63	10.61	4.030
33.1	600	2.00	2.009	698.8	708.8	13.74	10.86	6.193
26.4	700	2.00	2.160	650.9	807.8	13.80	11.01	7.321
833	30	4.00	1.167	859.4	273.2	18.68	27.30	0.778
498	50	4.00	1.203	856.5	292.8	16.32	18.21	0.864
353	70	4.00	1.235	850.1	311.4	15.42	15.36	1.021
273	90	4.00	1.268	844.1	330.0	14.96	14.02	1.199
160	150	4.00	1.367	827.2	386.2	14.34	12.35	1.760
91.7	250	4.00	1.520	789.0	477.0	14.18	11.94	2.837
52.8	400	4.00	1.743	727.2	612.1	14.08	11.69	4.443
30.7	600	4.00	2.028	635.7	789.3	14.18	11.94	6.808
24.4	700	4.00	2.172	589.1	880.4	14.30	12.25	8.145
791	30	6.00	1.227	796.6	394.3	19.42	30.68	0.874
468	50	6.00	1.250	785.4	409.3	16.89	20.18	0.958
332	70	6.00	1.283	780.5	427.2	16.02	17.22	1.145
257	90	6.00	1.317	776.3	445.4	15.56	15.78	1.349
150	150	6.00	1.410	757.5	496.9	14.98	14.08	2.006
86.0	250	6.00	1.559	722.4	581.2	14.73	13.39	3.180
49.2	400	6.00	1.772	662.5	704.4	14.63	13.12	4.984
28.5	600	6.00	2.044	575.5	865.0	14.70	13.31	7.585
22.5	700	6.00	2.182	531.1	947.6	14.79	13.55	9.012

2. Quantity of Boron-10 Required to Reduce k_∞ to Unity for U^{235}-H_2O Mixtures

The amount of Boron-10 to be added to a hypothetical U^{235}-H_2O solution to reduce k_∞ to unity and thus render the solution safe from criticality under all conditions of geometry and reflection has been calculated. The results are given in the table below.

The calculational procedure consisted of computing the thermal group constants with the "Tempest" code and using these results with the multigroup 9-Zoom code (18 energy groups).

CRITICAL PARAMETERS FOR INFINITE U-235-H_2O - B^{10} SYSTEMS

					Thermal Group Constants			
H/U^{235}	U-Conc (g/ℓ)	N^U (x10^{24}) atoms/cc	B-Conc (g/ℓ)	$N^B/N^{U^{235}}$	σ_f^{235}	σ_c^{235}	σ_c^B	$\sigma_a^{H_2O}$
11.5	2000	.005123	16.6	.1952	165	30	1349	0.22
26.7	1000	.002562	6.31	.1483	205	38	1600	0.26
30.8	800	.002049	4.81	.1415	220	40	1670	0.28
42	600	.001537	3.40	.1334	236	44	1810	0.30
63.6	400	.001025	2.24	.1317	287	52	2010	0.35
128.8	200	5.123 (-4)	1.178	.1386	344	61	2386	0.40
259	100	2.562 (-4)	.589	.1386	378	67	2857	0.45
520	50	1.287 (-4)	.246	.1155	424	74	3040	0.51
867	30	7.685 (-5)	.140	.1100	450	78.5	3205	0.53
1301	20	5.123 (-5)	.0581	.0683	463	80.5	3294	0.55

Assumed densities of pure materials.

U^{235} - 0.04785 x 10^{24} atoms/cc
B-10 - 0.1473 x 10^{24} atoms/cc
H_2O - 0.03337 x 10^{24} atoms/cc

3. Plutonium Cross Section Data for the Monte Carlo Code

In preparation for studying aqueous-plutonium systems by means of the HISMC Monte Carlo code, a new data tape has been generated which contains in addition to the data on the previous tape, the cross section data for Pu^{239} and Pu^{240}. Resonance integrals which have been calculated for these two isotopes of plutonium from the data stored on the data tape are summarized in the table below.

RESONANCE INTEGRALS CALCULATED FROM HISMC DATA TAPE

Isotope	Type	Cutoff Energy (ev)	$\int \sigma(E)\frac{dE}{E}$ (barns)
Pu^{239}	Fission	0.5	319
Pu^{239}	Capture	0.5	220
Pu^{240}	Capture	1.0	8533

The fission integral for Pu^{239} is in good agreement with the value quoted by Hardy of 327 ± 22 barns.(1) The capture integrals for Pu^{239} and Pu^{240} are in good agreement with the values calculated from resonance parameters given in BNL-325, including unresolved level contributions, of 214 barns and 8450 barns, respectively.

4. Effect of Reactor Exposure on Fuel Element Reactivity

Theoretical calculations made in 1956(2),(3) have shown that the reactivity of slightly enriched uranium fuel elements, irradiated in the Hanford reactors, initially increases and then decreases as a function of irradiation time. Estimates of the maximum reactivities that could be expected for a particular initial fuel element enrichment were made in 1957,(4) and these estimates have since been used as a basis for nuclear safety in processing irradiated fuel elements in the Separations Plants. To further these estimates, the effective maximum reactivities of several Hanford fuel elements have been calculated using the IBM-7090 Idiot code. In each case, pile exposure was taken up to 1000 MWD/T using the plutonium conversion tables(5), and then extrapolated to 1300 MWD/T. The effect of fission product buildup was neglected. The results of the calculations are given in the table below.

Fuel Element Model	Initial Enrichment w/o U-235	Maximum Equivalent Enrichment After Irradiation (up to 1300 MWD/T), w/o U-235
C-IV-N	.712	.773
K-Solid	.712	.769
K-IV-N	.712	.760
O-Solid	.712	.758
C-Solid	.712	.757
C-V-N	.712	.754
C-IV-N	.712	.752
O-III-N	.712	.749
K-IV-E	.946	.966
C-III-E	.946	.958
O-III-E	.946	.955
C-IV-E	.946	.950

(1) J. Hardy, Jr., et al., The Resonance Fission Integrals for U^{235}, Pu^{239}, and Pu^{241}, Nuclear Sci. & Eng. 9, 341-345 (March 1961).

(2) Triplett, J. R., Reactivity Changes & Isotope Yields at High Exposures, HW-33912, January 20, 1955.

(3) Heineman, R. E., Reactivity Dynamics of a Thermal Reactor, HW-35786, March 15, 1955.

(4) Ketzlach, N., Effects of Pile Exposure on Reactivity of Slightly Enriched Fuel Elements, HW-48400, February 11, 1957.

(5) Hoba, D. W., Plutonium Conversion Tables, HW-50300 B, June 30, 1961.

The maximum effective reactivities estimated in the previous study(6) were 0.79% U^{235} for natural uranium fuel elements and 0.98% U^{235} for slightly enriched fuel elements.

5. Calculated Critical Bucklings and Critical Masses for 0.75 and 0.80% U^{235} Fuel Rods

Critical bucklings and critical masses have been calculated for 0.75 and 0.80% U^{235} enriched uranium rods in light water. These values are to be added to the data for 1-5 percent U^{235} rods presented in HW-69273.(7) From this extension of data, more accurate critical parameters can be estimated for fuel rod lattices in the enrichment range of 0.7-1.0% U^{235}. The new maximum bucklings and minimum critical masses are as follows:

Enrichment, % U^{235}	Maximum B^2, 10^{-6}, cm^{-2}	Minimum Critical Mass Tons of U
0.75	478	76.6
0.80	1130	19.0

6. Interaction of Sub-Critical Systems

The interaction problem is being treated by decomposing the neutron currents at a boundary into bare, reflected and excess production components. The outward and inward currents in two groups are written in terms of diffusion theory albedos and leakage, resulting in four simultaneous equations. Solution of these current equations yield the proper group weighting functions for the albedo, leakage, and secondary absorption of the system. Experimental data for plane and cylindrical reflected systems are used for comparison. The method will be extended to interacting units if these calculations prove reliable.

Mass Spectrometry

The heavy element mass spectrometer for this program was inoperative during the month because of equipment failure. This equipment failure was determined to be due to electrical short circuits in the coils which provide the additional magnetic field to sweep the ion beam through a range of several mass units. A new set of coils was fabricated which was designed to provide greater strength against the forces due to the interacting magnetic fields. The magnet wires of the new coils are vacuum impregnated with "Scotch cast resin" and contained in machined lucite forms. The new coils have now been installed on the mass spectrometer, and the spectromer is again in satisfactory operating condition.

(6) Ketzlach, N., Effects of Pile Exposure on Reactivity of Slightly Enriched Fuel Elements, HW-48400, February 11, 1957.

(7) Brown, C. L., Calculated Critical Parameters for Slightly Enriched Uranium Rods in Light Water, HW-69273, April 1961.

Criticality Hazards Specifications

1. Nuclear Safety in HLO

Three specifications were issued this month:

A-2: Rules for the Storage and Transporting of Al 1.8 w/o Pu Alloy, 0.5 inch Diameter Fuel Rods - Reactor Lattice Physics, 305-B Building.

F-3: General Rules for the Storage, Handling, and Transporting of 93% U^{235} Enriched Uranium Metal - Fuel Element Design and Fuels Fabrication Development, 306 Building.

L-1: Rules for the Storage and Handling of 0.95% U^{235} Enriched NPR Fuel Elements - Mechanical Equipment Development, 314 Building.

2. Nuclear Safety in FPD

The shipping regulations for 0.95% U^{235} enriched uranium were reviewed for the Engineering Operation. As a result of this review, the limits for shipping I&E cores were revised and a new specification was established for shipping recycle uranium from the NPR fuel element fabrication process (333 Building).(1) The revised limit for I&E cores permits 14,480 lb. of 0.95% U^{235} enriched per railroad car, compared to the past limit of 5,300 lb., and provides for a further two-fold increase if natural uranium I&E cores are included in the shipment. The new specification for NPR uranium permits 8,500 lb. to 30,000 lb. of enriched uranium per railroad car, depending on the form and packaging. The bases for these changes are given in HW-71800 DR-1(2).

Instrumentation

A "C" Column Simulation is being studied in an attempt to develop a satisfactory mathematical model of a pulse column used in chemical separations. Analog programs for four possible models of varying complexity have been prepared and the analog computer will be used to determine the chemical constants necessary to give the best fit of each of the models to given experimental data. Only one model has been used in the simulation to date. The results have been very useful, particularly in determining the relative effects of variations in the various unknown constants.

The source of some of the line voltage transients at the Critical Mass Lab has been located. The false trips occurring as a result of them have been reduced. Some of the trouble resulted from a compressor starting at the Hot Semi Works. Other troubles seemed to have been caused by the Critical Mass Lab ventilation cooler compressor. This cooler compressor is now turned off while critical experiments are being run. Slightly low line voltage (105 volts) has caused a malfunction of a supposedly regulated power supply in one scanner. This scanner has been inoperative for several weeks for other reasons. Poor instruction books in conjunction with high

(1) Nuclear Safety Specifications - Fuel Element Manufacturing Processes, HW-47013, Sections NS 3.00 and NS 6.00.

(2) Brown, C. L., Nuclear Safety Specification Basis for the Off-Site Shipment of 0.95% U^{235} Enriched Uranium, HW-71800 DR-1, November 22, 1961.

prices and long delivery of plug spare parts are largely responsible. Increased maintenance in recent weeks has prompted the start of a routine maintenance program. Many of the problems seemed to result from weak electronic tubes. Most of these tubes have operated for about a year without replacement. This routine maintenance program will probably require an additional part time instrument craftsman.

In connection with the Gorton lathe problem, a mathematical study was made of a two-frequency dynamic vibration absorber. The method now appears practical and simple. A final decision on the practicality cannot be made until completion of a detailed computer study of the absorber equations together with equations representing the Gorton lathe.

NEUTRON CROSS SECTION PROGRAM

Slow Neutron Cross Sections

Measurements were made of the relative reflectivities of several different planes of a germanium crystal monochromator on the 105-DR spectrometer. Open beam spectral measurements were also made in order to determine the importance of competition and double diffraction in these reflections.

Inelastic Neutron Scattering from Water

A series of measurements has been started to determine the correction necessary for half-order contamination reflections in the Be/0002/ reflections used in the measurements of inelastic neutron scattering from water.

Fast Neutron Cross Sections

Samples of lithium, sodium, and potassium were successfully cast in stainless steel jackets for use as samples in total cross section measurements.

A first set of measurements was made on the program to measure neutron total cross sections in the energy region of 3 to 15 Mev by pulsed-beam time-of-flight techniques using the Van de Graaff. The measurements were made using the spectrum of neutrons from the $Li^7(d,n)Be^8$ reaction. The time resolution which was attained was about 0.43 ns/m at a flight path of 6.1 meters. This is equivalent to an energy resolution width of about 100 kev at 4 Mev neutron energy. Total cross section measurements were made on samples of Li, Na, K, Al, Fe, and Cu in a total running time of 48 hours. The statistical precision of the data over the range of observation was better than about three percent. The analysis of the data to obtain total cross section values is in progress.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Pu Values in Fast Spectrum Reactors

The plutonium "indifferent values" for a series of Pu compositions have been calculated for a large, sodium-cooled, fast oxide breeder (power level: 1000 MWth, core column: 2300 liters). The indifference values are based

on equating fuel costs in Pu and uranium fueled fast reactors. It was postulated that fuel cost differences arise only from differences in breeding ratio, fuel burnup costs and inventory charges. The plutonium compositions considered were assumed to be derivable from a typical pressurized water, thermal reactor at various stages of burnup. The fissile content of this plutonium varied from 64 percent to 78 percent.

Two methods were used in determining the Pu value. In the first method it was assumed that Pu-240 and Pu-242 were valueless, and that the thermally fissile isotopes Pu-239 and Pu-241 were equivalued. On this basis it was found that the per unit value of the fissile plutonium is considerably higher for the more irradiated plutonium composites.

In the second method, Pu-240 and Pu-242 were again assumed to be valueless, but separate values were assigned to the Pu-239 and Pu-241 isotopes. It was found that for the ranges of plutonium compositions examined, reasonably constant, separate values could indeed be assigned to the Pu-239 and Pu-241 isotopes. These values can at best only be approximately constant because Pu value changes with reactor environment. It was found that Pu-239 was about 60 percent more valuable than U-235, and Pu-241 was found to be more than twice as valuable as U-235 in the fast spectrum region.

The results of this brief investigation re-emphasize the usefulness of Pu in fast spectra and, therefore, suggest again the desirability of a fueling strategy involving a fast-thermal reactor complex. In such a complex, highly exposed Pu from a thermal reactor would be used to fuel the core of a fast reactor, and the relatively clean Pu-239 from the fast reactor blanket would be used to enrich the fuel of the thermal reactor.

The Critical Facility

The G. E. Safeguards Council has recommended that the reactivity which is available for addition into the PRP-CF from the control console be minimized. There are two ways in which reactivity can be added from the console. These are control and safety rod withdrawal and moderator level increases.

Most of the experiments which are planned for the facility can be accomplished with an amount of excess reactivity which is 3/4 (75 cents) of that which is required to take the reactor from delayed to prompt critical. Therefore the reactor will usually be loaded so that the excess reactivity would be 75 cents with the moderator level of interest and with all of the safety and control rods in their "out" position. The maximum height to which the moderator can be raised will be limited by mechanical stops for the weir. This prevents the moderator from being raised above a given level which is set while in the reactor cell. However, the mechanical stops do not interfere with adjusting the weir to levels lower than the maximum from the console.

In order to arrive at the desired excess reactivity, partial clusters or individual rods of fissionable material will be used. The partial clusters would be the normal fuel elements but with some of the rods missing from the cluster. The individual rods would be added by using special

hangars. The individual rods could be loaded in fuel element positions or in one-inch holes which are located in the reactor templates.

An alternate method is to introduce anti-reactivity in the form of boron, copper, or cadmium. A convenient method of doing this would be to use a control rod whose position is adjusted from the reactor cell. Its position could be locked for a particular part of an experiment. This would allow easy adjustment from the cell and still not allow more than 75 cents in reactivity to be added from the console.

Extensive measurements of the ratio β/l will be made in the PRP-CF by the ncise analysis technique. As a result time has been spent in becoming familiar with this technique and the instruments required to use it.

The work pertaining to photoneutron effects on reactivity measurements in the PRP-Critical Facility has continued. All of the input parameters corresponding to a Critical Facility fuel loading of 25 UO_2 and 30 Pu-Al elements with D_2O moderator have been assembled for use in the machine calculations for this loading. A first case which uses 34 photo- and delayed-neutron groups is now being keypunched. The IBM-7090 code, RE-126, for reactivity step calculations will be used. This code assumes all delayed neutron precursors in equilibrium at the time of the reactivity step increase. A second code for reactivity step calculations, RE-135, which allows cases with nonequilibrium precursors to be handled has been obtained from Mr. C. E. Cohn of the Argonne National Laboratory. This code will be modified for use in the photoneutron problem.

A broad scope survey of the multiplying system characteristics for a wide range of loading configurations has been completed for D_2O moderated, 19-rod clusters with L_x Pu or natural UO_2 fuel. This information was intended to satisfy the immediate needs of preliminary hazards evaluation and planning of the initial experimental program. The survey results and their interpretation will be reported with attention given to the conditions and effects outlined below. All calculations were performed using SWAP, a one-dimensional, three-energy group diffusion theory computer code, programmed in FORTRAN language for the IBM-7090.

The reactor models were specified in cylindrical geometry with axial dependence appearing in buckling terms. Material composition of each region is given in three-group values of nuclear parameters, where the bulk of the core is represented by flux- and volume-weighted cell-average values. Each region of the center cell of the core is represented in equivalent annular geometry and complete material detail to facilitate the calculation of perturbation coefficients of parametric variations expected or forced to occur.

The models treated and effects determined were the following:

I. Loadings at full moderator level were two-zone ($0 \longrightarrow 32$ UO_2 elements surrounded by $11 \longrightarrow 42$ Pu-Al elements) with all loadings limited to a total of 61 elements. Some loadings did not completely fill the 61 sites.

a. Reactivity worth of UO_2-Pu-Al element replacement and element deletion at center cell position and zone boundaries by exact solution for each model.

b. Radial distribution of fuel substitution worth by complex statistical weight distribution.

c. Estimation of change in multiplication due to fuel temperature - changed resonance cross section (perturbation calculation).

d. Reactivity worth of density changes in moderator and interstitial D_2O (perturbation calculation).

e. Effect of reactor cell H_2O flooding accident on all configurations.

f. Multiplying system effective values of prompt lifetime, age, diffusion length, and buckling.

g. Eigenvalue and normalized flux distribution.

II. Loading evaluation at reduced moderator levels was carried out for near critical, all Pu-Al loadings and full lattice two-zone loadings.

a. Results include all those listed in I.

b. Moderator level coefficient in terms of element substitution and Δk_{eff}.

PRTR Critical Tests Summary and Analysis

The effect of voiding coolant channels in the PRTR was determined experimentally during critical testing and the results have been recently summarized. These will appear in HW-61900 BA, "PRTR Critical Test Results." Theoretical estimates of the effects and the methods used to obtain them have been written for inclusion in that report. Treated are the calculations of a traverse of coolant void effect in the three-zone configuration used during critical testing, and the effect of loss of coolant in all channels.

The results of cell traverses which were done with lutetium foils have been compiled and will be reported in the same document.

Code Development

RBU

In an attempt to locate the error in the Monte Carlo causing excessive particle leakage, the fequency function of the distance to collision calculation was constructed, using the random logarithm generator. The function appeared to be satisfactory as far as distribution is concerned; no

information indicating the randomness of the generator was obtained, however. It is now felt that the trouble lies elsewhere.

The collision and geometry routines are once again undergoing logic and programming checks.

The change in the random number and random cosine generators (see November report) produced a small but detectable effect in the Monte Carlo execution time. The main trouble was not, however, eliminated by this revision.

Spectrum and Cross Section Codes

An informal document, HW-71953, describing the Spectrum V program has been issued. Two reports in the process of being reproduced are HW-72092 describing the UPDOC program which updates the 19-level ANP data tape, and HW-72117 describing program BARNS which computes average cross sections from the RBU basic library. Rough drafts of reports describing the C-6 program and the C-6 data tape program have been completed.

PRTR Fuel Element Irradiation Experiments

The first fuel elements to be irradiated in the PRTR primarily for obtaining physics data on fuel burnup were loaded into the reactor during the month. These were three of six modified low exposure Pu-Al Mark I elements; the remaining three will be charged during the next shutdown. The elements are identical to the standard Mark I elements, except that neutron flux monitors have been attached. These monitors consist of a Co-Al wire running the full length of the element and small Co-Al pins interspersed radially through the cluster at the mid-plane. Four $1\frac{1}{2}$" cadmium sleeves are placed along the wire to measure cadmium ratios. The elements will be removed individually from the reactor at approximately equal exposure intervals leading to 50 percent burnup ($\sim$84 MWD) of the initial fissile atoms in the last element to be removed. The data from these and other planned irradiation experiments will be used for checking fuel cycle analysis methods.

A standard Mark I L_x Pu-Al fuel element with a faulty end spider was recently discharged from the PRTR. The element has been irradiated to about 30 percent burnup ($\sim$60 percent of goal exposure), but because of its defective end spider, it cannot be used again in the reactor. It will be sent to Radiometallurgy for obtaining fuel samples at various locations on the element. The samples will cover a range of burnup of approximately 20 percent to 40 percent. They will be analyzed for plutonium and Cs-137 concentrations and plutonium isotopic composition. The experiment will be useful in checking out the sampling techniques used in Radiometallurgy, and exposure and isotopic analysis method that will be used on future irradiation experiments. It will also yield burnup data on the L_x plutonium over the exposure range indicated at a much earlier date than the planned physics tests described above. The data will not be as comprehensive, however, since the physics tests will include in addition to the above data flux measurements, gamma activity as a function of position on the element, and reactivity measurements in the PRCF.

Fuel Cycle Analysis Work

Physics development work on improving nuclear analysis methods for fuel cycles progressed through formulation and machine programming of a parallel-logic transmission routine designed to permit fuel cycle calculations to proceed in asynchronous simultaneity with multi-channel iterative tape storage and core loading. This compact (13-card) FORTRAN subroutine is expected to provide a substantial reduction of fuel-cycle analysis cost, by essentially eliminating tape transmission time from consideration.

The innermost loop of the improved fuel-cycle analysis has been put to active use somewhat sooner than had been anticipated. In response to a request from the Reactor Analysis Operation, the matrix diagonalization loop was provided with the input and output processing logic needed for immediate application to the isotope production and decay aspects of fast breeder analysis work. This skeleton version of the improved fuel cycle analysis has been given to the Reactor Analysis Operation as an operating machine program. Performance of the deck, in this application, proved to be satisfactory on the first run. Early release of the heart of the analysis for limited use, in this manner, provides opportunity for accrual of an unusually high level of checked-out reliability as analysis improvement work proceeds.

Theory development work on the non-linear aspects of nuclear analysis has led to discerning that the non-linear coupling between the neutron balance equation and the isotope balance equation has mathematical structure precisely analogous to that encountered in the geodesic equations for a curved Riemannian space. Exploitation of this mathematical analogy has begun, with indications that it may be possible to construct a few analytic solutions for idealized fuel cycles by use of the Hamilton-Jacobi transformation method of constructing geodesics.

Calculation of the Lattice Parameters for Pu-Al Fuels

In conjunction with the experimental studies of Pu-Al fueled systems, an analytical study of the 19 rod PRTR-type cluster was started. A physical model of the cell was established and a study of the neutron spectrum, prior to obtaining cross sections, is in progress.

Plutonium Fuel Temperature Coefficient

The PCTR experiments which were designed to measure the fuel temperature coefficients of Pu-Al fuels have been completed. The PCTR core cavity was loaded with a 3 x 3 array of 19 rod clusters of $\frac{1}{2}$-inch diameter Pu-Al rods, with the clusters arranged in a square $6\frac{1}{2}$-inch graphite lattice. The core loading was poisoned to unit multiplication with copper. The central test cell was replaced with special apparatus which permitted the fuel in the test cell to be heated. The change of the PCTR reactivity was measured as the fuel in the test cell was heated from room temperature to ~450 C. The measurements were made with a constant buffer region loading and with six different test cell loadings. The preliminary results are listed in Table I.

TABLE I

Test Cell Loading	Initial Slope of Reactivity Change	
Al Dummy fuel, unpoisoned	+ 8	($\times 10^{-4}$ ¢/° C)
Al Dummy fuel, poisoned	~ + 8	"
1.8 w/o LX Pu-Al, unpoisoned	- 4	"
1.8 w/o LX Pu-Al, poisoned	~ 0	"
2.1 w/o HX Pu-Al, unpoisoned	- 11	"
2.1 w/o HX Pu-Al, poisoned	- 6	"

Pu-240 Effective Resonance Integral

The experiment to measure the effective resonance integral of Pu-240 relative to the dilute resonance integral as a function of Pu-240 concentration is in progress. The sensitivity of the reactor to neutron absorption in Pu-240 is somewhat lower than would be desirable. It may be necessary to modify the loading in the PCTR somewhat in order to increase the sensitivity.

There has been no difficulty with Pu contamination to date.

Neutron Spectra

Measured thermal neutron spectra have recently been published by Coates and Grayther (AERE-R-3829). Their analysis of these spectra employ the approximation of effective neutron temperatures and the heavy gas model to yield effective masses as a function of neutron and moderator temperatures. Their results are in fair agreement with those derived from Hanford rethermalization measurements, reported in HW-68389.

Analyses were started in cooperation with C. W. Lindemeier on the British spectra using the "Hurwitz" spectrum, HW-70716, p. 4. A preliminary fit of their measured spectrum from 20 C graphite with a Hurwitz spectrum characterized by a moderator temperature of 20 C is not as good as was expected. Tentatively this has been attributed to the assumption of a constant value for $\xi\Sigma_s$ used in the derivation of the Hurwitz spectrum. The analysis of their spectra from graphite at 160, 244, and 321 C are being completed.

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

Experiments to determine nuclear parameters for 1.8 w/o Pu-Al fuel are being conducted in the tank in the TTR reactor room. The experiments will consist of approach to critical and exponential measurements with lattice spacing of 0.75, 0.85, 0.90, and 1.05 inches.

Fabrication of 4 sets of the templates has been completed. The set with a 0.75-inch lattice spacing has been assembled in the tank and the associated controls and instruments assembled and checked out.

Fabrication of the fuel rods has been completed and 493 rods were delivered with approximately 20 more to be delivered.

The measurements will help establish nuclear safety specification for PRTR fuel processing and will supplement similar measurements which are part of the PRP-CF program.

Activation Resonance Integral of Cu

The value of the activation resonance integral for dilute natural copper, 3.13b, given in HW-63576, p. 30, is in error by approximately 15 to 25%. A major portion of the error is due to a systematic error in the measurement of the cadmium ratio of the 1/v standard, a BF_3 chamber. If this error is corrected, a new value of 3.77 ± 0.39b is obtained. This value is for the energy interval 0.64 ev to ∞ and is comparable to the results of Dahlberg, et al, (JNE 14 53, 1961) and Sher (BNC - private communication) that yield 4.04 ± 0.49b.

The new Hanford value was inferred from the original data in HW-63576 and the recent results of Jacks (DP608) for the cadmium ratio of gold as a function of the foil thickness.

A ratio of partial resonance integrals for Cu has been calculated as a function of temperature for either a Westcott or Hurwitz spectrum. This ratio of the epithermal to epicadmium resonance integral is used in correcting bare activities to obtain thermal activities.

The detailed results of this work will appear in the October-December 1961 Quarterly Report.

Instrumentation and Systems Studies

Basic design requirements were completed on the shield-collimator system to be used for PRTR fuel rod, and later Co^{60} wire activation, studies. The electronic instrumentation requirements have been determined and can be met with equipment presently on hand. The shield-collimator assembly has provision for insertion of bored cylinders of lead to provide various collimation hole diameters as required for the large range of gamma intensity levels to be observed, and provisions were incorporated for simple detector assembly change and for a simple, visual alignment procedure. The completed basic design will be used by draftsmen to prepare detailed drawings for fabrication use.

A demonstration was set up to show the feasibility of one method of readout for a resonant microwave cavity. The cavity is to act as a length transducer for in-reactor measurements, e.g., creep. In this demonstration, a sweeping oscillator slowly increased frequency until the cavity was detected. The time interval between the start of the sweep and the detection of the resonance was measured with an electronic timer, and is indicative of the cavity length. The present cavity has a total travel of nine millimeters. The resolution is better than 10 microinches, but the stability of the sweep will not allow this sort of precision. A servo follower system, using a second cavity, will be built, and should provide a better test.

PRTR kinetics data were obtained on magnetic tape recordings made in conjunction with PRTR Test #12 during the last outage. Three ionization chambers of different sizes were installed to obtain information on the effects of

varying chamber parameters and to provide redundant information for cross-checking purposes. Two channels of information were recorded simultaneously during each of five recording periods. Recordings were made under the following conditions:

1. Pre-startup - background noise
2. Subcritical (5 kilowatts)
3. Critical (65 KW at 89-inch moderator level)
4. Critical (45 KW at 104-inch moderator level)
5. Critical (45 KW at 104-inch moderator level - different chambers).

Extreme difficulty was experienced in attempts to hold moderator level (power level) constant during the test. The moderator level appears to fluctuate somewhat randomly with sufficient magnitude to cause power level changes which overload the recording equipment at normal gain settings. The amplitude of the moderator fluctuations was greatest at the 89-inch level and the recording made at this level is probably useless. Shim rods were then inserted until operation at 104 inches could be attained and two more recordings were made, using all three chambers at least once each. Although the moderator level fluctuations were still large, it is believed that useful information can be obtained from these data by making use of the relatively large number of conditions recorded. These data are presently being processed in the laboratory. It appears that useful information could be obtained by recording the frequency spectrum of the moderator level variations. An analysis of the moderator contributions to the flux frequency response curve would improve the accuracy of the neutron lifetime measurement and would yield information which might indicate the source of the moderator fluctuations. A method of obtaining a continuous recording of moderator level is being investigated.

HTLTR Studies

The study of ways to minimize the temperature coefficient of the proposed High Temperature Lattice Test Reactor (HTLTR) has continued. Several models have been studied. A calculation of the fluxes and reactivity in a model with a "just-critical" core surrounded by graphite moderator, a fuel driver ring, and a graphite reflector, indicates that the temperature coefficient of the reactor is due to 1) a positive term due to an increase in thermal utilization, f, and 2) a larger negative term due to an increase in total leakage (fast plus thermal) giving a net negative coefficient. The most recent model studied consisted of the same system but with a U^{235} blanket added around the periphery of the reactor. The blanket was designed to utilize the increased leakage neutrons to increase f, thus reducing the net coefficient. The calculated reactivities indicate that the desired result was not obtained, the temperature coefficient remaining essentially unchanged.

PHOENIX FUEL

The Pu-Al sample which contains 27.2 w/o Pu^{240} has completed the second irradiation cycle in the MTR. The sample could not be removed from the container which positioned it in the MTR. In order to remove the sample the container was to be sawed in two pieces. During the sawing process, the cladding on the sample was penetrated but not the Pu-Al alloy. No particular contamination problem was encountered. However, the ARMF measure-

ments and future irradiations cannot be made with this sample. Therefore, a spare sample which contains 27.2 w/o Pu^{240} has been sent to the MTR to replace it.

NEUTRON FLUX MONITORS

Experimentation concerning actual foil fabrication by Plutonium Metallurgy personnel was carried out with pyrolytic carbon depositions made by the decomposition of methane; however, the initial coatings were of non-uniform thickness and inconsistent. Substitution of acetylene gas eliminated the stated problems; however, there is still a tendency of the deposited carbon to flake from the base material during coating. The flaking of the solid solution of the Pu nuclides in a carbon matrix is a problem yet to be resolved before the foils can be used for experimentation. A cost estimate was obtained from Analytical Laboratories, HLO, for ampoule preparation and chemical purification of Pu for mass spectrograph analysis of the foils. The ampoule-foil packages will be used for the direct in-core irradiation tests scheduled. Considerable effort was expended in revising and rewriting the interim report concerning the progress to date for the various Pu and U nuclide composition extended lifetime in-core flux monitors.

The properties of various gases and solids which might possibly be useful in a microwave neutron flux monitor at high temperature are being theoretically considered and compared. Some additional equipment has also been ordered to increase the sensitivity of the present experimental microwave system.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

Measurements show that increased instrument stability is required as the number of parameters to be identified by the developmental multi-parameter eddy current nondestructive test device is increased. As a result of these measurements, the operation of the equipment is being rechecked in detail, and modifications are being made to improve its stability. It also appears that the range of variation over which parameters can be separated decreases as the number of parameters is increased. Both of these effects are consistent with the theory; however, their degree depends upon characteristics of the equipment and the test coil-test specimen system which can be determined best by over-all measurements.

An experimental calibration procedure for adjusting the constants of the transformation circuit was developed and proved out for the two-parameter case. The procedure is general, and it is applicable for cases having more than two parameters, but the complexity of adjustments increases rapidly with the number of parameters.

A new compensating or balance circuit was built which will be used with the multiparameter equipment in a proposed fuel element cladding thickness measurement application.

A four-parameter summing network was fabricated for use with the orthogonal exponential expander and sampling circuits which are being developed concurrently with the Fourier series equipment completion. The new system has

been designed so that the number of signal descriptors measured can exceed the number of parameters of interest. Means to perform the required transformation of descriptors to parameter readout in such a case are under investigation. This has led to a study of the manipulation of rectangular matrices.

Heat Transfer Testing

The radiometer previously used during heat transfer testing was designed for use with an induction heater, rather than the plasma arc jet now being used as a heat source. A smaller, more compact radiometer suitable for use with the plasma arc has been installed. This unit should improve performance of the heat transfer testing system, since it is more rigid, and an enclosing cover has been provided to shield the chopper and other optical system components from stray heat and radiation. A thinner flame control shutter than previously used has been installed to allow closer spacing between the plasma arc nozzle and fuel element under test. An automatic clutch which engages the scanning unit simultaneously with the opening of the flame control shutter has also been installed. This allows the fuel element to arrive at full rotational velocity before start of the test, thereby eliminating unsymmetrical heat application during the first few seconds of the test.

Zirconium Hydride Detection

Seven hydrided Zircaloy-2 samples having hydrogen contents from less than 30 to 500 ppm (by weight) have been obtained for use in the ultrasonic hydride detection studies. Metallographic analysis of the 500 ppm hydrided Zircaloy sample previously used in these studies showed that it actually contained less than 50 ppm hydrogen (confirming the results of earlier X-ray and vacuum fusion analysis).

Further experimentation with the X-ray method showed that much of the preferred orientation problem observed last month was due to a defective filter which permitted the K-beta-1 copper line generated by the X-ray tube to reach the Zircaloy-2 sample. This line was reflected from the zirconium(101) planes and interfered with the gamma zirconium hydride (111) peaks. Installation of a new filter removed the copper K-beta-1 line. Gamma zirconium hydride (111) peak heights for Zircaloy-2 samples containing 400, 700, and 1000 ppm were then found to increase with increasing concentration. However, one sample containing 980 ppm hydrogen showed a much lower than expected peak. This sample will be checked metallographically.

In an effort to detect local hydriding in Zircaloy-2, eddy current testing techniques are being applied to measure the Hall coefficient and resistivity of a metal, simultaneously. Preliminary tests to demonstrate the feasibility of such a measurement were performed using a 6.4 x 1.6 x 0.2 cm bismuth sample, a material with a relatively high Hall coefficient.

Placing the sample in a one-kilogauss magnetic field, and using a test frequency of 100 kc, an output voltage, tentatively identified as Hall effect voltage, has been observed. The ratio of the output to input voltages (measured at the test coil terminals) was about 4×10^{-6} when originally observed. Circuit refinements have sinceresulted in a definite improvement in this ratio, which is now about 2×10^{-4}. Actual voltage measured at the

terminals of the pickup coil is now about 1 to 2 mv RMS. An improvement of at least 10^4 in sensitivity will be required before this method will be applicable to Zircaloy-2.

Small Crack Detection

A preliminary study was made to determine the feasibility of applying low frequency (5-30 cps) eddy current tests to sections of carbon steel piping. Based on penetration depth calculations, it appears extremely doubtful that cracks any appreciable distance below the pipe surface can be detected without magnetically saturating the sample to reduce permeability effects. Even then, magnetic saturation only raises the upper frequency limit to about 200 cps for a penetration depth of 0.250 inch. These calculations will be verified experimentally as soon as the proper equipment can be located and assembled.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Transducer response measurements continued. The techniques listed in ASTM Specification E-127-58-T for assuring proper ultrasonic system operation, leading to transducer response evaluations, were found to be inadequate following further evaluations. The method which was tentatively adopted for measuring transducer response with smooth ball bearings includes tests for near field, far field, and lobe configurations using a very small (0.039-inch-diameter) ball bearing. The response of nine transducers were measured using this method. A variety of flat and focused lithium sulphate units were included. The use of very small ball bearings permits the maximum detail in response data. The data were not corrected for system nonlinearity, the measurement of which continues to be a problem. Criteria for acceptable transducer performance have not yet been quantitatively determined, though intuitive judgment indicates undesirable qualities in some of the measured responses.

Effort to improve the ultrasound beam image produced with the preliminary Schlieren system continued pending delivery of improved optical components. Several white and monochromatic light sources were tried. However, significant improvements in image contrast and brightness were not observed. The dark field background continues to exhibit a mottled appearance which, because of its brightness, limits the sensitivity for observing the portion of light diffracted by the ultrasound. The use of a plate glass ultrasonic water cell in lieu of ordinary glass improved the image contrast slightly. Further improvements may be possible by eliminating optical defects which may be scattering light into the image background.

Two 1/16-inch-diameter lithium-sulphate transducers were procured for beam sampling tests which will be compared to similar data obtained with ball bearings.

By tuning the transducer and signal cable, a gain increase of 20 to 30 times the normal gain was achieved with the Immerscope. The increased gain permitted detection of small pits and impressed materials which

previously were not detected. The angle of entry of ultrasound for this improved response was very nearly the 20 degrees previously used. At this angle good results were obtained on two tubing sizes at frequencies of five and ten megacycles.

Further studies were made to determine the significance of the 20-degree entry angle which appears to be empirically prevalent for all thin wall Zircaloy tubing tests. A review of theory revealed a significant point. Thin specimens vibrate entirely in the direction parallel to the surface for Lamb wave modes with a phase velocity equal to the longitudinal mode velocity. The angle of entry for a Lamb wave with phase velocities equal to the longitudinal velocity in Zircaloy is about twenty degrees. For these causes then, the ultrasound will propagate further, with minimum attenuation, and thus intercept defects with more initial energy. This allows the reflected signals from defects to be sufficiently separated (in time) from the entry point of ultrasound where a confusion of noise signals exists. This appears to be the empirically favorable testing condition.

Selection of the optimum mode or modes for a particular application remains to be investigated. In order to examine the comparative properties of different modes, the ultrasonic system frequency bandwidths must be measured to determine the capabilities for selecting a single mode. The relative merits of narrow and broad banding must also be evaluated. Techniques for measuring bandwidths are being develcped.

Modification of the plate manipulator is complete. Excitation of particular Lamb modes in 0.0385-inch thick plates was attempted. The ability to select and identify particular modes for study also depends on system bandwidths. The control of ultrasound entry angle is also necessary and initial experiments indicate focused transducers may be preferable. Though the particular modes were not identifiable, the energy which appears to experience minimum attenuation was generated at about the same 20° angle as in tubing.

An AR600 Sperry Reflectoscope has been leased for evaluation.

PHYSICAL RESEARCH - 05 PROGRAM

Mechanism of Graphite Damage

Apparatus is being set up that will measure electron or hole mobilities in graphite.

A technique was devised for calibrating digital voltmeters (or voltage-to-frequency converters) against standard cells that allows for the internal impedance of each. The full accuracy of the standard cell is used.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Progress was made in the measurement of dry deposition of zinc sulfide tracer material on natural surfaces. In a stable atmosphere, zinc sulfide particles were dispersed upwind of a flat area covered with dry cheat grass standing approximately 30 cm. high. Samples of grass were taken over a measured area

both before and after the particles were deposited. Mass determinations were made on ashed samples using the liquid scintillation counter. The cheat grass was found to phosphoresce when ashed; however, this phosphorescence decayed much more rapidly than that for zinc sulfide, so that the zinc sulfide content could be measured after a twenty-second decay time.

Determinations of the mass of zinc sulfide particles deposited per unit area of ground surface were made at distances of 10 meters, 50 meters, and 100 meters from the source. Samples obtained at 200 meters' distance were below the detection limits of this technique. Deposition as a function of distance was determined by the product of isoplethed areas of deposition and area. Twenty-two percent of the released mass was found to be deposited within 100 meters from the source. This result supports the earlier estimates obtained from mass continuity determinations from air samples reported in HW-69292-REV. Deposition velocities computed from the ratio of the deposition per unit area to the integrated exposure in air were between 0.5 and 1.0 cm/sec, averaging slightly smaller than the values reported in the document.

In our precipitation scavenging work, samples were collected during a snowfall, utilizing the raindrop sampler. In addition, measurements of snowflake spectra were attempted with the University of Michigan raindrop spectrometer; however, the instrument is not yet calibrated for snow. Significant quantities of zinc sulfide tracer particles were collected by the falling snowflakes, but the scavenging efficiency cannot be determined because of equipment failure. The atmosphere diffusion measurements made simultaneously during the snow indicated that the maximum exposure at each distance was reduced substantially by the scavenging process. Several additional samples of drop and flake size during light rain and snow were obtained during the month.

The atmospheric diffusion field trial conducted in conjunction with the snow scavenging experiment included deposition measurements on snow covered filters. Several difficulties with the technique were encountered both in the field and in assaying the samples. Even so, the deposition filters averaged about 10 percent of the count on exposure samples at 1.5 meters height and a definite distribution corresponding to the 1.5 meter arcwise distribution of exposure was obtained.

Dosimetry

The I-131 counter at the Whole Body Counter was calibrated. About 5 pc of I-131 is the smallest amount that can be detected reliably in the thyroid. A survey of human subjects showed I-131 in all the milk drinkers but none in the non-milk drinkers. Thyroid levels were about 25 to 46 pc. Typical milk samples showed 40 to 50 pc per quart. Thyroid levels decreased during the month and are not detectable now. The I-131 probably came from fallout from recent Russian tests. Its disappearance is probably due to radioactive decay and to removal of cows from pastures to feed lots where their food is not exposed to fallout.

Another heavy gas, C Br_2 F_2, was tested for use in proportional counters. It would not count.

The positive ion accelerator operated satisfactorily during the month. The sparking stopped; we don't know why. The new alignment system for the Van de Graaff works so well that a long counter can now be used as a monitor for time-of-flight work; before, deuterons striking beam stops, etc., gave too high a background. A new beam-current integrator based on an accurate voltage-to-frequency converter has proved simple and useful; experiments requiring a particular integrated current can be controlled by using a scaler to count the converter output and to shut off the equipment at a selected total count.

Reproducibility tests were completed of BF_3 tubes and of cores and thermal shields of precision long counters. BF_3 tube reproducibility was about 0.3% for tubes that had not previously been exposed to strong gamma ray sources. Core reproducibility was about 0.3%. Thermal shield reproducibility was about 0.1%. Deadtime measurements made on some of the BF_3 tubes gave values of 4.5 to 5.0 microseconds.

Measurements made at Mound Laboratory confirm our estimate of the rate of increase of neutron emission of one of our neutron sources. They compared our source with one of theirs whose rate of increase of activity had been measured calorimetrically.

The Li I spectrometers are now light and vacuum tight and are being tested.

Preliminary work was completed on the measurement of neutron fluxes at near-background levels.

A mechanical stage was fabricated that enables good reproducibility to be obtained in reading large tissue-equivalent ionization chambers with the vibrating reed electrometer.

The calorimetric measurement of the half-life of Sb-124 is continuing.

The heat-amplifier calorimeter was set up again and an attempt made to determine the operating voltage for maximum amplification. Breakdown of equipment prevented completion of the experiment.

Radiation Instruments

Direct support was received from Radiation Protection Operation for fabrication of a second generation prototype coincident-count alpha air monitor. A number of features found desirable from tests with the present laboratory prototype will be incorporated in the new instrument which will be used for field demonstrations. The present unit continues to perform correctly. Design and layout details have been completed for the second unit. It should be noted that the coincident-count method used in this system can likely be directly applied to quick analysis counting of the plant standard 4 x 8 inch and 4 x 4 inch air filters which are used in great numbers. At present, these filters can be counted, with reliability, only after a several hour delay to permit decay of the accumulated radon-thoron products. Using the coincident method, the filters could be counted for Pu alphas immediately after removal from their

holders. Such a procedure might materially reduce the number of times that personnel protective breathing devices are prescribed when there are really only changes in radon-thoron background levels. A simple equation has been determined to provide graphic information relative to the actual Pu alphas on such a fixed filter counted in this manner.

In an effort to improve the low energy gamma response and the angular response of the thermoluminescent dosimeters, a new unit was fabricated from a thin steel wire (screen) matrix with the CaF_2:Mn phosphor bonded to the wire with the usual heat-stable silicon resin. The assembly (0.5 inch cube) is sealed in boroscilicate glass. Performance was similar to that for the previous "button" fabrication technique concerning response versus accumulated dose for high energy gammas. The results at lower gamma energies showed a marked energy dependent response; however, the characteristics of LiF_2, as opposed to the presently-used CaF_2:Mn, are such to indicate that this problem would not exist with LiF_2 and the matrix method does produce nearly a 4π geometry which is required for many proposed applications of the dosimeters. Experiments will commence, using LiF_2, as soon as the proper material can be obtained.

Detailed drawings and specifications were completed for the modified pencil dosimeter portion of the aural-alarm pocket-dosimeter with pre-selectable accumulated dose alarm levels. This detector uses the illuminated quartz fiber, positioned slit-shield, and CdS cell readout. The drawings and specifications will be sent off-site to obtain a bid for fabrication of six to eight of the units. These will be used, with on-site fabricated transistor circuitry, to prepare a number of demonstration field model units. A similar method will be attempted for securing a number of the modified pencil dosimeters, using a quartz rod and specially-mounted fiber, used in the pocket registering dosimeter. In addition, experimentation was started on this approach to provide selectable alarming in addition to the registering feature.

Experimental progress was made on the field-type continuous airborne luminescent particle monitor to be used by Atmospheric Physics for studies of wind transport. A sampling head was designed, fabricated, and tested. The airborne ZnS particles are drawn into the large chamber and exposed to ultraviolet light for about 20 seconds and are then transported through tubing to be viewed by a multiplier phototube for about 0.1 seconds. The resultant anode current from the phototube drives a dc microammeter which drives a chart recorder. Laboratory and out-of-building dispersal tests, using ZnS, were performed with additional filter samples being obtained simultaneously. Chart results, filter analyses, and calculations showed that the same system could be effectively used by Atmospheric Physics at the ARC-4 distance (3200 meters) from the dispersal tower for continuous monitoring. Efforts will be continued to improve the system sensitivity; however, field tests will be tried shortly with the laboratory instruments to quickly attempt determination of usefulness. It has been determined that dry ZnS provided the best system signal to noise ratio.

Considerable progress was made concerning the development of various types of transistor charge-sensitive, voltage, and current amplifiers. Using selected low-noise transistors, an input charge of 1×10^{-13} coulombs per pulse produced an output 30 millivolt pulse into a 50 ohm load and a 5:1

signal to noise ratio was obtained for an input charge per pulse of 5×10^{-14} coulombs. Using special Crystalonic Co. C-613 field effect transistors, an equivalent input noise level of about 1800 electrons was obtained. Selected 2N1279 transistors provide input equivalent noise levels of 1000 to 3000 electrons.

Discussions were held with Atmospheric Physics personnel regarding development and procurement of new instrumentation for use with the best sensors, wind direction sensors, and anemometers used on the APO portable mast. Reliable scanning operation with complete data recording is required. A basic system development plan is being formulated.

The wind speed and direction testing unit designed for the Atmospheric Physics radiotelemetry system was slightly modified, tested successfully, and returned to general use. Temperature tests of several data stations indicated a number of temperature sensitive transistors which were replaced by newer types. Investigations continued concerning the data station interrogation and identification system ("Vibrasponder" resonant reed relays) to determine operation difficulties. The tests indicate faulty adjustment may be the cause. Another chronic problem is the complete discharge of the data station 6-volt power battery when the data station fails to turn off at the end of call-in. No simple solution to this problem has been found as yet. Most of the chronic problems with the radiotelemetry system, except the one just stated, have now been solved. Now concentrated maintenance attention is needed to update and adjust the system. Most of the necessary conversion and modification work has been engineered and assigned to the maintenance personnel for incorporation into the system.

Experiments started on some transistor circuitry development to convert information from a particle size (> 0.1 micron) and flow detection system to pulse information suitable for direct use with a multichannel analyzer. In essence, the circuit should provide pulses linearly porportional to the observed particle size as viewed by the commercial head.

Development continued on the pulse height analyzer for the positive ion accelerator. It will have a 400 channel capacity but will also operate in split modes of 100 or 200 channel sections. Memory transfer of one section to another will be possible. Provisions for background subtraction are being incorporated as well as live time measurement and overflow count storage. The basic design of the analyzer has been completed and more detailed development is now under way. Although the analyzer is being designed basically as a pulse height analyzer, it will also be compatible for use with the Vernier Chronotron.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

The mass spectrometer for this program was used to provide analyses on program samples as received during the month.

During the month a spectrum of impurity ions was found to be causing a background at the mass 235 and 237 positions. Measurements made at lower masses showed peaks at mass 203 and 205 which were identified as elemental

thallium from the observed isotopic abundance. The impurity peaks at mass 235 and 237 positions are then presumably due to TlO_2. The thallium apparently originates as an impurity in the filament assembly and not from the loaded sample. This problem is currently under further study.

Vacuum testing has been started on the components of the vacuum ion-optical test bench.

Studies of the focusing properties of an ion energy-filter lens were resumed. The optics of an energy-filter lens design for electrons reported in the literature is being studied by electrostatic field plotting and tracing of ion trajectories. These studies are preliminary to the design of lens systems to be experimentally studied with the ion-optical test bench.

TEST REACTOR OPERATIONS

Operation of the PCTR continued routinely during the month. There were two unscheduled shutdowns due to electronic failure.

The Doppler coefficient measurement for Pu-Al fuel in a $6\frac{1}{2}$-inch graphite lattice was completed and the experiment to determine the effective resonance integral for Pu^{240} was started during the month.

The new physicist's console was put in service. The console holds instruments used by the experimenters but not necessary for actual reactor control.

Operation of the TTR continued during the month on an intermittent basis. There were no unscheduled shutdowns. The TTR facility was made available two nights during the month to the University of Washington Graduate Center.

A Pu-Be source was installed in the TTR during the month.

The gantry on the critical approach tank was revised to eliminate a potential safety hazard.

CUSTOMER WORK

Weather Forecasting and Meteorological Service

Consultation service was rendered on meteorological and climatological aspects of NPR safeguards analyses. Rainfall intensity-duration-frequency curves for the Hanford site were prepared for CE&UO for application to a design problem. Meteorological services, viz., weather forecasts, observations, and climatological services were provided to plant operations and management personnel on a routine basis.

Type of Forecast	Number Made	% Reliability
8-Hour Production	93	80.8
24-Hour General	62	82.8
Special	186	91.9

[redacted]

[redacted] 1249992

Average temperature and total precipitation were both near normal during December. However, there was much more sunshine and much less fog than usual. High wind occurred on several days.

Instrumentation and Systems Studies

The transistorized beta-gamma electronic circuit chassis for the Automatic Laundry Monitor was fabricated and tested. The corresponding alpha circuitry chassis was nearly completed and both sets of scintillation detector probes are also nearly done with all completed probes tested correctly.

Circuit design assistance was rendered to Chemical Effluents Technology regarding the adaptation of a Newlett-Packard 561-B printer to a CET project. The adaptation involved the design of transistor circuits (inverters) to permit acceptance of positive polarity pulses by the printer in addition to power supply circuit changes. Fabrication and installation will be handled by CET personnel.

A general design and cost estimate was requested by Radiation Protection Operation concerning the conversion of the present Columbia River Monitor to a pulse counting system. The required system sensitivity is 0-50 micro r/hr full scale for gamma radiation with incorporated high and low level dose-rate alarms, power failure alarm, and automatic circuitry for energizing sample taking devices. The design and estimate were started.

Rough draft functional specifications were prepared for a subcritical monitor for the 305 Pile at the request of FPD.

The tape perforator and programmer for the automatic readout of NPR fuel element dimensions is now installed and operating satisfactorily in the 306 Building.

Calibration of micro-displacement readout systems, to be used by Physical Metallurgy Operation for in-reactor creep measurements, has continued during December. The reference system has been transported to the Metrology Lab in 200-W Area for a normal periodic calibration. This was necessary both as a routine procedure and because additional measuring instruments have been added since the reference system was last calibrated. The invar, micrometer support system has been fabricated and installed and the Boeckler micrometer and Cleveland gage head appear to function very well.

The NPR heat dissipation instrumentation drawings being supplied by Bailey Meter Company are being reviewed for NPR Project Operation. Two memoranda were issued during the month.

Optics

During the four-week period (December 3 to December 31) a total of 416 man-hours' shop work was done. This work included:

1. Fabrication of two silica cylinders for General Engineering Laboratory.
2. Fabrication of 80 alundum specimens for compression testing for the Reactor Modification Design Operation, IPD.
3. Repair of two lenses for Radiomet Lab, HLO.

4. Modification of a borescope objective section for traverse work for Testing Operation, IPD.
5. Fabrication of a quartz window for the hot stage metallograph of Physical Metallurgy, HLO.
6. Testing the flatness of 40 braze bell jar windows for Facilities Engineering, FPD.
7. Modification of one telescope for CPD.
8. Preparation of a new vacuum pump for installation in the shop vacuum system.
9. Fabrication and assembly of the 12-inch HV traverse mechanism for Materials Testing, IPD.
10. Fabrication of components for the reference system for calibration of LVD transducers.
11. Fabrication of two lenses to be used in an arc image furnace by Ceramic Fuels Development Operation.
12. Repair of four crane periscope heads for Redox, CPD.
13. Fabrication of seven ceramic dies for Ceramic Fuels Development Operation.

Physical Testing

Service testing work continued at a rate comparable to last month. The major emphasis was on special assistance work being done on the NPR primary loop piping. A total of 10,798 tests were made on 6067 items totaling 75,233 feet in length. The greater part of this footage consisted of reactor pressure tubing. Test work included: autoclaving; borescoping; dimensional measurements (micrometric); eddy current (heat treatment); heat transfer (infra-red); mechanical tests (bend, impact, twist and tensile); metallography (macro and micro examination); penetrant (fluorescent O.D. and I.D.); radiography (gamma-ray, X-ray and fluoroscopic); stress analysis (electric resistance wire strain gages and X-ray diffraction); surface treatment (alkaline cleaning, pickling for autoclaving, steam detergent cleaning, vapor degreasing and ultrasonic-alcohol cleaning); magnetic particle; ultrasonic (flaw detection and thickness measurements). Work was done for five operating departments and J. A. Jones representing 35 different organizational components. Advice was given on 50 different occasions on general testing theory and applications.

A work stoppage by Kaiser Engineers and the transfer of manpower to meet several maximum efforts on the NPR primary loop piping problem sharply curtailed output in the tube shop during the month. Salvage operations continued with conditioning of 26 additional tubes which were released to final storage. A total of 975 tubes are ready for reactor installation. There are 26 tubes outstanding on the order of which 10 are at the supplier for sizing, 7 are being reclaimed and 9 are to be made from sponge on hand.

The greater part of the field activity was utilization of X-ray and gamma-ray equipment on the primary injection system and the fuel element rupture test facility at the PRTR site. Additional radiographic work was done at J. A. Jones 3000 Area, 303 Area, 100-H, 100-D, 1706 KER, NPR and Huico plant in Pasco. The fluorescent penetrant testing of over-bore nozzles proceeded routinely.

Ultrasonic thickness measurements were made on two programs. They involved preventative maintenance of oil circuit breakers for the Electric Utility Operation, and pressure vessel surveys at 100-F, 100-D, 200-E, 200-W and PRTR site. Magnetic particle tests were conducted on the rear face gas seal welds in 105-N Building. The strain-gage installation was completed on the gas tubes of 105-C reactor.

In the 300 Area Physical Testing Lab, the number of Zircaloy-4 fuel element sheath tubes received from the suppliers continued to be small. Manpower was mainly applied to cleanup of salvagable tubes and reprocessing through the penetrant tests to determine if acceptable. Review of longitudinal ultrasonic testing by the customer indicated that a better system for scanning the tube was necessary. Consequently, the longitudinal ultrasonic test has been curtailed until a new concept of translating the tube through a miniature tank with fixed transducer proves to be acceptable. Infra-red heat transfer testing of canned fuel elements continued still in the preliminary evaluation stage. Radiographic work on the non-isothermal loop is 100% complete.

Testing work on the NPR primary loop piping continued at an accelerated rate. The statistical study on six welding procedures used by Ladish was extended to include a seventh. Sectioning and metallographic work shows evidence of cracks similar to those found for the first six procedures. Qualification and acceptance of the fusion-welded pipe is now to be based on a guided bend test. Repaired welds did not meet the criteria of this test, and the new weld procedure is still under evaluation. Tests and metallographic examinations were also made on a number of samples from Taylor-Forge, on some seamless pipe samples, and on some Huico weldments.

Several possible methods for testing the primary loop piping for proper heat treatment are being evaluated. Encouraging preliminary results were obtained from laboratory tests of X-ray diffraction methods for residual stress analysis. Eddy current methods gave negative results because of masking effects due to the sandblasted surfaces on the piping.

Two new programs were initiated on determination of length of cracks in Ladish longitudinal weld, and on nondestructive testing of HAPO hand welding. Magnetic particle tests will be made on the prepared sample prior to welding, and then radiograph and penetrant tests will be made on the root pass and each additional weld pass. After angled radiographs of the finished weld, the final test will consist of metallographic examinations for weld quality.

Progress was made on the development of an ultrasonic test for weld defects in SAP (Sintered Aluminum Product) capsules. Ambiguous results were obtained first with the more conventional pulse-echo techniques because of interfering signals from acceptable irregularities on the inside of the weld or end cap. Since the defective welds have always been found to have cracks or unbonded interfaces extending to the surface, a method was needed for confining the ultrasonic energy to the surface. It was demonstrated that this can be done by using one transducer to generate appropriate Lamb waves in the capsule wall, and by using a second receiver crystal mounted at the correct position and angle to detect surface wave nodes generated at and reflected from the surface defect. Good signals were

obtained from defects by this method and tests are under way on a larger number of real and simulated weld defects to fully evaluate the test.

ANALOG COMPUTER FACILITY OPERATION

Studies

The major analog computer problems considered during December include:

1. Reactor Speed of Control
2. Reactor Transfer Function Description
3. Reactor Instrumentation
4. CPD "C" Column

Equipment Operation

The GEDA was 90% and the EASE 90% operational during the month. The computer operations were as follows:

GEDA	EASE	
157	157	Hours up
11	11	Hours unscheduled downtime
24	24	Hours scheduled downtime
0	0	Hours idle
192	192	Hours total

Maintenance

During the last two months, the two instrument men have been alternately working day and swing shifts every other week. All indications to date show that this is a more favorable and effective method of carrying out a preventive maintenance program.

INSTRUMENT EVALUATION

The rechargeable Ni-Cd batteries under test have undergone 65 charge-discharge cycles over an eight-month period with generally satisfactory operation although they now tend to not accept a charge when first applied to the charger. General application use in portable instruments seems quite possible from the economic standpoint.

The experimental transistorized scintillation gamma (background) compensated beta-gamma hand and shoe counter continues to perform satisfactorily in field tests at Purex. No malfunctions have occurred.

The experimental scintillation transistorized combination alpha, beta, gamma hand and shoe counter continues to perform correctly in field tests at Redox and complete satisfaction is reported by the users. Before movement to Redox, the unit was operated next to two obsolescent "4-Fold" (alpha) and "5-Fold" (beta-gamma) counters in the 329 Building. The comparison operation from the sensitivity and maintenance standpoints was [redacted]arkedly in favor of the new scintillation instrument.

Tests continued on a new, transistorized, scintillation alpha portable "poppy" instrument developed to replace the old vacuum tube units. The circuit board used had to be rebuilt because of original fabrication difficulties.

One ORNL-type chirping and flashing gamma intensity level "pencil" was received for evaluation. Initial performance was reasonably good concerning aural indication; however, the light flashes were nearly indistinct. Cost was $78.

One inexpensive "Civil-Defense" type portable gamma ionization chamber instrument was obtained for $50 from the Victoreen Co. The unit uses two standard flashlight batteries, an electrometer tube circuit, and a small transistorized oscillator-rectifier B supply. Ranges are full scale of 5, 50, and 500 r/hr. Possible use as a "disaster" instrument at HAPO is being considered.

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CHEMICAL RESEARCH AND DEVELOPMENT OPERATION

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Reactor Radioisotope Reduction Studies

Sodium silicate added to process water at a concentration of 10 ppm (as Si) was found to inhibit adsorption of As-76 (as arsenate) and P-32 (as phosphate) onto new aluminum turnings at 92 C just as effectively as 50 ppm sodium silicate. A further reduction of concentration to 2 ppm showed greatly reduced effectiveness. Ammonium silicate and silicic acid, each at 10 ppm, were tested and found to give a substantial reduction in arsenate and phosphate adsorption. The silicic acid appeared to be about as effective as the sodium salt but the ammonium salt was significantly less effective.

Addition of 50 ppm (as Si) sodium silicate in process water was made to aluminum turnings on which a corrosion film had been allowed to develop in process water. A very significant reduction in the adsorption of arsenate and phosphate was observed although little change occurred in the appearance of the film. Although the reduction obtained is not quite as great as with new aluminum surfaces, the success of this test has prompted an early trial of sodium silicate as an additive to two process tubes which are neither new nor recently purged. This test was started on December 21 at KE reactor, using one aluminum and one zircaloy tube. The first few days' results show only small changes in the reactor effluent radioisotope concentrations.

Radiochemical analyses of effluent water from two channels which contain nickel clad fuel elements have not shown any significant increase in radioisotope concentrations in the reactor effluent water for the first two-month period studied. The last two samples analyzed, however, have shown an increase in the P-32 by about an order of magnitude. Further sampling is planned to check these results.

Tests were made of arsenate ion removal from dilute aqueous solutions onto positively charged ferric chloride colloids. After 10 minutes of stirred contact the colloidal material filtered off was found to contain 98 percent of the As-76 -- tagged arsenate ion from solutions of pH range from 6.0 to 7.2. Adsorption of the negatively charged arsenate parent material from solution in the process water treatment by the alum floc is similar to this process. Further studies of this type are planned to provide an understanding of the effect of various electrokinetic factors on the removal of trace ions in water treatment processes.

Sorption of Nuclides on 100 Area Soil

A total of 828 column volumes of zinc-65 in tap water have passed through a column of 100 Area soil in the past two months without detectable breakthrough. Accumulative adsorption for this period is approximately 15 μc/gram of soil.

The adsorption of zinc, arsenic, neptunium and chromium on 100 Area soil at 30 C and 60 C has been measured. In batch adsorption experiments zinc and arsenic exhibited only minor temperature dependence, but neptunium adsorption increased from 22 to

39 percent during a temperature increase from 30 C to 60 C. In column runs arsenic adsorption increased from 53 to 73 percent and chromium adsorption increased from 34 to 48 percent for the same temperature change.

SEPARATIONS PROCESSES

Disposal to Ground

Twelve core samples taken from the sediment in the UO_3 Plant south basin averaged 1.4×10^3 μg U/gram of dry sediment. The depth of material in the basin varied, averaging about five inches deep around the periphery of basin with very little sediment in the center. Plans are being made to clean out these deposits.

Tritium fission product is present in the Purex and Redox condensates which are discharged to ground. Water samples were obtained for tritium analysis from 50 monitoring wells located in the region surrounding the 200 Areas. These analyses show concentration of tritium in the ground water up to 4×10^{-2} μc/cc four miles southeast of the 200 East Area, decreasing in concentration in wells farther east nearer the Columbia River. One well, 699-27-8, located three miles west of the river and about seven miles from 200 East Area, showed a tritium concentration at the water table of 4×10^{-5} μc/cc. Twenty-one wells showed detectable amounts of tritium in the ground water. Of these 21 wells, 20 showed the highest concentration at the water table surface. The one well which did not show highest tritium concentration at the water surface is located in the 200 East Area. This well has consistently shown the greatest radioactivity density at depth. Tritium data will assist in determining the zone and rate of maximum ground water movement from the Separation Areas to the Columbia River.

A revision in the present test well sampling and analytical program was made. The new schedule will reduce the total number of samples collected during the year by about 3,000. However, the addition of a large number of tritium analyses will maintain the total analytical cost at about the same level.

Iodine-131 Sampling of Separation Area Stack

Reproducibility in sampling stack gases for I-131 and subsequent laboratory anslysis was determined to be 9, 4, 2, 5 and 3 percent, respectively, in five samplings with four separate collectors operating nearly identically in each run. The error stated is the standard deviation of the four replicates. Reproducibility is considered entirely adequate. Intercomparison of identical samples taken in two samplings from the 20-ft.-level sampler and from the 50-ft.-level sampler revealed that the 20-ft.-level sampler gave higher iodine concentrations by about 20 - 25 percent than samples drawn from the 50-ft.-level sampler. These data were obtained during periods of no metal dissolution taking place in the process and must be confirmed under similar and other processing and weather conditions.

The I-131 loss due to absorption in the condensate in the 20-ft. elevation sampling system was determined to be of little significance, on the order of one percent.

Incinerator Off-Gas Sampling and Analysis

A sample of solids collected in the dry plutonium-waste incinerator off-gas scrubber was prepared for electron microscope examination. The individual particles observed

ranged from about 0.03 to 0.1 micron in diameter, with many agglomerates, some as large as five microns. The small size of particles found is consistent with the observed rather low efficiency of the scrubber. Preparation was made to collect particles directly from the entering gases for electron microscope study. Examination of these particles should better characterize the sub-micron particles to be removed from the hot gases.

Photochemical Preparation of Uranium(IV)

Several bench-scale runs have been made to supplement prior laboratory investigations on the photochemical reduction of uranyl nitrate to uranous nitrate. Hydrazine was used as the reductant in a majority of the runs, although formaldehyde was used in one run. The GE UA-11 photochemical lamp employed is the same lamp that was used in earlier laboratory studies.

The results of the bench-scale studies failed to duplicate the laboratory findings. Under comparable conditions, the reduction rates obtained were one-fourth those measured in the laboratory. This diminished rate was found with either hydrazine or formaldehyde as the reductant. The reduction rates were similar in either stainless steel or glass equipment.

The lower reduction rates have been ascribed tentatively to impurities in the stock uranium solution used in the bench-scale experiments. This stock solution was obtained by concentrating the aqueous product from pilot plant solvent extraction studies with a Purex flowsheet. Another run made with the uranyl nitrate obtained from the dissolution of uranium dioxide yielded reduction rates twice those obtained from the concentrated solvent extraction uranium product but still only one-half the laboratory rates. The investigations are continuing and will explore the effects of using Redox uranium employed in the laboratory studies.

WASTE TREATMENT

Solvent Extraction Treatment of Purex FTW

Batch contact studies were continued to determine distribution ratios for the extraction of inert and radioactive constituents of simulated Purex FTW into D2EHPA - TBP - Shell Spray Base solvent. The composition of the simulated FTW used corresponded to the composition estimated for Purex FTW after certain plant improvements, such as use of U(IV) instead of ferrous sulfamate. The simulated waste was made 0.35 M in acetic acid, 0.12 M in HEDTA (hydroxyethylenediamine triacetic acid) and adjusted to the desired pH with sodium hydroxide. In so doing, the volume was increased by a factor of 2.7. The amount of HEDTA used was 0.05 M in excess of that required to complex all of the iron, chromium, nickel and aluminum present on a 1:1 basis. Distribution ratios for sodium, iron(III), chromium(III), europium, cerium, ruthenium, promethium, zirconium, nickel, strontium and aluminum at equilibrium pH values ranging from 2.9 to 4.6 were obtained. Analytical data are not yet available for nickel and aluminum. Distribution ratios, E_a^o, were less than 0.02 for iron, chromium, ruthenium and zirconium for all pH values. For europium, promethium, cerium and strontium, E_a^o values ranged from one to greater than 1000. They increased in the order given and, except for strontium, increased with increasing pH. The data indicate that separation of strontium and rare earths from other fission products and from inert constituents (other than calcium) should be possible in this system. Extraction of promethium is limiting. Optimum pH

appears to be in the range 3.5 to 4.0. Other distribution ratio studies made show that chromium(III) reacts more slowly with HEDTA than do iron, cerium and promethium and that feed solutions should be allowed to stand at least three hours after addition of HEDTA for maximum complexing and minimum extraction of chromium.

Recovery of Aluminum from Redox Wastes

A few batch contacts were made to investigate the possibility of using D2EHPA solutions to extract aluminum from acidic Redox wastes. Aluminum extracts from a solution containing 0.1 M aluminum and 0.5 M sodium acetate at pH 4.5 into an organic phase of composition 1.2 M D2EHPA - 0.5 M TBP - Shell Spray Base at distribution ratios of about 1000. However, the aluminum cannot be stripped from the organic with dilute (1 - 3 M) nitric acid. Strip distribution ratios, E_a^o, were in excess of 30. In this respect, the behavior of aluminum in D2EHPA solutions is similar to that of iron. Aluminum can be stripped from the D2EHPA solutions with sodium hydroxide.

Waste Transfer Program

The waste tank sluicer has been cold-tested successfully in the 200-W mock-up shop by (1) pumping water at 250 psig through the 13/16 diameter sluicing nozzle, and (2) calibrating the mechanism for measuring the progress of sluicing. Both tests were successful. Sluicer rotation (panning motion) tests will be concluded after fabrication of turning equipment.

Sludge Sampling

Final equipment assembly and tests have been made in preparation for sampling sludge in the 241-A-103 tank by core drilling. A double-core barrel, 7/8 inch ID stainless steel sample tube, and 1-3/8 inch OD holesaw-type steel bit will be used with oil for bit flushing. Sampling will be scheduled after current tank dome coring operations.

TRANSURANIC ELEMENT AND FISSION PRODUCT RECOVERY

Radiolysis in Strontium Citrate Solutions

The final product solution from the Hot Semiworks recovery of strontium may be a strontium nitrate-citric acid-nitric acid solution. Since it may be desirable to store this solution for some time before final precipitation of strontium carbonate, it is of interest to know the effects of prolonged irradiation on these solutions. Solutions containing 2.0 M citric acid, 0.075 M strontium nitrate and from zero to 2.0 M nitric acid were subjected to Co-60 gamma radiation to total doses of 4 and 8×10^8 R. About half of the citrate and nearly all of the nitric acid in these solutions were destroyed at 8×10^8 R. Precipitation of strontium after this exposure varied with the initial nitric acid concentration; about 98 percent precipitated from the solution initially 2.0 M in nitric acid while only three percent precipitated from the solution containing no nitric acid. Irradiation of the solutions to higher doses is in progress.

Ion-Exchange Recovery of Cesium and Technetium from Purex Supernate

A run was completed in the A cell ion-exchange columns investigating the removal of technetium and cesium from Purex waste-tank supernate. The run was similar in all respects to two previous runs except that excess sodium hydroxide, equivalent

to two moles per liter, was added to the feed. The feed was then passed at ambient cell temperature (30 - 35 C) through the two ion-exchange columns: (1) an Amberlite 401, anion column to absorb technetium, and (2) a Duolite C-3 column for cesium absorption. Flow rate was 3 ml/min, cm^2. Technetium breakthrough (50 percent) was reached after passing 470 liters, or about 94 column volumes, through the 2-inch diameter column. This capacity is approximately the same, or slightly less, than observed in the two previous runs, indicating that excess caustic is neither beneficial nor particularly detrimental to technetium recovery (which is already quite satisfactory). The column was then washed with several column volumes of water and 0.25 M nitric acid, eluted with 8 M nitric acid, and the product saved for further processing.

Fifty percent cesium breakthrough was reached at 51 column volumes (equivalent to 32 volumes of tank supernate prior to jet dilution and caustic addition). This represents an improvement in capacity by a factor of 2.6, compared to the runs without added caustic, and is very close to the capacity predicted by small-scale laboratory experiments with synthetic waste. The laboratory experiments show, however, that very nearly the same increase in capacity would be realized with as little as 0.25 M excess caustic and that there is no significant advantage in much higher additions. The cesium was eluted with 1 M sulfuric acid (vice 1 M HNO_3 in the previous run). Peak concentration, about five times feed concentration, occurred at 8 liters of acid and the bulk of the cesium was removed in 65 liters, or about 14 column volumes.

Technetium Purification

The technetium recovered by anion exchange from alkaline Purex waste supernate requires additional decontamination before it can be handled without shielding. Further concentration is also desirable. A series of experiments were carried out to find a way to achieve both these objectives. The resultant process appears fully amenable both to hot cell use and to plant application. It involves the following steps: (1) addition of ruthenium carrier (100 mg/l) and argentic oxide (2 g/l), (2) filtration, (3) passage through a cation exchange column (Dowex-50) to remove zirconium-niobium and ruthenium, (4) subsequent passage through a small anion column to absorb and concentrate the technetium, and (5) elution of the purified technetium product. In the laboratory experiments, using synthetic solutions spiked with hot-cell product, decontamination factors exceeded a thousand. At months end, a large scale run was in progress in the A cell columns to complete purification of the pooled technetium product of the three technetium runs (about 15 grams). Preliminary indication was that in-cell process behavior was fully as satisfactory as in the laboratory.

In other technetium studies, the effect of elevated temperature on the capacity of an Amberlite 401 column for technetium was determined. Increasing the temperature from 23 C to 60 C reduced capacity by a factor of two. Use of temperatures above ambient is accordingly not recommended and there might, indeed, be advantages in providing technetium absorption columns with cooling jackets for low temperature operation.

Rare Earth Recovery

Experiments were reported last month which were aimed at elucidating the unexpected cerium-promethium separation which is observed in the Purex plant double-sulfate

 [redacted]

precipitation. The pnenomena remains something of a mystery in that it has not been possible to fully duplicate it in the laboratory with synthetic solutions; however, the greater solubility of promethium sulfate and/or the greater complexing tendency of promethium may well be involved. It is also conceivable that some cerium(IV) may be formed in the intensely radioactive solutions and that this may improve cerium recovery relative to the trivalent rare earths.

Other works has been aimed at defining conditions for the oxalate precipitation of promethium from the sulfate supernate. Good recovery was obtained over a fairly wide range of conditions. Use of 1 M oxalic acid in any amount over two times the volume of the 1WW, one hour of digestion, and a wide range of pH values (1 to 4 with 2 optimum) were acceptable. Hot-cell experiments, starting with either 1WW or Purex sulfate waste, gave comparable results and lend confidence that a promethium-rich concentrate can be obtained from the plant in this manner. Related hot-cell runs (in support of strontium recovery) were aimed at removing all rare earths directly from 1WW by oxalate precipitation. Although analytical data are not complete, these results also appear promising.

Cesium Solvent Extraction

Additional measurements were made on the extraction of cesium with dipicrylamine (in nitrobenzene) as a function of (1) temperature, and (2) dipicrylamine concentration. The extraction coefficients, E_a^o, increase with decreasing temperature and obey an Arrhenius-type relation, approximately seven kilocalories of heat being evolved per mole of cesium extracted. Dependence on dipicrylamine concentration is first order. The reaction is described over a wide range of conditions by the simple, equilibrium reaction of one molecule of dipicrylamine with one atom of cesium. The equilibrium constant for this reaction (from 1 M NaOH into nitrobenzene) is about 3×10^3.

Both the radiation and thermal stability of dipicrylamine were studied. The "half life" of dipicrylamine to cobalt-60 radiation exceeds 10^8 R. Nitrobenzene is apparently somewhat less stable, as shown by interface "cruds" formed after exposures of 10^7 R; however, its stability is probably adequate and will be further tested in a more realistic use test involving repeated exposures and extractions with the dose tailored to that expected in practice. Heating dipicrylamine or its salts on a hot plate or in a flame resulted only in charring the material (at about 250 C). Striking with a hammer or exposure to high pressure (1500 psi) had no effect.

Two compounds which are structurally rather similar to dipicrylamine, o-nitro-aniline and 2,4-di-nitrophenylamine, were tested and found quite ineffective as cesium extractants. Synthesis and testing of other compounds is continuing.

Special Hazards Studies

A rough draft was prepared of a study evaluating the release of plutonium during a fire. Entitled "Probable Volatilization of Plutonium During a Fire," it is being documented as an unclassified report, HW-71743. The chief conclusion was that the probable maximum percentage of plutonium which could be released during an accidental fire was 0.08 percent. The most likely mechanism for releasing Pu is dispersal of small PuO_2 particles by air currents. High temperature volatilization would become significant at temperatures above 1600 C, but temperatures in that range are extremely unlikely.

Bulk Fission Product Packaging

The development of equipment stations for bulk fission product packaging is now nearing completion. The overall packaging process includes the following steps:

1. Canister filling and thermal stabilization of the product in the canister is performed in a loading station. Process design is based on precipitation of a filterable intermediate compound of the fission product, washing of the precipitate, slurrying of the intermediate into a filter canister wherein thermal conversion to a stable compound is accomplished. With this system, mechanical handling of dry fission product material is avoided.

2. The canister containing the stabilized fission product is removed to a capping station where a helium purge is applied followed by sealing with a mechanical, all metal closure.

3. A third station completes the incapsulation with a remotely-welded closure.

4. Decontamination and leak detection stations will complete the packaging process.

Initial operating experience on some of the above-mentioned functions is described below:

During the initial run of the filter canister loading station, all mechanical equipment operated satisfactorily. The corrugated metal gaskets, sealing between slurry inlet and filtrate outlet, held a 24-inch mercury vacuum with no detectable leakage. The gaskets were compressed in place with an 8-inch bore air cylinder operating with 100 psig air pressure. The air cylinder actuators for positioning the canisters and opening and closing the clam shell induction heating coil operated smoothly.

During pyrolysis at a maximum temperature of 675 to 775 C, the stainless steel surfaces at the container opening must be protected from air oxidation to permit subsequent weld closure. The canister loading head is provided with cooling channels and cold clamp band to lower the temperature of the area to be subsequently welded. These provisions functioned satisfactorily. Later experiments will be run to determine if cooling of the loading head and the weld area of the container is indeed required.

In the induction heating system for the pyrolysis step, the prototype design necessitated use of a stainless steel radio frequency electrical connection between the clam shell coils. The measured low temperature (150 C) of this connector indicates that it will be satisfactory.

In chemical engineering studies of the filter canister loading station, initial strontium peroxide runs have been made. Strontium was precipitated as the strontium peroxide octahydrate, washed in the precipitator vessel, and slurried into the filter canister. An easily noticeable increase in the differential pressure across the filter signalled that the canister was full of filter cake. Continued operation, however, caused the deposition of considerable filter cake in the feed line to the canister indicating that some modification of the loading head will be required to obtain automatic shut-off with highly permeable filter cakes. The filter cake density was 0.30 g Sr/cc. The canister was then heated by the induction heater.

Although an effort was made to dry the cake slowly to maintain a porous cake structure, at some point the heating rate was too rapid, and the cake liquified, filling the space inside the filter stick and plugging the filtrate dip leg. All other phases of the process worked satisfactorily.

In other packaging studies, several ceramic mixes have been found that will bond with strontium oxide at or below the anticipated calcination temperature to produce dustless cakes with improved mechanical stability and hopefully, reduced dispersability in the event of a shipping accident. The simplest binder is a mix of lead oxide and silica. The best bond has been obtained with a frit compound of: 11 percent Li_2O, 22 percent Na_2O, 11 percent NaF, 19 percent Al_2O_3, 9 percent B_2O_3, 28 percent P_2O_5. Ten weight percent of this binder fuses strontium oxide into a glassy melt at about 650 C. No suitable binder has yet been found for cerium oxide. Other binders are being tried, and their mechanical and chemical properties will be determined.

The remote welding station fabrication is proceeding on schedule and is approximately 75 percent complete.

EQUIPMENT AND MATERIALS

Corrosion of Titanium in Fluoride-Containing Purex 1WW

Corrosion testing of titanium in simulated Purex 1WW containing varying amounts of fluoride was continued. Liquid and vapor phase corrosion rates for A-55 titanium in simulated "current" and "future" Purex 1WW containing up to 10 times anticipated fluoride concentrations were all equal or less than 0.1 mils/mo. A titanium bayonet sample was exposed as a heat transfer surface in liquid and vapor phases of boiling current Purex 1WW containing 0.004 M fluoride for 490 hours. There was a temperature gradient along the sample and vapor phase temperature was between 150 and 200 C. There was gross accumulation of salts on the sample in the vapor phase (splashing and drying) but no evidence of accelerated attack or crevice corrosion.

Remote High Pressure Steam Connector

Improved silver reactor performance for I-131 removal, achieved at higher operating temperatures, would be possible if 400 psig steam could be employed in the Purex vessel vent off-gas heater. To this end, a 2-inch Hanford connector, with a Teflon-impregnated asbestos gasket (normally employed for steam pressures up to 100 psig), was evaluated for steam service at 500 F and 440 psig. After 137 hours, the leakage was one ml of water and there was no apparent physical damage to the gasket. A second test is in progress.

Hydraulic Equipment

A canned motor pump with gasketed stator housing failed after 5640 hours of operation with water as a test fluid. Leakage of the water past the O-ring seals into the stator winding chamber resulted in electrical failure, burning out motor windings. In view of the difficulty of modifying this pump to approximate the better design of other canned motor pumps, it was decided to abandon further development of this unit.

A Hastelloy C canned motor pump with 0.050-inch can walls continued to operate smoothly and without incident pumping 50 C water against a 50-foot discharge head. Operation has continued for 4160 hours.

Materials of Construction

A sample of Minnesota Mining "Fluorel" coated Dacron cloth was tested by static immersion at room temperature. It was undamaged by Purex HAX, Recuplex CAX, carbon tetrachloride, and trichlorethylene. Hexone caused a slight softening and partial dissolution of the elastomer and complete failure was observed in 60 percent nitric acid and in 50 percent sodium hydroxide. "Fluorel" coated glass cloth reacted the same as the above sample except that 50 percent sodium hydroxide caused no damage.

A sample of Sierracin 880 F transparent sheet plastic was tested by static immersion at room temperature. This material is an attractive candidate for hoods and glove box panels. It withstood 60 percent nitric acid for approximately 10 days and was etched by trichlorethylene and hexone but was undamaged by Recuplex CAX, Purex HAX, carbon tetrachloride and 50 percent sodium hydroxide. Sierracin 880 F is a water-white, optically-flat, transparent-sheet polyester. The results of this test indicate that it should be seriously considered for field test in both Purex and Recuplex.

PROCESS CONTROL DEVELOPMENT

Recuplex Solvent Extraction Control

A series of C-column runs was completed to compare and evaluate various control techniques having potential for application at Recuplex. Constant ratio control, ratio control with concentration feedback, and ratio control with gradient feedback were investigated. Preliminary analysis of the test results leads to the following conclusions: (a) changes in pulse frequency do not greatly affect the column aqueous profile at constant column density and constant flow ratio, (b) efficiency control is best accomplished by concentration or gradient feedback from measurements in the phase from which solute is transferring, and (c) under conditions studied, gradient feedback control of efficiency is superior to simple concentration feedback.

Development of Improved pH Electrodes

Development of techniques for pH measurement in high activity solutions continued. Specially designed glass electrodes were irradiated in a Co-60 facility to an exposure of 3.9×10^9 R. Extensive radiation damage to the buffer solution was evident at this point. However, the addition of small pressure relief valves has eliminated mechanical breakage of electrodes previously encountered. The exposure attained in the gamma facility corresponds to an approximate nine-month plant operating period in the E-3 waste neutralization tank.

Neutron Multiplication Monitor

An experimental instrumentation system for monitoring neutron multiplication was installed in Purex N cell. The system uses both a BF_3 counter and a solid state detector for comparison of the two techniques. The BF_3 neutron detector is functioning satisfactorily; however, the solid state device (a PN silicon detector) has given somewhat erratic results to date. A voltage regulator was installed at

month's end to improve its performance. Neutron count rates are less than expected, and relocation of the detection instruments adjacent to a vessel containing high plutonium concentration may be necessary to obtain a satisfactory evaluation of the neutron multiplication technique.

Control System for Redox Plutonium Cycle

An instrumentation system has been devised to maintain constant plutonium concentration in the 1B column product stream and to maintain a constant ratio of scrub to extractant stream flows. A key component required for the system is a pneumatic analog computing station. This instrument performs multiplication of product stream concentrations and feed flow rate signals to calculate set points for flow controllers on other column streams. Laboratory calibrations were made of the instrument, and it was found to be sufficiently accurate for the proposed application. The computing station should also be useful in other plant applications where multiplication, division or square root signals are required.

Mathematical Model Studies

Several analog computer runs were made to investigate the adequacy of proposed C column models. It was determined that parameter values which cause the analog output to closely duplicate observed conditions, are greatly dependent on initial conditions or column end values. Concentration of inlet and outlet streams must be accurately known to determine meaningful parameters for a given mathemetical model. Debugging was essentially completed of the IBM-7090 subroutine for non-linear least squares fitting of the model to the experimental data.

Analysis of the calibration data from the C column flowmeters, the mid-column photometer, and the laboratory absorptiometer was completed, and standard errors for each measurement were calculated.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Separation of Plutonium from Uranium in Molten Chloride Systems - The precipitation of plutonium dioxide by metathesis of plutonium chloride with stannic oxide (SnO_2) has been studied in three chloride melt systems: LiCl-KCl, KCl-2.6 LiCl, and NaCl-2.6 LiCl. Each melt contained 0.1 weight percent (w/o) plutonium and either zero or 10 w/o uranium as uranyl chloride. The most promising results were obtained with the LiCl-NaCl system. With this salt solution at a temperature of 575 C, and using a chlorine sparge, addition of one gram of pre-formed SnO_2 precipitated about 45 mg plutonium, a recovery of 90 percent. Plutonium recovery was not affected by the presence of 10 w/o uranium. Decontamination factors of 90 and 260 were measured for the promethium tracer used as a representative rare earth, and separation factors of 150 and 500 for uranium were determined.

Uranium Chemistry in Molten Chloride Salt Solutions - The continuing study of means of separating plutonium from uranium and rare earths in molten chloride salt solutions has led to a closer look at the chemistry of uranium in various melt systems. Most recently, work was begun on an investigation of the rate of air oxidation of lower-oxidation-state uranium to uranium(VI).

Solutions containing up to one weight percent reduced uranium were prepared by reactions thought to produce uranium(IV):

1. UO_2 (suspended in the melt) + HCl (g)
2. UO_2Cl_2 (sol'n) + HCl (g)
3. UO_2Cl_2 (sol'n) + C + Cl_2
4. Dissolution of Cs_2UCl_6 in the melt

The melts were sparged with air at 600 C and the reduced uranium content (measured as uranium(IV) in an aqueous solution of an aliquot of the melt) was determined as a function of sparging time.

In the melt systems employed in this work (LiCl-KCl, KCl-3LiCl, and NaCl-3LiCl), the lower-oxidation-state uranium formed by HCl sparging UO_2Cl_2-bearing melts was found to exist in at least two species, one being oxidized much more rapidly than the other during an air sparge. The relative amounts of the two species were different in the different salt systems, the rapidly oxidized fraction being lowest in LiCl-KCl and highest in NaCl-3LiCl. In KCl-3LiCl, dissolution of UO_2 by HCl was found to proceed more rapidly in the presence than in the absence of uranium(VI). The lower-oxidation-state uranium thus obtained in the absence of uranium(VI) was apparently all the same species in that it was all slowly oxidized by air. The lower-oxidation-state uranium obtained in the presence of uranium(VI), however, was found to exist in at least two species.

Lower-oxidation-state uranium obtained by the reaction of uranyl chloride with carbon and chlorine or by dissolution of Cs_2UCl_6 in the melt was apparently all the same species in that it was all slowly oxidized by air. The rates of oxidation in these instances agreed quite well with the rates of oxidation of the slowly oxidized fractions obtained by HCl sparging UO_2Cl_2-bearing melts or by dissolving UO_2 with HCl in the presence of UO_2Cl_2.

These results, while illustrating differences in different melt systems, do not permit complete interpretation. It is thought likely that the rapidly oxidized material is either a uranium(V) species or uranium(IV)-oxychloride (or a mixture of the two) and that the more slowly oxidized material is non-oxygenated uranium(IV)-chloride.

Studies of Electrode Processes in NaCl-KCl Molten Salt Systems - EMF values for the reaction

$$UO_2\ (s) + Cl_2\ (g) = UO_2Cl_2\ (sol'n)$$

have been measured in NaCl-KCl at temperatures of 670, 720 and 770 C and at UO_2Cl_2 concentrations of 0.03 to 0.9 molal. From these data, the oxidation potential and free energy change for the reaction, with a reference state of 1.0 molal UO_2Cl_2 at 720 C, have been calculated as 0.550 ± 0.003 volts and -24.4 kcal/mole, respectively. Reference to a standard state and calculation of standard EMF and free energy values have not yet been made, because a needed activity coefficient term cannot yet be evaluated.

Preliminary results reported last month regarding the mechanism of cathodic reduction of UO_2Cl_2 in molten NaCl-KCl have now been verified. Refined measurements of capacitance as a function of potential have shown that the following reactions occur:

(1) $UO_2^{++} + e^- = UO_2^+$

(2) $UO_2^+ + e^- = UO_2(s)$

The capacitance voltage plots show two capacitance peaks. The first is a symmetric peak which does not change potential with a change in UO_2Cl_2 concentration and is indicative of a reaction with a soluble product. The second is an asymmetric peak for which the potential is dependent upon concentration in a fashion indicating a one-electron change and an insoluble product.

UO_2 Crystal Growth Studies - UO_2 crystal growth (by cathodic deposition from UO_2Cl_2 solution) in LiCl-KCl is quite similar to growth in the $PbCl_2$-2.5 KCl system. UO_2 crystals weighing as much as 1.5 grams have been produced, with O/U ratios as low as 2.005 and impurity levels of < 20 ppm potassium and 5 ppm lithium.

Studies of the effect of melt agitation on crystal growth phenomena in $PbCl_2$-2.5 KCl failed to show any variation in crystal properties with various rates of agitation.

Melt "Dryness" Determination - A major problem in pilot plant studies of the production of UO_2 by cathodic reduction in molten chloride salt solutions has been control of melt moisture content and, hence, O/U ratio in the product. To assist in resolving this problem, a practical means was devised for determining the relative "dryness" of a melt. The method is based upon measurement of the shape of the curve for current vs. time as plotted at the start of a micro-scale deposition carried out in the melt. Laboratory experience has shown a definite correlation, "wet" melts giving a rapid increase of current due to rapid increase of cathode surface area with the formation of a needle-type deposit. "Dry" melts, on the other hand, show a decrease of current with time, as the relatively smooth UO_2 layer builds up on the cathode without any great increase in cathode surface area.

Determination of Lattice Parameters with an X-Ray Spectrometer - The measurement of crystal lattice parameters by X-ray diffraction is usually done photographically with a precision focusing-type camera. Interest in a less time-consuming method for such applications as the determination of O/U ratios in non-stoichiometric UO_2 led to the development of methods permitting the use of an X-ray spectrometer. The two methods developed, while lacking the precision of the photographic method, were found to be adequate for the purpose stated.

In one method, ThO_2 is added to the UO_2 as an internal standard, to permit correction of the measured UO_2 lines for errors resulting from sample absorption and instrument misalignment. This method required an hour's work per sample and gave a precision of about ± 0.0003 A° in the lattice parameter. The other method involves plotting the lattice parameter calculated at each angle vs. the function

$$1/2\left(\frac{\cos^2\theta}{\sin\theta} + \frac{\cos^2\theta}{\theta}\right)$$

and extrapolating the linear plot to $\theta = 90°$. Errors due to sample absorption and instrument misalignment become smaller as θ approaches 90°. Furthermore, precise sample alignment was found to be unnecessary.

Strain Measurement Near Pipe Welds - The X-ray spectrometer was used to measure strain in the neighborhood of welded connections in heavy walled, carbon steel pipe. The method is based on measurement of the variation of the lattice parameter (a) with strain in the metal. Preliminary tests showed welding-induced stresses as high as 50,000 psi, with an estimated measurement precision of ± 4,000 psi.

Reduction of Uranium by Aluminum Metal in Molten $AlCl_3$.KCl - The computer study has been completed on this system using a new value for the entropy of solid $AlCl_3$. This, of course, affects the value for K_1. The re-evaluated equilibrium constants (expressed in mole fraction) are given below:

$$UCl_3(l) + Al(l) \overset{1000°K}{\rightleftharpoons} U(l) + AlCl_3(l);\ 1x10^{-10} \leq K_1 \leq 3.55 \times 10^{-8}$$

K_1 used = 1.82×10^{-9}

$$2AlCl_3 + Cl^- \rightleftharpoons Al_2Cl_7^- \ ;\ K_2 = 5.9 \times 10^{13}$$

$$2Al_2Cl_7^- + Cl^- \rightleftharpoons 2AlCl_4^- \ ;\ K_3 = 1.22 \times 10^{3}$$

$$UCl_3 + 3Cl^- \rightleftharpoons UCl_6^{\equiv} \ ;\ K_4 = 6.6 \times 10^{4}$$

The uncertainty of K_1 is reflected in K_2 more than in K_3 or K_4 as is shown below from previous calculations:

$K_1 = 1.25 \times 10^{-6}$

$K_2 = 1.34 \times 10^{8}$

$K_3 = 2.0 \times 10^{3}$

$K_4 = 8.9 \times 10^{4}$

An article for publication is now in preparation covering this study.

Engineering Studies - The UO_2 product from previous pilot plant electrolysis in the $PbCl_2$-2.5 KCl systems has had O/U ratios above 2.02, and a fraction of the deposit next to the electrode has had high void spaces and a higher O/U ratio. By sealing all holes in the lid with silicone rubber gaskets and sparging with 51 pounds of chlorine (a 1.4-fold excess) over a 21-hour period to dissolve the charged U_3O_8, a 40-pound deposit with an average O/U ratio of 2.011 was produced during electrolysis. However, there was still poor quality UO_2 next to the electrode and the current density could not be maintained as high as in previous runs.

For the next run, to insure that the surrounding atmosphere did not reach the molten salt, a Hastelloy C partition was placed in the vapor space, dividing it into a lower chlorine atmosphere zone and an upper zone in which the chlorine is diluted with inleaking air. A micro-scale deposition system (described above) was employed to determine the "dryness" of the melt, which is a measure of the suitability of the salt for electrolysis. Chlorination with 157 pounds of chlorine (a 5.3-fold excess) over a 54-hour period at 700 C dissolved the U_3O_8 charged and dried the melt considerably. A two-hour HCl sparge at 600 C and a one-hour pre-electrolysis with graphite electrodes at 600 C were also effective in drying the melt further. The pre-electrolysis cathodes with the deposited UO_2 became the anodes for the product electrolysis.

A 50-pound deposit with an O/U ratio of 2.0036 was produced by electrolysis with a 61 percent current efficiency over a 113-hour period at 580 C. The low density structure usually found near the electrode was entirely absent. The melt was much drier after the product electrolysis than it had been before. The current density was less than 50 percent of what it had been when higher O/U ratios were produced, indicating a possible limitation on production rate.

Examination of the deposit and the Hastelloy D cathode revealed a small amount of undesirable corroded metal which adhered to the inner surface of the deposit, and accelerated interfacial corrosion of the cathode. This, together with the inherent brittleness of Hastelloy D, suggests that a better metallic cathode should be developed.

Electrolytic UO_2 Grinding and Classifying - In scouting studies of hydraulic classifiers, hydroclones and sorting columns showed little promise for electrolytic UO_2 classification, while a "centri-clone" offered good separation in the 3 to 10 mesh range. The "centri-clone" is a cone-bottomed fluid bed with a tangential fluid inlet. Studies will continue on (1) centri-clone effectiveness on smaller particles, (2) hydraulic dense-phase transport of UO_2, (3) screen classification, and (4) mechanical grinding.

Frozen Wall Containers - Work was started on development of equipment and techniques for the application of frozen salt wall techniques in molten salt containers to circumvent the limitations currently encountered with ceramic containers.

A charge of 42 m/o KCl in LiCl was introduced into a 304 L stainless steel water-cooled pot. A melt of 420 C was obtained with a graded frozen wall of 3/8-inch to 2-inch thickness. Tubular resistance heaters supplied 2500 watts of heat for 300 hours of successful continuous operation.

Melt temperatures excursions to 550 C were obtained by immersion of graphite electrodes into melt and passing alternating current at 600 amperes and 10 volts. A graded solid salt wall was maintained at 1/8-inch to 1-1/4-inch thickness.

Materials of Construction for the $PbCl_2$-KCl System - Samples of Hastelloy B, Hastelloy C, Hastelloy D, "A" Nickel, Permanickel, Duranickel, Haynes 25, Haynes R-235 and Haynes Multimet have been exposed to a chlorine-sparged 60 w/o $PbCl_2$ - 40 w/o KCl melt at 600 C for six hour periods. Corrosion rates based on overall weight loss were 930, 1250, 840, 130, 140, 27, 380, 270 and 650 mils/mo, respectively. All samples except Duranickel showed accelerated attack at the melt-vapor interface.

Dissolution of PRTR Spike Fuels

Efforts to devise a more satisfactory procedure for dissolution of PRTR spike elements in the Redox multipurpose dissolver were continued. Attention has been concentrated on a solution containing 6 to 8 M nitric acid, 0.002 to 0.05 M mercuric nitrate and 0.3 M ammonium fluoride. Activation of the spike element alloy at 0.002 M mercuric nitrate is not reproducible. Activation at higher mercuric nitrate concentrations (0.005 or greater) is reproducible but initial dissolution rates are too high to permit charging eight PRTR assemblies to the dissolver at one time. Preliminary results indicate control of the dissolution rate by the presence of low concentrations (0.01 to 0.2 M) of nickel salts in the solvent is possible. This approach is being explored further.

RADIOACTIVE RESIDUE FIXATION

Ruthenium Chemistry

The vapor pressure of ruthenium tetroxide over nitric acid was measured as a function of concentration and found, as with sulfuric acid, to decrease with

increasing concentration of nitric acid. Whether this decrease is due to the basic character of the tetroxide or to complex formation is uncertain.

The absorption spectra of ruthenium tetroxide was measured with the Cary spectrophotometer, and a paper is being prepared describing and interpreting the results. The Model 12 Cary was also modified to permit measurement of the kinetics of ruthenium tetroxide decomposition at controlled, elevated temperatures, and preliminary measurements have been made establishing the feasibility of the method for following the gas phase reaction.

Strontium Glass Formation

Two runs were made in the "cold" 8-inch demonstration spray calciner to explore applicability of spray calcination to strontium glass formation. The runs were scaled to be equivalent to 30,000 curies of strontium-90 or 375 grams total strontium per run (about a kilogram of glass). Results were only partially successful. Most of the Calrod heaters in the upper section of the calciner have failed and, even with channel recycle, it was not possible to heat the top of the calciner hot enough to achieve efficient drying. Consequently, drying of the spray droplets was ineffective and much of the feed tended to adhere to the upper part of the column, in the nozzle region, and did not fall into the melt pot. A "mastic" stage was also observed during the drying and may have contributed to this difficulty. Quality of glass produced was, however, very good and off-gas decontamination from strontium was excellent.

In-Cell Calciner

Mock-up of the in-cell calciner equipment is now estimated at 45 percent complete and should proceed rapidly. The gear pump used on the demonstration unit failed during the month and was found to exhibit extensive wear of the end faces of the gears. On the basis of this experience, it was decided to use a diaphragm pump for the in-cell unit. This will be tested on the mock-up. A magnetic rotameter controller is being evaluated for possible use in monitoring and automatically controlling the flow of feed. Development work is also being done on a pot scanner (to determine solids level in the melt pot), and design is continuing on racks and shielding needed for the in-cell installation.

Kinetic Studies of Ion Exchange Materials

The effects of temperature and flow rate on loading rates of cesium in the film diffusion region were ascertained.

Temperature changes caused variation in the value of the liquid diffusion coefficient that, in turn, led to changes in cesium loading rates. The diffusion coefficient normally increases about two percent for each degree rise in temperature. For example, the diffusion coefficient for K^+ is 1.69 cm^2/day at 25 C, compared to 5.6 cm^2/day at 100 C. The effect of temperature on loading rate is proportional to effects on the liquid diffusion coefficient.

Loading rates were found to vary linearly in respect to linear flow velocities within the ranges studied. When velocities increased, loading rates increased a proportionate amount. With the relationship between loading rate and flow velocity determined, the decontamination factor that can be attained with a given column and

flow velocity can be computed. Average film depths were computed by two different methods for clinoptilolite. Under the conditions of this study, the two film depths obtained were 13.3×10^{-4} cm and 12.4×10^{-4} cm. The film depth may be used to compute other unknowns, such as particle radii, or the loading rate itself.

Elution of Cesium from Duolite C-3 with Ammonium Salts and Sulfuric Acid

Ammonium salts are superior eluting agents to sulfuric acid, but mixtures of the two are not greatly different than the salts alone. For example, 3.4 column volumes of 5 N ammonium sulfate elute 98 percent of the cesium vs. 10.3 column volumes of 5 N sulfuric acid. A mixture containing 4 N ammonium sulfate and 1 N sulfuric acid elutes 98 percent of the cesium in 2.7 column volumes.

Cesium Removal from Synthetic Formaldehyde-Treated Waste

Synthetic waste, defined in HW-67728-Pt. 2, as diluted, centrifuge-clarified, formaldehyde-treated Purex waste (including cake wash), was used for studies on high efficiency cesium removal. Column experiments with clinoptilolite have been completed at residence times, 1, 5 and 25 minutes. The column volumes to 50 percent cesium breakthrough are 92, 116 and 114, respectively, and the two percent breakthrough volumes are 20, 44 and 94 column volumes, respectively.

Based on results of these tests, a conceptual-plant column one foot in diameter and 10 feet long, with a second column in series, could process the waste equivalent of 29 tons of uranium per day. The waste from about 135 tons of uranium could be processed for highly efficient cesium removal before breakthrough of the second column would be evident (e.g., one percent). Laboratory work is continuing to provide necessary data for scoping this type of a cesium removal step.

Condensate Wastes

Equilibrium experiments show that clinoptilolite is highly selective for both trace cesium and trace strontium in solutions containing magnesium and aluminum ions. A bed of clinoptilolite would be expected to remove trace cesium and trace strontium from over 100,000 bed volumes of 0.01 M magnesium or aluminum ion solutions before reaching 50 percent breakthrough. Magnesium and aluminum ions are less competitive for exchange sites on clinoptilolite than sodium. For example, a K_d value of 4×10^5 for trace strontium in 0.01 M magnesium ions is observed vs. 1.2×10^5 in 0.01 M sodium ion. The more normal competitive effect is observed with Amberlite IR-120 (a sulfonated polystyrene resin) where K_d values of 8×10^2 and 2.5×10^5 are observed for 0.01 M magnesium and sodium ions, respectively.

Micro Pilot Plant Run 20, evaluating the decontamination performance of a synthetic alumino-silicate gel zeolite (Decalso) in the sodium form was initiated. Feed to the column of Decalso was Purex Tank Farm condensate which had been steam stripped to remove ammonia and organic. After the stripping operation, the pH of the stream was about 7. Operation during the initial part of the run indicated that the Decalso satisfactorily removed cesium-137 to below its MPC_w, but the strontium-90 concentration in the effluent always exceeded its MPC_w. Steam stripper malfunction caused by freezing weather conditions prematurely terminated the run.

The steam stripper equipment was "winterized" and Micro Pilot Plant Run 21 was performed to re-evaluate steam stripper efficiency for ammonia removal. The feed to

the stripper contained about 100 mg NH_3/l, and the ammonia concentration in the stripper bottoms remained fairly constant at about 2.3 mg NH_3/l; these data indicated normal stripping efficiency.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Geology and Hydrology

Well drilling footages to the terminating formations exceeded estimates by only 1.7 percent on completion of the first 15 of the 16 wells on Project CAH-921. However, five of the ten wells completed to basalt encountered bedrock at depths varying more than 50 feet from the predicted depths. Well 699-53-55 hit basalt 154 feet deeper than predicted. Those wells defined bedrock location where knowledge of the basalt surface was inadequate. Inasmuch as the altitude and attitude of the post-basalt sediments are in part determined by those of the basalt, inadequacy of knowledge of those sediments was also demonstrated.

The first satisfactory measurements of Hanford soil parameters (capillary conductivity and capillary pressure) under unsaturated flow conditions were achieved. A petroleum industry core test fluid was used in the tests in preference to water. Methods exist to convert results to a "water basis." Capillary pressures up to 85 cm of oil were obtained with 150-fold reduction in capillary conductivity. The displacement pressure for the oil was about 8 cm. The data obtained from these measurements are used in crib-drainage and tank-leakage type problems.

The semi-annual ground-water contour map was completed this month. A ground water mound is now shown beneath the 105-B Area. Evidence of this mound was obtained from a leveling survey of the monitoring wells in the vicinity of 100-B. It is believed that mounds exist beneath all of the reactor areas, but they cannot be delineated in some cases because of the limited number of monitoring wells. The 405-foot contour of the 200 East mound encloses a larger area to the north and east than is shown in previous maps. The height of the mound beneath the 200 East swamp has gradually lowered four feet, while the elevation of the water table beneath the Gable Mountain swamp shows a continued rise, thus indicating the expansion of the mound to the north.

Tests were made on the two-dimensional, fluid-flow analog model to determine effects of resistance (permeability) change on observed potentials. A resistor near the center of the model was changed by 20 percent. This large change in resistance caused only a 7 percent change in potential adjacent to the resistance change. The resistance change did not cause a potential change near the analog boundaries nor along adjacent streamlines. These tests indicate that accurate resistance (permeability) values may not be determined from a study of potential patterns. Another two-dimensional, fluid-flow analog model was designed and is being fabricated. This model will be used to test digital methods of calculating resistance value from analog potential measurements.

Preliminary permeability measurement using the well-packer method was made at the 200-210 foot depth in well 699-37-82A. Permeability was measured at approximately 1 $ft^3/ft^2/hr$ (200 gal/day/ft^2).

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Radiation Chemistry

The "Gammacell 200" cobalt-60 irradiator was received from Atomic Energy of Canada, Ltd., and put to immediate use. Nominally it contains 1500 curies of cobalt-60 but since no isodose curves were received for its sample compartment, calibration was necessary. This was done using the standard ferrous sulfate dosimeter. The absorbed dose in 5-ml volumes of aqueous solution was measured when this solution was contained in the screw-cap bottles used for erythrocyte irradiations. Samples were arranged on marked positions in the irradiation compartment and the absorbed dose was found to be independent of position to within 1 percent. The absorbed dose rate was found to be $(5.3 \pm 0.01) \times 10^3$ rads min^{-1}.

The new cobalt source was used for the irradiation of erythrocytes and found very satisfactory and a considerable improvement over previous irradiation facilities. Additional runs have appeared to confirm the relative abilities of valine, histidine, tryptophan and erioglaucine as protective agents. Analysis of the data is still tedious. At the radiation doses used, the ferrous sulfate dosimeter is inapplicable due to oxygen depletion in the solution. This condition does not appear to bother the protected erioglaucine dosimeter. The versatility of this dosimeter can be seen in this application where absorbed doses around 10^5 rads were given. In general the absorbed dose which will leave 37 percent of the dye unbleached is given by: $D_{37_p} = 500 + 7.60 \times 10^8 C_E^o + 7.60 \times 10^8 (P.I.)_p C_p^o$

where C_E^o is the initial concentration of erioglaucine and C_p^o is the initial concentration of the protective agent whose Protection Index is $(P.I.)_p$. Using ethanol with a protection index of 0.127, the D_{37_p} value and hence the useful range of the dosimeter can be varied from 5000 to 10^7 rads merely by varying the ethanol concentration from 0 to 0.1 M. For doses in the 10^5 rad range, an ethanol concentration of 5.7×10^{-4} M was selected.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Aerosol Generation and Characterization

Polystyrene latex spheres of a narrow-size range were utilized to calibrate and study the performance of a commercial particle counter. Air flow rate through the instrument was found to influence the size distribution curve, and the sum of particles counted in various size increments did not equal the integral count over the same size range. These performance characteristics were not resolved. Adapting the basic optical system to pulse height analysis and readout with a multi-channel analyzer is being attempted. An instrument which will accurately record size distributions will be of great worth in anticipated studies.

Particle Deposition in Conduits

Methods for controlled dispersal of dry powder and uniform injection into test sections were investigated. A turn-table and air-aspirator device was found to have the serious drawback of non-uniform delivery rate. An air-aspirator concept was initially tested and holds promise for the controlled dispersal of very low concentrations of dry powder.

W. H. Reas
Manager
Chemical Research and Development

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[redacted]

BIOLOGY OPERATION

A. ORGANIZATION AND PERSONNEL

No significant changes occurred during December.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

The installation of the improved river water supply system in the 1706-KE laboratory was initiated by Minor Construction.

Columnaris

A trip to Oxbow Dam was made and samples taken from both salmon and scrap fish (202 total). Two of 87 salmon were positive for columnaris, one from the gill and one from the kidney. All other samples were negative. Due to the low water temperature, no conclusions can be made as to the incidence of the disease at Oxbow.

Aside from the low incidence (two fish out of 87 fish sampled) of columnaris found, this field trip emphasized again the need to distinguish clearly between populations of fish in the river and groups of fish held under adverse artificial conditions. Although no data are available, the high mortality reported (42%) among the salmon spawners at Oxbow was most likely because of adverse holding conditions.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Phosphorus

The experiment to determine the effect of chronic feeding of P^{32} on the reproduction of chichlids was initiated with 960 three-month-old fish. The four treatment levels with replicates are 0, 0.25, 1.0 and 4.0 μc/g food. Radiation damage will be measured in part by mortalitieis among the initial group and its progeny and the success of reproduction. Some of the fish are expected to reach sexual maturity during early summer. The test will continue about a year.

Strontium

When phosphate (Na_2HPO_4) was added to either the external or internal compartment of the trout gill perfusion preparation, the uptake rate of strontium increased from 5×10^{-8} to 35×10^{-8} cm/sec. We are trying to find if this phenomenon can be cancelled with metabolic poisons.

Additional data obtained on the absorption of Sr^{85} and Ca^{45} from the perfused small intestine of the rat indicate no effect of pH in the range from 5 to 7. At pH 4 and 3, however, the uptake of Sr^{85} increased as pH decreased, while Ca^{45} uptake remained unchanged.

Female swine maintained on 3.1 mc Sr^{90} per day for 60 days have exhibited a progressive drop in leukocyte counts to approximately one-third pre-feeding or normal values. The neutrophils have shown the most marked decrease and after two months are about one-tenth of the normal value. Platelet counts are approximately one-third of normal values. Lymphocyte values are only slightly reduced. One animal has developed a definite hemoconcentration with hematocrit measurements 25 per cent above normal. Erythrocyte numbers, hemoglobin, mean cell volume, mean cell hemoglobin have remained normal.

One animal at this level died 41 days after initiation of the Sr^{90} feeding, due to cardiac trauma during routine bleeding. There were no outstanding lesions excepting a severe aplasia of the bone marrow evident at autopsy.

Neptunium

Dialysis studies have been initiated to determine the ability of plasma to bind plutonium, neptunium, and uranium in vitro. Plasma from rats injected with these materials will also be subjected to dialysis under varying conditions to compare in vitro binding with physiological binding. Electrophoretic studies of individual plasma protein fractions are planned.

Arrangements are being made to obtain Np^{237} from Oak Ridge, and preparations for its counting are being made using Ce^{144}-Pr^{144} as a stand-in.

As a follow-up on an earlier observation which suggested chemical toxicity due to neptunium injection in sheep, a young ewe was administered 12 mg Np^{237}/Kg (about 5 [illegible]) intravenously. The animal died on the second day after injection, manifesting severe hemorrhages in the liver.. Preparatory to further toxicity studies with neptunium, a study was performed on the use of I^{131}-labeled Rose Bengal as a liver function test in sheep. Clearance of the intravenously injected labeled Rose Bengal from the blood is determined by monitoring a vascular area in the anterior neck region near the angle of the jaw. Data from a limited number of animals revealed that thirty to fifty per cent of that observed at two minutes post-injection was still present in the blood ten minutes after injection. (This method of expression was used due to the non-exponential character of the disappearance from the blood.) This early rate of loss is approximately twice that observed in humans.

Plutonium

Based on the amount of plutonium retained in the rat femur five days post-treatment, neither urethane (prior to or following plutonium injection) nor ultrasonic treatment (following plutonium injection) had a significant effect on plutonium deposition, under normal circumstances, or when the animals were treated with DTPA.

Fifty-two intradermal injections of $Pu^{239}(NO_3)_4$ in doses varying from 0.0016 µc to 5 µc/site were made in an additional two miniature swine. This brings the total number of swine injected to eight. In this last series of injections, special effort was made to determine the early loss of plutonium from the site of injection. In the one and 5 µc sites, about 80 per cent of the plutonium

detected one hour after injection still appeared to remain in the injection site after eight days. This, of course, assumes no appreciable change in counting efficiency, e.g., an increase in tissue or fluid between the deposited plutonium and the monitoring device. The 5 μc sites showed minimal changes by eight days, with only slight redness and swelling and pin-point scabs evident.

Radioactive Particles

One of 22 dogs exposed more than two years ago to plutonium dioxide aerosols is showing respiratory distress with abdominal breathing and a three-fold increase in respiratory rate. Twelve other dogs are being prepared for addition to this experiment. Pre-exposure radiographs and blood data are complete.

Excretion and retention of plutonium in three dogs given plutonium nitrate by intravenous injection were compared with three dogs given plutonium nitrate by inhalation. Thirty days following intravenous injection about 80 per cent of the body burden was in the liver, 7 per cent in bone, 6 per cent in spleen and 3 per cent in muscle. Following inhalation, about 70 per cent was in lung, 10 per cent in liver, 16 per cent in bone, 0.1 per cent in spleen and 2 per cent in muscle. Ten days following intravenous injection of plutonium nitrate the daily urinary excretion of plutonium was about 0.01 per cent of the body burden and at 30 days it was about 0.006 per cent. Ten days following inhalation the daily urinary excretion was about 0.03 per cent of the body burden and at 30 days it was about 0.015 per cent. Throughout the 30-day period urinary excretion of plutonium was consistently greater in all dogs given plutonium nitrate by inhalation. The conclusion is that the rate of urinary excretion of plutonium is markedly dependent upon the distribution of plutonium within the body.

Bone Marrow Studies

Techniques have been perfected to the point where C3lF$_1$ strain mice injected with 9 x 10^6 cells of isologous marrow within three hours of irradiation (950 r) show 100 per cent survival after 30 days. Unprotected, similarly irradiated, animals show no survival. Isologous spleen stored over night in the refrigerator shows similar protective effects. Skin grafting and Ouchterlony techniques are also being investigated as tools for use in this area of study.

Basic Radiation Effects

The absorption of C^{14} labeled alpha-amino isobutyric acid (AIB) in subcellular fractions of normal and regenerating liver was studied. Very preliminary data (two animals) indicate that three hours post-hepatectomy three to five times as much AIB is present in the regenerating liver as is present in controls. AIB is neither metabolized nor incorporated to an appreciable state into proteins and has been used as an indicator of amino acid transport.

Studies on DNA, RNA and protein synthesis have shown that DNA synthesis occurs at mid-interphase, RNA during all periods except metaphase and anaphase and protein synthesis goes on over the entire mitotic cycle. These studies are preliminary to a study of radiation effects on RNA and protein synthesis.

The sigmoidal nature of the X-irradiation survival curves for yeast cells grown in D_2O can be changed to a simple exponential curve by using old cultures or shaking suspensions with glass beads. This would indicate that the sigmoidal curves were experimental artifacts. Deuteration was shown to have no effect on endogenous respiration in yeast, whereas marked inhibition of respiration is seen when an exogenous substrate is added. These data indicate that effects on uptake mechanisms are the primary cause of physiological disturbances rather than D_2O effects on the enzymatic reactions involved.

Single and mixed species populations of Tribolium confusum and T. castaneum were given 0 to 6,000 r from X ray. Radiation effects on populations, as indicated by total number of adult progeny per female for 10 weeks, were comparable for single and mixed species cultures. Life span of adult T. confusum was shortened by 6,000 r. All other groups showed no significant reduction in adult life span and their reproductive rate increased with time.

Plant Ecology

The pH values of soil profiles under greasewood and sagebrush communities on the Hanford reservation were compared. Alkalinity generally increased with depth, lowest values always occurring in the surface layer. On the surface of greasewood soil profile pH values ranged from 7.2-7.7 to 10.3; while under sagebrush pH values ranged from 8.0 on the surface to 9.2 at 1 meter depth.

Fallout Studies

Iodine-131 concentrations in thyroids of native deer and caribou continued to decrease from peak values which occurred about October 26 in Alaska caribou and about November 2 in Colorado mule deer. During the peak period Colorado thyroids often contained greater I^{131} concentrations than did Hanford deer thyroids. During December, Colorado samples increased steadily in radioiodine concentrations while thyroids from other locations decreased with time. Relative thyroidal I^{131} concentrations at mid-month were as follow:

Hanford mule deer	- 100
Colorado mule deer	- 50
Maryland whitetail deer	- 10
Alaska caribou	- 4

General

R. L. Junkins was a guest at a meeting of Biology's Internal Emitter Committee. Potential hazard problems associated with the separation and shipment of megacurie quantities of fission products and with the possible use of reactor fuel elements using uranium blended with plutonium were discussed.

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C. Lectures

a. Papers Presented at Meetings

None

b. Seminars (Off-Site)

J. J. Davis, "Effects of environmental factors upon accumulation of fallout in natural populations," Annual Called Conference of Oregon State Health Officers, Portland, Oregon, December 7, 1961.

L. K. Bustad, "Studies on host response to disease; application of radionuclides," Dedicatory Symposium for NADL, Iowa State University, Ames, Iowa, December 12, 1961.

c. Seminars (Biology)

J. F. Cline, "Radiobiology in Tunisia," December 5, 1961.

N. L. Dockum, "Autoradiography of Zn^{65} in Rat Tissue," December 5, 1961.

J. R. McKenney, "Studies with calcium, strontium, and phosphorus in sheep serum," December 19, 1961.

D. R. Kalkwarf, "Electron spin resonance and charged transfer complexes," December 19, 1961.

d. Miscellaneous

Several talks were presented by Biology personnel at the Science Fair Workshop, Columbia High School, Richland, December 4 and 11, 1961.

D. Publications

a. Documents

None

b. Publications in Open Literature

Sullivan, M. F. 1961. Absorption from gastrointestinal tract of the rat after X-irradiation. Am. J. Physiol. 201: 1013-1016.

OPERATIONS RESEARCH AND SYNTHESIS OPERATION
MONTHLY REPORT - DECEMBER, 1961

ORGANIZATION AND PERSONNEL

There have been no changes in organization or personnel in Operations Research and Synthesis during December, 1961.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Fuels Preparation Department

Fuel Element Performance

Further work was done in connection with the assessment of the accuracy and precision of variables measurements performed on the pre-irradiated fuel element. Variances and covariances of the parameter estimates were derived, and a sampling experiment was conducted in order to verify the results. A discussion of the model is contained in HW-72118. An IBM-7090 program, called the MERCY program, is available for rapid analyses of data using this approach.

A report was issued jointly with W. F. Stevenson in which a general approach to the problem of routinely checking measurement stations within FPD was presented. Initial data from the UT-2 tester, taken in four testers over four weeks time are being analyzed using the MERCY program, and will serve as the basis for control of variables data originating at this tester. Initial data from the bond testers were also collected and analyzed. Recommendations were made to select additional standards prior to embarking on the routine program of assuring valid results.

The analysis which investigated segregation of impurity content and grain size along an ingot was completed. In breaking down the interaction sum of squares to individual sets of degrees of freedom, corresponding to the linear x ingot, quadratic x ingot, etc., effects, insight was given as to consistency of segregation from ingot to ingot.

Several discussions were held relative to the sequential testing of component vendors with respect to rupture performance. Specifically, a brief investigation was made of the effect of a different ratio of ingot-to-ingot cores in the test of each vendor, and recommendations were given for removing effects of possible bias in the results due to this cause.

Initial work was done in re-evaluating dimensional distortion models based on Quality Certification data. Data are being used from the initial four runs in which 80 and 90 type profiles, indicative of gross measurement errors or unusual profiles, are being eliminated.

General

Data were analyzed from an experiment designed to evaluate possible sources contributing to variations in cooling rates of NPR fuels prior to quenching. The probability of meeting specifications on the temperature at quench was computed, and a report issued.

Two three-by-three factorial experiments, each with two replications, were run to obtain further information on the effects of canning cycles on fuel quality, as measured by the bond testers and by the stud pull test. One used lead plugs, and the other, standard cores. The data were analyzed, and a report issued.

Several surveys contributing to a better understanding of fuel element failures are being made by FPD personnel. At their request, a critique is being made of these surveys in order to attach objective probability levels to the inferences drawn, and, more important, to alert interested personnel to the pitfalls existing in analyses in which hypotheses are formulated largely from the data being used to test these hypotheses.

Wall thickness data from NPR extrusions are being examined to determine how best to analyze these data on a routine basis.

Conferences were held with members of Manufacturing Studies, FPD, and Data Processing, to discuss and evaluate the results of trial runs of the linear programmed manufacturing forecasting model. These trials, run on the SHARE Program LP90, proved to be very satisfactory.

Further discussions were held, and additional work has been done on the manufacturing scheduling model, and the first trial run is in the process of preparation for the computer.

The excellent agreement between theory and experimental results on the behavior of Lamb waves in slabs of materials has prompted a request for mathematical assistance on a similar theory for cylindrical geometry. The behavior of Lamb waves at ultrasonic frequencies plays a major role in one method of nondestructive testing.

Irradiation Processing Department

Reactor Optimization Studies

A rough draft report was issued summarizing results to date in the study concerned with objectively controlling certain work rates within IPD. Suggestions were offered relative to how some of the excessive variations existent in past data might be reduced, and IPD personnel are at present following these suggestions by obtaining additional data of a different kind.

With respect to the computer simulation of reactor operations, considerable progress was made in arranging for necessary data to be put in a form amenable

to analysis. These data relate to radiation exposures, craft-hour expenditures, and work assignment distributions. In addition to processing past data, recommendations have been given relative to progressively updating these data in a form that permits analysis. Basic FORTRAN packets to standardize the files and categorize the information for analysis are currently in progress.

Process Tube Leaks

As a part of the problem concerned with obtaining a better model to describe internal process tube corrosion, attention is being given methods of processing R data, indicative of nonuniformity of temperature distribution circumferentially around a tube. A rough draft report was issued suggesting how such data should be processed to best characterize this nonuniformity for any given tube.

As requested, a critique was made of a rough draft report summarizing sources of error in interpreting probolog traces. This report was based on an analysis performed by the Operations Research and Synthesis Operation.

General

Two analyses were completed relative to C-basin data validity. One was performed on data in which a number of standards were measured several times both on 300 Area and on C-basin equipment. The MERCY program was utilized in this analysis. The other was a re-evaluation of postirradiation measurements from the nickel-plated production test which afforded an opportunity to assess the effects of cleaning on the various measurements. This analysis was done in order to determine if it were ever permissible to omit the cleaning step should the occasion warrant it.

Work continued on the problem of estimating the probability of detecting cracks in welded primary piping for the NPR Project.

The differences between calculated and chemically measured burnup values were studied for experimental metallic fuels irradiated over the past 2 1/2 years and an evaluation of the results published as Report HW-72005. Twenty-nine burnup results from fuels irradiated in Hanford test loops were reviewed. The average heat of fission value which yielded the best over-all agreement between the results was 197 $\pm$ 10 Mev, where the limits represent one standard deviation from the average value. The sources of error in the data were determined in so far as possible and included in the discussion.

Chemical Processing Department

Machining Development

Work continues on the EDPM program to produce an accurate and reliable magnetic control tape for the Gorton lathe.

General

To permit the use of analytical measurement data for both process control and demonstration of satisfactory product purity, the use of 20 casting samples per week with replicate measurements will be adopted for the quarter beginning January 1, 1962. The process parameters will be reviewed each week by process technology personnel. This eliminates the long delay between the casting step and the analysis step.

The feasibility of extending the above sampling program to other process control characteristics, such as density, is being reviewed.

Work continued on a motor pool problem for the Facilities Engineering Operation. A digital computer program to determine the minimum cost reallocation of cars has been written and submitted to EDPO.

Relations Operation

General

In connection with the blood count study being performed for the Occupational Medical Operation, a model was tested which included the variables age, time of examination, area, radiation exposure and sex using a population of the exposed personnel and the controls. While radiation was not a significant factor, ignoring both radiation exposure and area increased the residuals significantly. This situation arose because the sample itself promoted a strong relationship between exposure and area - there were few unexposed people in the 100 Areas and few exposed people in the 700 and 1100 Areas. Further tests using only 100 Area data showed exposure to be insignificant, but the small control population here made the test relatively insensitive.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HLO

2000 Program

Pulse Column Test Facilities

The calibration functions for the 1CF- and 1CX-rotometers and the midcolumn photometer have been programmed into the IBM data reduction system. Estimates have been made of the precision with which flow rate and aqueous uranium concentration can be estimated using these calibration functions and experimental data. The absorptiometer calibration, with the original four test cells, was also completed. Data from a second set of four test cells are currently being processed.

Further consideration was given to the modification of the NELLY nonlinear least squares routine so that it could process more than a single dependent variable. The data from which the column mass transfer dynamics will be estimated with the aid of NELLY includes both aqueous and organic concentration estimates as dependent variables, hence the need for the modification.

Physical Properties Testing

A statistical analysis was completed of a set of Zircaloy II hardness data for Physical Metallurgy Operation. Hardness was correlated with percent cold work and the relative magnitudes of within sample, between sample within plate, and between plate hardness variability were estimated. This information will provide a standard for the evaluation of hardness tests on irradiated Zircaloy II.

The revised, multifeatured, EDPM program used in design studies for a proposed neutron flux monitor based on the Phoenix fuel concept, was completed and turned over to Instrument Research and Development Operation.

Several discussions were held with Chemical Research and Development concerning the sampling procedure of a forthcoming deionized water experiment in a KE test loop.

4000 Program

Swelling Studies

Further work was done on programming of statistical procedures for analyzing micrograph pore size distributions. The programming is complete for the first stage of the statistical analysis; i.e., unbiased estimation of the moments of the theoretical distribution in the uranium matrix from the observed distribution on the micrograph. The next stage of the analysis, which is currently being programmed, is the estimation of the functional form of the theoretical pore size distribution in the matrix using the moment information.

Plutonium Recycle

A review was initiated of the D_2O inventory procedure for PRTR. Propagation of error techniques are being used to estimate the total amount of error in a cold D_2O inventory due exclusively to instrument imprecision.

Standard statistical analyses are being run on data obtained from several screening tests of particulate fuel element mixtures. These mixtures are known to produce high density fuels under certain conditions, and it is desirable to ascertain the size and mixture ratios of their constituents.

Further work was done on the mathematics of a statistical procedure for quantitative estimation of a mixture of pure death processes by the method of minimum chi-square.

6000 Program

Biology and Medicine

Work continued on the problem of fitting curves to data from an experiment, conducted by the Biology Operation, to determine the effects on longevity of mice exposed to an aerosol containing radioactive particles.

At the request of the Biology Operation, work was begun on statistical analysis of data from an experiment involving rats subjected to a partial hepatectomy.

Consultation was provided to interested personnel of the Biology Operation on a paper concerned with Cs^{137} in milk. Further checking of calculations is needed before submission of the paper for publication.

General

A method of constructing an equal-area grid on the interior surface of a metallic hemispherical shell was devised. The problem arose in connection with an instrument being constructed to measure the field of available vision through various types of face masks.

Other

Atmospheric Diffusion Studies

APE diffusion data from three towers are being used to screen candidate mathematical models for vertical concentration profiles. A set of the most promising models will be fitted to all tower data.

Instrumentation

An IBM program was written to estimate the precision of multichannel gamma energy analysis quantitative isotope estimates within a deterministic mathematical framework; i.e., when k isotopes are estimated from the information in k peak region channel groups. The program is currently being debugged with several test spectra.

Radiation Protection Studies

A report on the dose-film density relationship was forwarded to the External Dosimetry Operation for comment. Work continued on the correlation of badge and pencil data.

Carl C. Burnett

Manager
Operations Research and Synthesis

CAB:dgl

PROGRAMMING OPERATION

DECEMBER, 1961

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Cross Section Evaluation

The cross section evaluation work carried on concurrently with the Physics groups has been examined and indicates the need for further work to refine the cross section model of the MELEAGER Burn-Up Codes. Expansion and refinement of this work will be done by the Physics groups with assistance by Programming to evaluate the impact of the physics improvements on fuel cycle costs and on plutonium values. Past experience indicates that a change in the physics model may or may not seriously affect plutonium values.

The cross section work to date has been concentrated on developing a relationship to rapidly supply the most probable neutron temperature which is necessary to properly use the Westcott cross sections used by the MELEAGER code. The following proposed relationship has been developed and involves the 2200 meter-per-second absorption cross sections, isotope concentrations, cell averaged slowing-down power (moderator effectiveness), the moderator temperature, and a constant labeled "C". This expression has been updated to the form shown in Equation 1.

$$T_n = T_m \left[1 + \frac{\sum_i C_i N_i \sigma_{aoi}}{\xi \Sigma_s \sqrt{\frac{T_m}{293}}} \right] \qquad (1)$$

T_n = Neutron temperature ^oK

T_m = Moderation temperature ^oK

C_i = A constant derived for the "i"th isotope

σ_{aoi} = 2200 meter-per-second absorption cross section

$\xi \Sigma_s$ = Cell averaged slowing-down power

Using Equation (1), neutron temperatures were computed by a series of computer runs for a typical mixture of plutonium isotopes in U-238 (tails composition) and are compared with results computed by the SPECTRUM code in Table I.

It is planned to adopt an expression of this type (Equation (1)) within the MELEAGER code subroutine which computes effective nuclear cross sections. At this same time, additional alterations may be made with respect to selection of other options involved in the Westcott formulations.

TABLE I

ERROR ANALYSIS OF THE NEUTRON TEMPERATURE EQUATION FOR A FUEL MIXTURE OF PLUTONIUM IN TAILS UO_2 AT FULL DENSITY

(Plutonium had a Pu-239 to Pu-240 ratio of 80 to 20.)

SDPV*	Percent Fissile Included 0.4 Percent (U-235)	Computed Neutron Temperature From Equation(1) °K	Computed Neutron Temperature From Spectrum °K	Percent Error °K
1.5	0.4	494	494	0.0
1.5	1.0	513	516	-0.6
1.5	2.0	545	547	-0.4
1.5	3.0	578	575	0.5
1.5	5.0	642	625	2.7
2.0	0.4	488	489	-0.2
2.0	1.0	503	505	-0.4
2.0	2.0	527	531	-0.8
2.0	3.0	551	552	-0.2
2.0	5.0	600	592	1.4
3.0	0.4	483	483	0.0
3.0	1.0	493	494	-0.2
3.0	2.0	509	512	-0.6
3.0	3.0	525	530	-0.9
3.0	5.0	558	556	0.4

* An approximation employed to account for different fuel-to-moderator ratios.

$$SDPV \cong \xi \Sigma_s \left(\frac{\text{Volume Moderator}}{\text{Volume Fuel}} \right) \qquad (2)$$

Zoned Spectrum Reactor

Some reactivity lifetime computations were completed for the Zoned Spectrum Reactor concept in which there are different amounts of moderator in various zones so that excess neutrons will be absorbed by fertile fuel rather than by control systems. As previously indicated, such reactors are similar to the Edlund Spectral Shift Reactor under study by the Babcock & Wilcox Company wherein H_2O is introduced in D_2O to adjust the moderating power. Like the Edlund reactor, the Zoned Spectrum reactor appears to improve the reactivity limited fuel lifetime from 50 to 100 percent over conventional batch cycle techniques. This gain is essentially the same as that achieved by the continuous or graded cycles in which the freshest fuel is put in appropriate proximity of the highest irradiated fuel.

Preliminary results of the study indicate that a U-235:U-238 fueled D_2O moderated zoned spectrum reactor has slightly higher fuel costs than a reasonably comparable D_2O moderator using the conventional graded cycle. However, to be added is the effect of the variation in fuel power throughout the irradiation for the graded system. This does not exist for the zoned spectrum reactor. The computer codes are being further debugged to allow study of plutonium or U-233 enrichment with thorium as well as with U-238 fertile materials.

Salt Cycle Reprocessing Economics

The Salt Cycle computer program work again this month involved efforts to develop and debug the material balance routine for the Salt Cycle economics code. A general outline of the procedures being developed was described last month. The difficulties with the program result from the trial and error type computation involved. There are numerous conditional statements in the program which require much time consuming and tedious checking and debugging.

SPECIFIC FUEL CYCLES

Plutonium Value Computations

The major effort during December was directed toward performing calculations for a report of the computed indifference values of recycled plutonium in five reactor characterizations. These characterizations approximately simulate the APWR, the BWR, the OMR, the D_2O moderated, and the Gas Cooled types of reactors.

Values were computed by use of the MELEAGER CHAIN code of Hanford Laboratories Operation and substantially agree with the results of the MELEAGER CHAIN plutonium value studies completed in August, 1961.

The variations studied included Batch and Graded fuel cycles, AEC and private owndership of fuel, and premium costs for plutonium fuel processing in zoned and unzoned reactors. The solutions neither account for exchange of fuel materials between reactors nor for over-all power costs of the simulated reactors. For these reasons, high plutonium values do not necessarily imply lower power costs nor do low plutonium values imply higher power costs. With the post-July 1, 1961, uranium price schedule, the computed plutonium values for the first recycled plutonium vary between \$12.50 and \$9.50 per gram (fissile) for the simulated reactors under the conditions studied except for the OMR. The OMR neutron spectrum posed a special problem for the physics models employed and the computed value for the first recycled plutonium appears to lie between \$7.50 and \$11.80 per gram (fissile) in the OMR. For all reactors and all conditions studied, after five recycles the values decrease to a high of \$10 and to a low of \$4 per gram (fissile), a noticeable increase in spread.

The computations also included an error analysis to show the effects of the various parameters on plutonium values. This consists of extensive variations in the appropriate physics and economics inputs to the codes used. The results reflect more insensitivity than generally anticipated.

The consistency of the results was improved over those developed in August, 1961, by adding two recycle steps to each recycle series and by using the plutonium values calculated during August for the initial estimate needed for the minimized fuel cost calculations.

The importance of these factors was shown by the following experiment where the values obtained from the August plutonium studies with five recycle steps are compared with alternate seven-step iterations. Different initial estimates were used for the two seven-step iterations as noted in Table II as alternates A and B.

TABLE II

CALCULATED PLUTONIUM VALUES USING DIFFERENT INITIAL VALUE ESTIMATES

Step Number	August Value	Alternate Iteration A		Alternate Iteration B	
		Estimated	Calculated	Estimated	Calculated
1	13.10	12.35	12.32	10.83	12.10
2	12.10	11.85	11.60	10.62	11.52
3	10.90	11.26	11.28	10.30	10.80
4	8.85	10.25	10.92	9.75	10.86
5	7.25	9.45	9.90	8.89	9.92
6		8.64	8.85	7.95	9.02
7		7.90	7.91	6.90	8.20

Note: All values are in $/gram (fissile).

The addition of two steps was quite important so that consistent results could be obtained, as can be seen by examination of the data in tables that are similar to Table II but in which the sixth and seventh steps have been deleted for the value computation.

The effect of the initial estimates used in Table II is summarized in Table III where the percent difference in the calculated value is shown as a function of the percent difference in the estimate for each step.

Tables II and III indicate a difference between the August, 1961, values and those resulting from the iterations. One more iteration is planned to ascertain whether an additional step is worthwhile.

TABLE III

PERCENT DIFFERENCES OF ESTIMATES SHOWN IN TABLE II

Number	Percent Difference Between Estimates A and B	Percent Difference Between Resulting Calculated Values
1	12.3	1.8
2	10.4	0.7
3	8.5	4.2
4	4.9	0.6
5	5.9	-0.2
6	8.0	-1.9
7	12.7	-3.7

Note: The reversal in Steps 5, 6 and 7.

Estimate for Cost of Producing Mercury-204

The report on the cost of producing pure mercury-204 isotope by the monoisotopic photosensitization process was completed and issued. Details of the results of the study were covered in the last monthly report. A figure of about $50/pound was obtained. The unresolved problem of ultimate and effective removal of the very high cross section mercury-199 was recognized along with the possibilities of even lower cost production of mercury-202 which in some applications may be only slightly less suitable than mercury-204. Major cost reductions appear achievable only by major increases in plant capacity since 70 percent of the total cost relates to charges for fixed expenses including operating labor. At $50/pound some specialized reactor applications for mercury-204 as a boiling heat transfer agent may be anticipated.

OTHER ACTIVITIES

Alternate Solvent Extraction Techniques

A preliminary write-up of an extraction process which would utilize a volatile solvent with essentially no salting agent was prepared. Study modifications of such an alternate was continued.

Hazards Analysis

Work neared completion on the estimation of potential damages which could result from a severe accident during shipment of 500,000 c of cerium-144. A preliminary draft report, HW-72163, "The Consequences of Accidental Releases During Shipment of Radioactive Cerium," was completed in conjunction with others. Some modifications of the report are known to be necessary in light of more recent data on volatility of cerium and other fission product elements present as impurities.

W.K. Woods

Manager,
Programming

WK Woods: jm

RADIATION PROTECTION OPERATION REPORT FOR THE MONTH OF DECEMBER 1961

A. ORGANIZATION AND PERSONNEL

Effective December 1, 1961, G. W. R. (Bill) Endres was transferred from the Nuclear Physics Research Operation to Radiological Development and Calibrations. M. H. (Kathy) Davis was reactivated on December 6, 1961, replacing Alice L. Didier of Internal Dosimetry who transferred to the Instrument Research and Development Operation on December 11, 1961. R. B. (Bob) Hall was transferred from the Irradiation Processing Department to Environmental Studies and Evaluation effective December 15, 1961. Also transferring from the Irradiation Processing Department was Ray L. Pierce who joined the External Dosimetry group effective December 15, 1961. Voda S. O'Bannon retired from the General Electric Company effective December 29, 1961. The work force now totals 133.

B. ACTIVITIES

Occupational Exposure Experience

There were six cases of plutonium deposition confirmed by bioassay analyses during the month. The total number of plutonium deposition cases that have occurred at Hanford is 283 of which 205 are currently employed. Five of the six plutonium deposition cases were the result of previously reported radiation occurrences involving CPD employees. Inhalation was the mode of deposition for four of the cases and one was the result of a contaminated minor injury. The sixth deposition case was detected by the routine bioassay program. The maximum deposition observed was less than ten percent of the maximum permissible body burden. At the end of calendar year 1961 there were 19 confirmed plutonium deposition cases, all of which were estimated to be less than ten percent of the maximum permissible body burden.

A CPD maintenance employee received a plutonium contaminated puncture wound in the 234-5 Building while repairing electrical leads in a recuplex hood. Examination of the wound site in the Whole Body Counter showed 5.7×10^{-3} μc plutonium. No contamination was detected after an Industrial physician excised a small piece of tissue from the wound site. Analysis of the excised tissue showed a plutonium content of 5.8×10^{-3} μc plutonium.

An IPD maintenance employee received face and nasal contamination to 500 c/m while performing grinding operations on contaminated material without respiratory protection. Examination in the Whole Body Counter showed a body burden of 8×10^{-3} μc of scandium-46 and 4×10^{-2} μc of zinc-65. The measured body burdens represented only a small fraction of one percent of the maximum permissible body burden for these radionuclides.

Removal of wet storage basin filters in the 327 Building necessitated work in radiation fields up to 160 rads/hour. Rigid radiation controls prevented any unusual occurrences.

Considerable effort was required at PRTR in the maintenance of the pri- pumps and nozzle caps. Hand dose rates to 25 rads/hour and body dose rat 50 mrads/hour were experienced during maintenance on the primary pumps.

Routine water samples from the moderator, reflector, and primary coolant systems with the reactor operating at 70 megawatts posed dose rates to 530 mrads/hour, 2 rads/hour, and 125 mrads/hour, respectively. The maximum tritiu bioassay sample observed for PRTR operational personnel was 14 μc T/liter. Th tritium concentration in the moderator, reflector, and primary systems was 48(μc T/ml, 283 μc T/ml, and 53.5 μc T/ml, respectively.

Air samples continued to be high during the cleaning of pens housing swine receiving 3.1 millicuries of strontium-90 per day. The maximum air concentrat noted was 1.6×10^{-8} μc Sr^{90}/cc of air. Respiratory protection is required du ing cleaning operations.

Environmental Experience

The average concentration of air-borne radioactive materials in the Pacifi Northwest diminished to about one-half that observed in November and was in th range of 3 to 10 μμc/m^3 at various collection sites.

A total of 55 fish were taken from sampling locations in the Columbia Rive at Priest Rapids, Hanford, and Ringold. Eighty-eight tissue samples from thes fish were submitted to the laboratory for radiochemical analyses. One hundred fifty-four tissue samples from 77 ducks taken at selected locations on the Han ford reservation were submitted to the laboratory for radiochemical analys This concluded the local waterfowl sampling program for this season. Eleven hundred twenty-four duck and goose heads were received from various locations in Washington, Idaho, Oregon, and California. The total number received and submitted for radiochemical analyses to date is 3180. This special program carried out in cooperation with the Radioecology Operation is now virtually co pleted.

An expanded cattle thyroid sampling program is now in progress. All parti cipating veterinarians have been visited, contracts signed, and shipping con- tainers and instructions provided. The veterinarians have expressed interest in the program and have been cooperative in providing the necessary samples.

Automatic sampling equipment for continuously monitoring I^{131} and radioact particulates in the PRTR stack was installed. The calibration of the air flow rate through the fixed filter and the iodine sampling section of the apparatus was completed. Source calibrations were completed for the iodine sampler. Th sampler has a linear response over the first four decades but will require a calibration curve for the final decade. The performance of this system with Ba^{133} sources and with I^{131} sources showed close agreement; therefore, Ba^{133} sources will be used as permanent calibration standards.

Studies and Improvements

The Columbia River Emergency Plan, a detailed plan for action to be taken in the event significant contamination is detected in the Columbia River, approved.

The distribution of Emergency Radiation Monitoring Kits to all selected field locations was completed. Supporting high-range instruments and dosimeters are stored at several additional locations to provide backup supplies and emergency equipment.

A report showing the location of criticality detectors, alarms, and dosimeters was revised to reflect corrections and revisions submitted by field Radiation Monitoring personnel. Work was initiated on a similar report which will show the location of: (1) air sampling stations, (2) fixed monitoring devices such as HM chambers and Kanne chambers, (3) recorders for fixed monitoring chambers, and (4) the direction of air movement within the buildings.

Work continued on the CPD Hazard Analysis. Modifications of the objectives of the report to include not only the worst conceivable accident but also a more likely accident resulted in additional evaluations. On the basis of whole body dose, the worst conceivable accident was judged to be an accident involving the Purex dissolver. A more likely accident was judged to be an accident involving transportation of irradiated fuel elements.

Several improvements were made in the rubber finger ring dosimeter. The aluminum light shield was replaced with a plastic light shield, thereby increasing the sensitivity of this dosimeter by a factor of two. More flexible rubber was used to mold the dosimeter body. This should result in a reduced number of lost readings due to light leaks. The identification numbers are clearly imprinted on the new plastic shields. These dosimeters will be placed into routine service on January 1, 1962.

A study of neutron attenuation as a function of borated polyethylene thickness was made. Both dose and flux attenuation measurements through various thicknesses of borated and plain polyethylene were completed. The measured attenuations were compared with calculated values and found to be in agreement. Calculations were then made to determine the amount of shielding necessary to reduce the neutron dose rates to an acceptable level around the plutonium metal storage hoods in the 234-5 Building. To reduce the dose rate to 1.5 mrem/hour would require 4 inches of Fe or 7-1/2 inches of polyethylene of which the first and last half-inch would be borated.

The successful bidder for fabrication of the new Hanford personnel film dosimeter is Product Engineering Company of Portland, Oregon, with a bid of $1.61 per dosimeter.

The design of the new Hanford personnel film dosimeter processing machine is about 75% complete. Construction prints for fabrication of the processing machine are being obtained from Drafting. All pneumatic controls, air cylinders, turntable, X-ray head and timers for the machine were ordered.

A study has indicated that the present film dosimeter processing machine can be converted for processing the standard Hanford personnel neutron dosimeter. The machine can be programmed to remove both the neutron and gamma film packets from this dosimeter.

The sensitive scintillation type radiation measuring instrument was exposed at selected locations to a one curie Co^{60} source at distances of 500 and 350 feet to study the possibility of providing a general 300 Area radiation surveillance system. The sensitivity of the instrument was such that it was possible to measure a dose rate from this source at distances in excess of 500 feet in air. The 300 Area could be "surveyed" by three of these instruments appropriately located, one each, in the 328 Building, 326 Building, and 314 Building. The desirability and cost of providing this radiation surveillance is being studied.

The use of a Perlow or similar spectrometer to measure neutron spectra at Hanford work locations was studied. Previous measurements with a Perlow instrument indicated resolution of only 12 to 20%. It appears that the resolution can be improved by narrowing the acceptance angle of the collimators of the spectrometer.

Bids were received for the Scintran and the BF_3 meters. The Instrument Laboratory, Inc., of Seattle was the low bidder on both instruments with a bid of $30,868 for 50 Scintrans and $8,833.50 for 30 BF_3 meters.

Portable instruments repair billing costs were reviewed and updated during the month. New costs were based on the previous year's portable repair shop actual costs. New charges will be effective beginning January 1, 1962.

C. VISITORS AND VISITS

The following visitors met with various members of the Radiation Protection staff during the month:

Dr. W. F. Scott)
Mr. David Rahm) - Washington State University, Pullman, Washington

D. Wagstaff - Oregon Board of Health, Salem, Oregon

Janakiram Naidu - Atomic Energy Authority of India, Bombay, India

Members of the Radiation Protection Operation visiting off-site during the month included:

C. M. Unruh - Oak Ridge, Tennessee, to attend annual Industrial Health Personnel meeting.
- Oak Ridge National Laboratory, Oak Ridge, Tennessee, to discuss radiation dosimetry and observe radiation protection practices and criticality monitoring equipment.

R. D. Tillson - New Mexico State, Las Cruces, New Mexico, technical recruiting.
- Texas Western, El Paso, Texas, technical recruiting.
- University of New Mexico, Albuquerque, New Mexico, technical recruiting.
- Texas Tech., Lubbock, Texas, technical recruiting.

D. RELATIONS

Four suggestions were submitted by the personnel of the Radiation Protection Operation during the month of December. A total of fifty-five suggestions were submitted during the year. Five suggestions were adopted and six were rejected. Nine suggestions submitted by RPO personnel are pending evaluation.

There were two medical treatment injuries during the month for a frequency of 0.88. No security violations occurred during the month of December.

Radiation protection orientation included lectures to: Local 370 - International Union of Operating Engineers; personnel of Engineering Auxiliaries, Biology Research, Chemical Effluents Technology, Electrical Distribution, and Transportation.

Two Civil Defense talks were presented at Plant information meetings.

A discussion of Emergency Plans at HAPO was presented to personnel in the Technical Administration Operation.

Marked improvement in safety and housekeeping standards throughout the Radiation Protection Operation has been noted. Safety meetings were held throughout the various Subsections during the month.

E. SIGNIFICANT REPORTS

HW-71203 - "Evaluation of Radiological Conditions in the Vicinity of Hanford, July-September, 1961" by Environmental Studies and Evaluation Staff.

HW-71971 - "Radioactive Contamination in Liquid Wastes Discharged to Ground at the Separations Facilities through June 1961" by G. E. Backman.

HW-72012 - "Summary of Radiological Data for the Month of November, 1961" by R. F. Foster.

HW-72155 - "Monthly Report - December 1961, Radiation Monitoring Operation" by A. J. Stevens.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

External Exposures Above Permissible Limits

	Dec.	1961 to Date
Whole Body Penetrating	0	0
Whole Body Skin	0	1
Extremity	0	0

Hanford Pocket Dosimeters

Dosimeters Processed	2,622	45,902
Paired Results - 100-280 mr	10	432
Paired Results - Over 280 mr	0	21
Lost Results	0	0

Hanford Beta-Gamma Film Badge Dosimeters

Film Processed	10,219	133,676
Results - 100-300 mrads	1,038	12,759
Results - 300-500 mrads	105	1,790
Results - Over 500 mrads	23	322
Lost Results	43	430
Average Dose Per Film Packet - mrad (ow)	8.38	8.67
- mr (s)	23.24	23.69

Hanford Neutron Film Badge Dosimeters

Slow Neutron

Film Processed	845	19,285
Results - 50-100 mrem	0	1
Results - 100-300 mrem	0	0
Results - Over 300 mrem	0	0
Lost Results	3	99

Fast Neutron

Film Processed	230	5,104
Results - 50-100 mrem	35	495
Results - 100-300 mrem	18	271
Results - Over 300 mrem	0	0
Lost Results	3	99

Hand Checks

Checks Taken - Alpha	28,365	392,739
- Beta-gamma	41,987	562,943

Skin Contamination

Plutonium	16	350
Fission Products	27	514
Uranium	0	47
Tritium	0	20

Whole Body Counter	Male	Female	Dec.	1961 to Date
GE Employees				
Routine	32	4	36	643
Special	2	0	2	63
Terminal	19	0	19	130
Non-routine	11	2	13	36
Non-employees	2	1	3	77
Pre-employment	0	0	0	38
	66	7	73	987

Bioassay

Confirmed Plutonium Deposition Cases	6	19*
Plutonium - Samples Assayed	314	5,663
- Results Above 2.2×10^{-8} µcPu/Sample	5	133
Fission Products - Samples Assayed	590	6,455
- Results Above 3.1×10^{-5} µcFP/Sample	0	18
Uranium - Samples Assayed	135	2,759
Biological - Samples Assayed	0	196

Uranium Analyses

Sample Description	Following Exposure Units of 10^{-9} µc U/cc Maximum	Average	Number Samples	Following Period of No. Exposure Units of 10^{-9} µc U/cc Maximum	Average	Number Samples
Fuels Preparation	74.8	5.5	36	15.1	3.0	34
Fuels Preparation**	0	0	0	0	0	0
Hanford Laboratories	13.7	3.6	24	17.2	3.9	28
Hanford Laboratories**	0	0	0	0	0	0
Chemical Processing	8.4	4.3	3	10.2	7.3	2
Chemical Processing**	8.6	6.5	2	165.9	84.4	2
Special Incidents	0	0	0	0	0	0
Random	1.6	1.5	4	0	0	0

Tritium Samples	Maximum	Minimum	Count	Dec. Total
Urine Samples				
Routine	4.9 µc/l	<1.0 µc/l	87	
Samples Above 5.0 uc/l	14.0 µc/l	5.0 µc/l	32	
				119
D^2O Samples				
Moderator	480.0 µc/ml	394.0 µc/ml	4	
Primary	53.5 µc/ml	49.7 µc/ml	4	
Reflector	283.0 µc/ml	251.0 µc/ml	4	
				12
Water Samples	0.05 µc/ml		14	14
				145

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* The total number of plutonium deposition cases which have occurred at Hanford is now 283, of which 205 are currently employed.

** Samples taken prior to and after a specific job during work week.

Calibrations

	Number of Units Calibrated	
	Dec.	1961 to Date
Portable Instruments		
CP Meter	844	11,037
Juno	247	2,956
GM	504	6,290
Other	176	2,014
Audits	100	1,247
	1,871	23,544
Personnel Meters		
Badge Film	1,344	16,487
Pencils	0	7,174
Other	393	4,077
	1,737	27,738
Miscellaneous Special Services	381	9,605
Total Number of Calibrations	3,989	60,887

for Carl M. Unruh
Manager
RADIATION PROTECTION

AR Keene:CMU:ljw

GENERAL

There was one security violation charged to the Operation.

There were no major injuries; the minor injury frequency rate was 3.05 for the month and 2.93 for the year-to-date.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 20,252 hours. This includes 13,241 hours performed in the Technical Shops, 5,722 hours assigned to Minor Construction, and 1,289 hours assigned to off-site vendors. Total shop backlog is 19,130 hours, of which 60 per cent is required in the current month with the remainder distributed over a three-month period. Overtime hours worked during the month was 3.8 per cent (645 hours) of the total available hours.

Distribution of time was as follows:

	Man-hours	% of Total
Fuels Preparation Department	4,040	19.95%
Irradiation Processing Department	1,012	5.00%
Chemical Processing Department	594	2.93%
Hanford Laboratories Operation	14,588	72.03%
Construction Engineering & Utilities	18	.09%

Requests for emergency service decreased requiring a 3.8 per cent overtime ratio compared to a 6.1 per cent ratio for the previous period.

At the close of the reporting period, there were no open personnel requisitions.

There were nine medical treatment injuries which is within the forecasted parameters established for this Operation.

A seminar on the numerical control of machine tools was held in San Francisco on December 7 and 8 and was attended by the Shop Manager.

CONSTRUCTION OPERATION

There were 76 existing J. A. Jones Company orders at the beginning of the month with a total unexpended balance of $153,848. Two hundred and three new orders, 3 supplements and adjustmants for underruns amounted to $83,628. Expenditures during the month on HLO work were $101,806. Total J. A. Jones backlog at month's end was $135,670.

Summary

	HL		CE&U	
	No.	Unexpended Balance	No.	Unexpended Balance
Orders outstanding beginning of month	75	$ 151,377	1	$ 2,471
Issued during the month (Inc.Sup.& Adj.)	203	83,628		
J.A. Jones Expenditures during month (Inc. C.O. Costs)		101,806		
Balance at month's end	93	135,670	1	2,471
Orders closed during month	185	81,576		

Current Maintenance Work Orders - 4 - Face Value $10,400.

FACILITIES ENGINEERING OPERATION

Project Activity

At month's end Facilities Engineering Operation was responsible for 11 projects having total authorized funds in the amount of $2,722,600. The total estimated cost of these projects is expected to be $7,536,000. Expenditures on these projects through November 30, 1961 were $636,000.

The following summarizes the status of FEO project activity:

Number of authorized projects at month's end 11

Number of new projects authorized in December 0

Projects completed in December 2
- CAH-902 - Uranium Scrap Burning Facility
- CAH-919 - Air Conditioning - 314 Building

New project proposals submitted to AEC in December 1
- CGH-951 - A-C Column Facility - 314 Building

New projects awaiting AEC approval: 3
- CAH-917 - Field Service Center
- CAH-932 - 300 Area Retention Waste Expansion System
- CGH-951 - A-C Column Facility - 314 Building

Project proposals complete or nearing completion are as follows:
- Modifications to the H-1 Loop - 105-H Building
- CAH-949 - Critical Mass Lab. - Stage II
- Graphite Shop Addition - 300 Area
- Second Floor Modifications - 308 Building

Engineering Services

Title	Status
Pressure Vessel and Piping Systems - Engineering & Inspection Service	Gas Loop - Field review by third party inspector. Rupture Loop - Code review in progress. Progress depends on completion of various branches by Contractor. Non-Isothermol Loop - 314 Building Reviewed all data and third party inspected loop.
"Split-Half" Machine for Critical Mass Studies	Design is complete and fabrication will be complete the end of January.
HLO Electrical & Signal Systems	Buildings are routinely metered and records maintained. Lighting level will be increased in lobby of 3760 Building. Central alarm for criticality system is being installed. Standard signals have been applied to all HLO buildings in 300 Area with minor exception.
Monorail Trolley - 326 Building	New insulated electrical conductors will be installed during January.
Room Air Conditioner - Room 21-A 326 Building	Equipment is being installed.
Salt Bath Furnace Alarms - 306 Bldg.	A new alarm system has been designed and procurement has been initiated.
Ceramic Fuels Space - 325 Bldg. Basement	Equipment relocation and utility requirements is being evaluated.

Drafting and Design Services

The work load in the 3706 Building drafting room and in 327, 306, 308, and 1707-D Buildings is steady with a minor amount of overtime required.

The equivalent of 126 design drawings were completed this month as compared to 135 last month.

Major design and drafting work in progress during the month includes the following:

1. Fission Product Packaging Equipment - 43 drawings being reviewed.

2. Shim Rod Drive Mechanism - PRTR - 7 drawings required - 50% complete.

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3. Hood for Furnace - 5 drawings required - 80% complete.

4. TF-23 Test Loop - 10 drawings required - 50% complete.

5. Transistorized Scintillation Monitor - 6 drawings - 75% complete.

6. Relocation of 280 Ton Pressure - 12 drawings required - 75% complete.

7. Welding Equipment for 8 foot Element - 7 drawings required - 60% complete.

In addition to the above, several design jobs of one to two sheet magnitude were completed during the month.

Plant Maintenance Operation

Total costs for November were $148,445. Fiscal year to date costs are 99% of forecast.

Analysis of Costs

Maintenance costs were controlled sufficiently this month to bring the total expenditures to date about $7,500 under the forecast. The utility costs for the month were about 10% less than forecast.

Analysis of Improvement Maintenance

Item	November
Relocation and alteration	$ 8,309
Repainting	1,755
Heat & Vent Modifications	2,347
Electrical Modifications	289
	$ 12,700

Significant Activities

1. Painting was confined to minor touch-up work.
2. The contract for relocating TC-1 was negotiated.
3. The alloy storage room in 306 was essentially completed.
4. The arc melt furnace room exhaust improvement job has reached the stage where shop fabrication is complete and installation is beginning.
5. The restoration of the insulation to the walls and underside of the roof in 3718 was completed.
6. The modification of the blue change room in 309 was better than 50% complete.
7. The rearrangement of office space in the 309 M&M wing was completed.

Waste Disposal and Decontamination Service

Quantities of Waste Removed

	November	October
Concrete Barrels	23	12
Loadluggers-Hot Waste	4	4
Milk Pails	41	32
Gattling Gun	1	0
Crib Waste	400,000 gal.	245,000 gal.

A work order has been issued to install a continuous sampler in the retention influent line.

The construction of the new 300-North Area disposal garden has been contracted, and work is scheduled to begin in January.

The study of the decontamination and high level waste disposal facilities is continuing.

Plant Engineering and Miscellaneous

Approximately 21,000 square feet of prints were reproduced during the month.

The total estimated value of the sixteen requisitions issued during the month was $10,000. This procurement is primarily for approved HL projects.

Plans were completed to improve the compressed air service in 209-E Building. Procurement and installation of a new separate compressor will be required.

A design is complete for a constant proportional sampler system for the 307 influent line. Field work will start during January. Also preliminary design of automatic valve application to 307 lines is continuing.

Recommendations have been made to Biology Operation for 141-M Building heating improvements. A continued study will be made to develop sequence of field action.

The semi-annual five year forecast of electrical power requirements was completed.

Design and purchase specifications were completed for an addition to the motor control center of Critical Mass Laboratory, 209-E Building.

Plans are being developed for laboratory changes in 3706 and 321-A Buildings.

Space rearrangement in the basement of 325 Building and installation of equipment may require supplemental air conditioning equipment. Studies are being made to ascertain the proper amount and method of application.

J L Boyd

Manager
Laboratory Auxiliaries

JL Boyd:jw

SEMI-MONTHLY PROJECT STATUS REPORT

HW- 72142

GENERAL ELECTRIC CO. — Hanford Laboratories

DATE 12-31-61

PROJ. NO.	TITLE	FUNDING
CAH-822	Pressurized Gas Cooled Facility	4141 Operating

AUTHORIZED FUNDS				
$ 1,120,000	DESIGN $ 40,000	AEC $ 0	COST & COMM. TO 12-3-61	$ 883,993
	CONST. $ 1,080,000	GE $ 1,120,000	ESTIMATED TOTAL COST	$ 1,170,000*

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 8-19-59	DATE AUTHORIZED 2-3-61		DESIGN 4-29-60	
CONST. 10-17-60	DIR. COMP. DATE 9-30-61		CONST. 5-1-62	

ENGINEER

REDO - 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	100	100	100
GE-TIT. II			
AE-TIT. II			
CONST.	100	100	93
PF			
CPFF	17	100	91
FP	7	100	100
Govt. Eq.	76	100	92

SCOPE, PURPOSE, STATUS & PROGRESS

*A project proposal revision has been submitted which requests an increase of authorized funds of $50,000 and extension of the project completion date to 6-30-62 to accomplish replacement of the NaK heater.

Instrument calibration and design tests are in progress by plant forces.

The in-reactor section prototype has been mounted in B cell for testing and insulation is in progress.

PROJ. NO.	TITLE	FUNDING
CAH-842	Critical Reactivity Measuring Facility	58-e-15

AUTHORIZED FUNDS				
$ 400,000	DESIGN $ 45,000	AEC $	COST & COMM. TO 12-3-61	$ 178,972 (GE)
	CONST. $ 355,000	GE $	ESTIMATED TOTAL COST	$ 400,000

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 11-17-59	DATE AUTHORIZED		DESIGN 2-1-61	
CONST. 10-3-60	DIR. COMP. DATE 3-15-62		CONST. 3-15-62	

ENGINEER

REDO - WS Kelly

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	99	99
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100	88	88*
PF			
CPFF	60	80	80*
FP	40	100	100

SCOPE, PURPOSE, STATUS & PROGRESS

The revised project proposal requesting additional funds and an extension of the scheduled completion date has been approved by the AEC. A new schedule is being submitted which reflects the changes in percent complete.

*Percentages reflect changes resulting from the approval for expenditure of additional funds.

CPFF percentages includes Government Furnished equipment.

H-3

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 72142

DATE 12-31-61

PROJ. NO.	TITLE	FUNDING
CGH-857	Physical & Mechanical Properties Testing Cell - 327 Bldg.	0290

AUTHORIZED FUNDS					
$ 460,000	DESIGN $ 45,000	AEC $ -	COST & COMM. TO 12-17-61	$ 84,885
	CONST. $ 415,000	GE $ 460,000	ESTIMATED TOTAL COST	$ 460,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
	DESIGN 11-2-59	9-22-61	DESIGN 3-15-61	
	CONST. 5-15-62	DIR. COMP. DATE 12-15-62	CONST 12-15-62	

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION	2	707
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		
PF			
CPFF			
FP			

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.

Requisitioning is now about 50% complete. A new requisition for the cell has been forwarded to Purchasing.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS				
$	DESIGN $	AEC $	COST & COMM. TO	$
	CONST. $	GE $	ESTIMATED TOTAL COST	$

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
	DESIGN		DESIGN	
	CONST.	DIR. COMP. DATE	CONST.	

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 72142

DATE 12-31-61

PROJ.NO.	TITLE	FUNDING
CGH-858	High Level Utility Cell - 327 Building	0290

AUTHORIZED FUNDS					
$ 400,000	DESIGN $ 50,000	AEC $	COST & COMM. TO 12-17-61	$	339,494
	CONST. $ 350,000	GE $ 400,000	ESTIMATED TOTAL COST	$	400,000

STARTING DATES			EST'D. COMPL. DATES	
DESIGN	11-1-59	DATE AUTHORIZED 4-6-61	DESIGN	2-15-61
CONST.	5-15-61	DIR. COMP. DATE 2-28-62	CONST.	2-28-62

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		140
PLANT FORCES		
ARCHITECT-ENGINEER		35
DESIGN ENGINEERING OPERATION		930
GE FIELD ENGINEERING		6

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TITLE II	95	100	100
AE-TITLE II			
Vendor	5	100	100
CONST:	100	48	20
PF			
CPFF	100	48	20
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facilities to prepare specimens from irradiated materials for use in determining their physical and mechanical properties and involves the installation of a cell in 327 Building.

Delivery of cell by 1-15-62 is still possible. Plans are being made to continue with construction based on this premise.

Delivery of other components continues to look favorable. No schedule revision is presently planned.

Electrical construction work is proceeding with the installation of motor control center additions, panels and conduit runs. Hole penetrations through the canyon floor for cell services are being completed.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

7214c

DATE 12-31-61

PROJ. NO.	TITLE	FUNDING
CAH-866	Shielded Analytical Laboratory - 325-B Building	61-a-1

AUTHORIZED FUNDS				COST & COMM. TO / ESTIMATED TOTAL COST	
$ 700,000	DESIGN $ 60,000 / CONST. $ 640,000	AEC $ 546,500	GE $ 153,500	12-3-61	$ 99,197 / $ 700,000

STARTING DATES		DATE AUTHORIZED		EST'D. COMPL. DATES	
	DESIGN 9-5-59 / CONST. 6-28-61	DIR. COMP. DATE	5-31-60 / 6-30-62	DESIGN 11-14-60 / CONST. 6-30-62	

ENGINEER

FEO - RW Dascenzo

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE	8	1116
PLANT FORCES	0	3
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING	1	

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	10	100	100
AE-TIT. II	90	100	100
CONST.	100	21	21
PF	3	1	1
CPFF	2	0	0
FP	95	30	20

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.

Exterior metal siding and roof deck failed to arrive as scheduled. It arrived at the site one month late on December 27, 1961.

L&F Machine Company overbored some plug holes in three castings and will have to put in shrink-fitted sleeves.

The painting contractor primed all available material and left the jobsite for a California Christmas. Rejected sheet metal work has delayed the painter.

The electrical sub-contractor has received most of his equipment and should have a full crew on the job soon.

A project construction status chart has been submitted to AEC for approval.

Mr. W. DeWitt of S&E Contractors visited L&F Machine Company to expedite the machining of the cell-castings and establish priorities.

SEMI-MONTHLY PROJECT STATUS REPORT

HW-73142

GENERAL ELECTRIC CO. — Hanford Laboratories

DATE 12-31-61

PROJ. NO.	TITLE	FUNDING
CAH-867	Fuel Element Rupture Test Loop	58-e-15

AUTHORIZED FUNDS					
$ 1,500,000	DESIGN $ 130,000	AEC $ 770,000	COST & COMM. TO 12-3-61	$ 518,131 (GE)
	CONST. $ 1,370,000	GE $ 730,000	ESTIMATED TOTAL COST	$ 1,500,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN	8-1-60	DIR. COMP. DATE 6-30-62	DESIGN	3-15-61
CONST.	11-2-60		CONST.	6-30-62

ENGINEER

REDO - PC Walkup

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	17	1210
COST PLUS FIXED FEE	11	1471
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	9	100	100
CONST.	100	75	67
PF	2	0	0
CPFF	57	95	85
FP (1)	10	100	100
(2)	31	37	26

SCOPE, PURPOSE, STATUS & PROGRESS

(1) G. A. Grant Company
(2) Lewis Hopkins Construction Company

This facility is to be used for fuel rupture behavior studies with respect to physical distortion and rate of fission product release.

Project is behind official schedule because the FP contractors' schedule is not consistent with the official schedule and because of delays in delivery of material being procured by J. A. Jones Construction Company.

PROJ. NO.	TITLE	FUNDING
CAH-888	Biology Laboratory Improvements	60-h-1

AUTHORIZED FUNDS					
$ 420,000	DESIGN $ 44,000	AEC $ 400,000	COST & COMM. TO 11-30-61	$ 396,577
	CONST. $ 376,000	GE $ 20,000	ESTIMATED TOTAL COST	$ 420,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN	8-8-60	DIR. COMP. DATE 3-31-62	DESIGN	3-31-61
CONST.	7-10-61		CONST.	6-15-62

ENGINEER

FEO - JT Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	10	1270
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	NS	100
TITLE I			
GE-TIT. II	17	NS	100
AE-TIT. II	83	NS	100
CONST.	100	77	54
PF	1	30	30
CPFF			
FP	99	76	53

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides additional space for biological research supporting services, and involves an addition to the 108-F Building.

Fabrication of the radiation handling facility has been started.

The windstorm of December 17, 1961 blew down the scaffolding erected for concrete block installation destroying numerous block and scaffolding. A number of sheets of Robertson roof decking were blown off the main roof and probably damaged beyond use. Metal lath partitions are being installed.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW 72142

DATE 12-31-61

PROJ.NO.	TITLE	FUNDING
CAH-896	Stress Rupture Test Facility	60-1

AUTHORIZED FUNDS					
$ 90,000	DESIGN $ 7,500	AEC $ 78,500	COST & COMM. TO 12-3-61	$ 11,616 (GE)
	CONST. $ 82,500	GE $ 11,500	ESTIMATED TOTAL COST	$ 90,000

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 7-29-60	DATE AUTHORIZED 3-6-61		DESIGN 12-1-60	
CONST. 3-20-61	DIR. COMP. DATE 10-15-61		CONST. 11-3-61	

ENGINEER

FEO - H. Radow

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		650
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	100	100
PF	2	100	100
CPFF			
FP	98	100	100

SCOPE, PURPOSE, STATUS & PROGRESS

This project involves a facility for deliberately rupturing tubing to establish service conditions.

The remaining accumulator cap has been received, installed and tested. The only exception is the installation of the replacement space heaters.

PROJ. NO.	TITLE	FUNDING
CAH-902	Uranium Scrap Burning Facility	61-j

AUTHORIZED FUNDS					
$ 36,000	DESIGN $ 5,000	AEC $ 27,500	COST & COMM. TO 12-3-61	$ 5,751 (GE)
	CONST. $ 31,000	GE $ 7,500	ESTIMATED TOTAL COST	$ 36,000

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 5-15-61	DATE AUTHORIZED 12-15-60		DESIGN 7-28-61	
CONST. 9-12-61	DIR. COMP. DATE 12-31-61		CONST. 12-29-61*	

ENGINEER

FEO

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		244
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		Through 12-10-61

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100	100	100
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides a means of making uranium scrap material safer for storage and off-plant shipment by converting this scrap to a stable uranium oxide. The facility will be adjacent to the 333 Building.

Building has been completed and accepted. The project was complete with exceptions 12-29-61 and will no longer be reported.

*With exceptions.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-7[illegible]42

DATE 12-31-61

PROJ. NO.	TITLE	FUNDING
CAH-916	Fuels Recycle Pilot Plant	4-62-d-3

AUTHORIZED FUNDS					
$ 385,000	DESIGN $ 385,000	AEC $	COST & COMM. TO 12-3-61	$ 321,952
	CONST. $ -0-	GE $ 385,000	ESTIMATED TOTAL COST	$ 5,000,000

STARTING DATES		DATE AUTHORIZED 10-27-61	EST'D. COMPL. DATES	
DESIGN	3-1-61	DIR. COMP. DATE --	DESIGN	6-15-62
CONST.	9-1-62		CONST.	7-1-64

ENGINEER

FEO - RW Dascenzo

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION	23	
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	34	34
TITLE I	11	100	100
GE-TIT. II	89	26	26
AE-TIT. II			
CONST.	100	NS	
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.

A total of 64 drawings have been issued for comment and two for approval. Twenty-five drawings have been issued for comment in the last month.

PROJ. NO.	TITLE	FUNDING
CAH-917	Field Service Center - Atmospheric Physics	61-j

AUTHORIZED FUNDS					
$	DESIGN $	AEC $	COST & COMM. TO	$	
	CONST. $	GE $	ESTIMATED TOTAL COST	$ 154,000	

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN	2-15-62*	DIR. COMP. DATE	DESIGN	6-15-62*
CONST.	8-1-62*		CONST.	4-15-63*

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 facilities necessary to conduct atmospheric physics research and development programs.

*Based on AEC authorization by 1-15-62.

There has been no change in the status of this project since the last report.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 72142

DATE: 12-31-61

PROJ.NO.	TITLE	FUNDING
CAE-921	Geological & Hydrological Wells - FY 1961	61-j

AUTHORIZED FUNDS					
$ 79,000	DESIGN $ 1,000	AEC $ 69,500	COST & COMM. TO 11-30-61	$ 63,21
	CONST. $ 78,000	GE $ 9,500	ESTIMATED TOTAL COST	$ 79,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
	DESIGN 4-15-61	3-24-61	DESIGN 5-15-61	
	CONST. 5-22-61	DIR. COMP. DATE 12-31-61	CONST. 2-6-62	

ENGINEER

FEO - HE Ralph

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	6	966
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTU.
DESIGN	100	100	100
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100	100	97
PF	0		
CPFF	3	100	100
FP	97	100	96

SCOPE, PURPOSE, STATUS & PROGRESS

This project involves the continued drilling of special research, test and monitoring wells.

Approximately 4,600 feet of hole has been completed. Contractor is working on the 16th and final well.

A request for a contract modification has been submitted via Design Change No. 2. This will extend construction work until 2-6-62.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS				
$	DESIGN $	AEC $	COST & COMM. TO	$
	CONST. $	GE $	ESTIMATED TOTAL COST	$

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
	DESIGN		DESIGN	
	CONST.	DIR. COMP. DATE	CONST.	

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUA
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 72142

DATE 12-31-61

PROJ. NO.	TITLE	FUNDING
CAH-922	Burst Test Facility for Irradiated Zirconium Tubes	62-k

AUTHORIZED FUNDS				COST & COMM. TO	
$ 29,600	DESIGN $ 29,600	AEC $		12-3-61	$ 29,600
	CONST. $	GE $ 29,600		ESTIMATED TOTAL COST	$ 228,000

STARTING DATES			EST'D. COMPL. DATES	
DESIGN	11-7-61	DATE AUTHORIZED 10-23-61	DESIGN	5-15-62
CONST.	6-15-62	DIR. COMP. DATE	CONST	12-1-62

ENGINEER

FEO - H. Radow

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	8	15
TITLE I			
GE-TIT. II	38.5	2	2
AE-TIT. II	61.5	12	23
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 now available because of the limited operating history of Zircaloy-2 pressure tubing in reactors.

AEC Directive No. AEC-199, dated October 23, 1961 authorizing $29,600 for design has been issued. Work Authority No. CAH-922 (1) dated October 25, 1961 was received.

General Electric portion of design started 12-12-61. Bovay Design of the architectural phase is progressing favorably. Final approval signatures are being obtained on the Design Criteria. The official design schedule has been submitted for approval.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 7214[illegible]

DATE 12-31-61

PROJ. NO.	TITLE	FUNDING
CAH-924	200 KW Induction Heating System - 306 Building	0290

AUTHORIZED FUNDS					
$ 31,000	DESIGN $ 3,200	AEC $ 24,650	COST & COMM. TO 12-10-61	$ 4,983 (GE)
	CONST. $ 27,800	GE $ 6,350	ESTIMATED TOTAL COST	$ 31,000

STARTING DATES			EST'D. COMPL. DATES		
DESIGN	5-1-61	DATE AUTHORIZED 3-27-61	DESIGN	10-15-61	
CONST.	10-15-61	DIR. COMP. DATE 2-28-62	CONST.	2-28-62	

ENGINEER

FEO - LF Higginson

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		38
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	NS	100
TITLE I			
GE-TIT. II		NS	100
AE-TIT. II			
CONST.	100	*	50
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide a source of power for induction heating for R&D work in the 306 Building.

*The construction schedule has not been issued by the AEC.

All construction materials have been ordered and actual installation will start when the first materials are on site.

The work stations have been uncrated and installation at external cabling has been started.

PROJ. NO.	TITLE	FUNDING
CAH-927	Additions to the 271-CR Building Waste Treatment Demonstration Facility	61-j

AUTHORIZED FUNDS					
$ 80,000	DESIGN $ 11,000	AEC $ 64,300	COST & COMM. TO 12-17-61	$ 14,697 (GE)
	CONST. $ 69,000	GE $ 15,700	ESTIMATED TOTAL COST	$ 80,000

STARTING DATES			EST'D. COMPL. DATES		
DESIGN	6-15-61	DATE AUTHORIZED 5-16-61	DESIGN	2-29-62	
CONST.	3-29-62	DIR. COMP. DATE 6-30-62	CONST.	7-1-62	

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER - Bovay Engineers	2	100
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	60	75
TITLE I			
GE-TIT. II			
AE-TIT. II	100	60	75
CONST.	100		
PF			
CPFF			
FP			

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 is now expected to be complete by 1-12-62.

HW-72142

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

DATE 12-31-61

PROJ. NO.	TITLE	FUNDING
CAH-932	300 Area Retention Waste System Expansion	62-k

AUTHORIZED FUNDS				
$	DESIGN $	AEC $	COST & COMM. TO	$
	CONST. $	GE $	ESTIMATED TOTAL COST	$ 70,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES		PERCENT COMPLETE			
	DESIGN 2-15-62*			DESIGN 5-1-62*		WT'D.	SCHED.	ACTUAL
	CONST. 6-1-62*	DIR. COMP. DATE		CONST. 10-1-62*	DESIGN	100		
ENGINEER					TITLE I			
FEO - OM Lyso					GE-TIT. II			
MANPOWER		AVERAGE	ACCUM MANDAYS		AE-TIT. II			
FIXED PRICE								
COST PLUS FIXED FEE					CONST.	100		
PLANT FORCES					PF			
ARCHITECT-ENGINEER					CPFF			
DESIGN ENGINEERING OPERATION					FP			
GE FIELD ENGINEERING								

SCOPE, PURPOSE, STATUS & PROGRESS

This project will increase the basin capacity commensurate with the increased volumes handled. This will permit transfer to crib waste of contaminated waste if required, and still permit adequate sampling time for the normal flow.

The project proposal was submitted to HOO-AEC for authorization on 5-5-61.

The proposal was returned unapproved on September 12, 1961, with a letter suggesting alternate solutions. These were reviewed for feasibility and practicability. The project proposal was returned to the Commission for further review and approval and is scheduled to be reviewed at the next Review Board Meeting.

*Based on AEC authorization by 1-15-62.

PROJ. NO.	TITLE	FUNDING
CAH-936	Coolant Systems Development Laboratory	62-k

AUTHORIZED FUNDS				
$ 93,000	DESIGN $ 12,000	AEC $ 78,000	COST & COMM. TO 12-17-61	$ 14,486
	CONST. $ 81,000	GE $ 15,000	ESTIMATED TOTAL COST	$ 93,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES		PERCENT COMPLETE			
	DESIGN 9-8-61	8-9-61		DESIGN 1-1-62		WT'D.	SCHED.	ACTUAL
	CONST. 4-9-62	DIR. COMP. DATE 10-31-62		CONST. 10-31-62	DESIGN	100	100	100
ENGINEER					TITLE I			
FEO - KA Clark					GE-TIT. II			
MANPOWER		AVERAGE	ACCUM MANDAYS		AE-TIT. II	100	100	100
FIXED PRICE								
COST PLUS FIXED FEE					CONST.	100	NS	NS
PLANT FORCES					PF			
ARCHITECT - ENGINEER - Bovay Engineers		2	166		CPFF			
DESIGN ENGINEERING OPERATION					FP			
GE FIELD ENGINEERING								

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides facilities for the conduct of corrosion and decontamination studies for nuclear reactor coolant systems, by the addition of 2,700 sq. ft. laboratory facility on the west side of the 1706-KE Building.

Design completion, in accordance with design criteria revision No. 1, and the comments from interested parties, is expected by 12-31-61.

Following this, a cost-to-complete estimate will be developed and bid information furnished.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 72142

DATE 12-31-61

PROJ. NO.	TITLE	FUNDING
CGH-937	Safety Improvements to 231-Z Building	61-J

AUTHORIZED FUNDS					
$ 45,000	DESIGN $ 5,500	AEC $	COST & COMM'L TO 12-3-61	$ 39,720
	CONST. $ 39,500	GE $ 45,000	ESTIMATED TOTAL COST	$ 45,000

STARTING DATES				EST'D COMPL. DATES		PERCENT COMPLETE		
	DESIGN 7-12-61	DATE AUTHORIZED 6-29-61		DESIGN 9-29-61		WT'D-	DENED.	ACTUAL
	CONST. 7-10-61	ENG. COMP. DATE 5-15-62		CONST. 2-1-62	DESIGN	100	NS	100
					TITLE I			
					GE-TITLE II		NS	100
					AE-TITLE II			
					CONST.	100	92	63
					PF			
					CPFF	100	92	63
					FP			

ENGINEER

FEO - JT Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES	6	285
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides supplemental ventilation and installation of a fire detection system in the 231-Z Building.

The two major pieces of Fire Alarm equipment have been received.

Fabrication of ductwork is 95% complete.

The sheetmetal men are back from the scheduled outage in 200 East Area. Installation of ductwork is now approximately 30% complete.

PROJ. NO.	TITLE	FUNDING
CGH-951	A-C Column Facility - 321 Building	0290

AUTHORIZED FUNDS					
$	DESIGN $	AEC $	COST & COMM'L TO	$	
	CONST. $	GE $	ESTIMATED TOTAL COST	$ 55,000	

STARTING DATES			EST'D COMPL. DATES		PERCENT COMPLETE		
	DESIGN	DATE AUTHORIZED		DESIGN		WT'D-	DENED.	ACTUAL
	CONST.	ENG. COMP. DATE		CONST.	DESIGN	100		
					TITLE I			
					GE-TITLE II			
					AE-TITLE II			
					CONST.	100		
					PF			
					CPFF			
					FP			

ENGINEER

FEO - E. Radow

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide a closely integrated "A" column in series with the relocated "C" column to permit the development of a mathematical model for the mass transfer of uranium, as well as the exploration of the possibilities of computer optimization of a combined "A-C" extraction battery.

The project proposal has been submitted to the AEC and will be on the agenda of the January 4, 1962, Review Board Meeting.

TECHNICAL ADMINISTRATION OPERATION

MONTHLY REPORT

EMPLOYEE RELATIONS

Fifteen non-exempt employment requisitions were filled during December with 25 active requisitions remaining to be filled.

Advanced Degree - Three PhD applicants visited for an employment interview. Four offers were extended -- no acceptances occurred and two rejections were received. Current open offers total five. During December, six universities were visited for PhD recruiting purposes.

BS/MS - Thirty-six students home for the holidays were interviewed for summer employment consideration. Two Technical Graduate Program offers were made and are still open.

Technical Graduate Program - One Technical Graduate was placed on permanent assignment and two (E&SP members) transferred to other Company locations. Current program members total 73.

TECHNICAL INFORMATION

This month completes the first six month's cycle for Personal Document Inventories since the program went on a semi-annual basis. An average of 226 inventories are distributed monthly under the new system as opposed to 445 monthly average under the quarterly system.

One classification memorandum was distributed to the field. It is HW-71974 "Classification: Uranium Recycle".

O. E. Boston, Manager
Technical Administration

TABLE II PROFESSIONAL PERSONNEL PLACEMENT

A. Technical Recruiting Activity - HAPO - September 1, 1961 to December 31, 1961

	Visits to Richland				Offers				
	Cases Considered	No Interest	Invited	Visited	Offered	Rejected	Accepted	Open	On the Roll
PhD	264	132	60	21	18	9	5	4	3
Exp. BS/MS	124	94	6	4	3	3	-	-	-
Prog. BS/MS	129	54	-	-	3	1	-	2	-

B. Technical Recruiting Activity - HL - September 1, 1961 to December 31, 1961

	Visits to Richland				Offers				
	Cases Considered	No Interest	Invited	Visited	Offered	Rejected	Accepted	Open	On the Roll
PhD	264	132	60	21	14	8	4	2	2
Exp. BS/MS	82	63	2	1	1	1	-	-	-

In addition to the above activity, 22 exempt employees have transferred into HL from other HAPO departments and 5 technical graduates have accepted off-program placement in HL to date.

C - Technical Graduate Program
Month ending December 31, 1961

Number Personnel on Assignment	73
(HAPO Tech Grad Program 60	
(Engineering & Science Program 13	

Distribution of Assignments by Departments

IPD	31
HLO	25
CPD	9
FPD	7
C&AO	1
CE&UO	0

Distribution of Assignments by Function

Research & Development or Engineering	56
Other	17

FINANCIAL OPERATION MONTHLY REPORT
DECEMBER 1961

GENERAL ACCOUNTING OPERATION

Travel activity continues at a level below that established the two previous fiscal years as shown by this schedule:

Trips Started - Year to Date	
FY 1960	705
FY 1961	645
FY 1962	542

Following is a summary of the status of letters or agreements covering specific actions requiring AEC concurrence:

No.	Title	Status
AT-52	Expanded Use of Whole Body Counter	Still pending
AT-104	Fission Products Dispersal Handbook	Rewritten and resubmitted
AT-105	Symposium on the Biology of Transuranic Elements	Still pending AEC concurrence
AT-196	Participation in Standardizing Activities	Approved by AEC 12-13-61
AT-205	Sanitary Engineering Advisory Committee-Washington State University	To AEC 12-28-61
AT-206	Delay in Disposition of Residence of Transferred Employee	Approved by AEC 12-22-61
AT-210	Delay in Disposition of Residence of Transferred Employee	Approved by AEC 12-22-61
AT-211	Delay in Disposition of Residence of Transferred Employee	Approved by AEC 12-26-61
AT-212	Delay in Disposition of Residence of Transferred Employee	To AEC 12-21-61
AT-215	Delay in Disposition of Residence of Transferred Employee	To AEC 12-28-61

During the month $266,391 of equipment was transferred to classified plant account from Equipment Work in Progress and $7,255 from Construction Work in Progress.

Nuclear Materials Accounting advised us of the forthcoming Survey 19, Part 2, which will consist of a verification of HAPO inventories of enriched uranium, thorium, and U-233 as of the end of January 1961. All material custodians were advised of the survey and requested to submit inventory information to enable us to prepare and submit a physical inventory schedule for HLO.

A letter was issued to the field requesting information required periodically by HAPO, the AEC and the Office of Defense Mobilization on machine tools usage for the period July 1, 1961 through December 31, 1961. Upon receipt of the requested information for HLO the report will be submitted to Contract and Accounting.

Certification inventory reports for the quarter ending December 31, 1961 were prepared and forwarded to HLO holders of Other Special Materials for completion and certification. Upon receipt of the requested information and reconciliation with HLO Financial records a report of results will be issued.

Preparations were completed for the physical inventory of movable catalogued equipment in the custody of Chemical Research and Development Operation. The inventory count will start on January 8, 1962 and continue through February 1962.

The calendar year 1961 square footage report indicated a net increase in HLO floor space of 55,997 square feet detailed below:

Total Reported Area at January 1, 1961			581,917 sq. ft.
Increases:	New Construction	70,926	
	Remeasurement of Existing Facilities	488	
	Total Increase	76,594	
Decreases:	Demolition	505	
	Area assigned to Others	20,092	
	Total Decrease	20,597	
Net Change			55,997
Total Reported Area at December 31, 1961			637,914 sq. ft.

The Property Management Manual was revised to reassign manual chapters to coincide with HAPO OPG numbers for reference ease. In connection with the reassignment of manual chapters the Table of Contents was revised and the Manual was divided into two Sections, Section I containing procedures peculiar to HLO and Section II containing HAPO OPGs.

Revised Property Management OPGs issued during the month are indicated below:

OPG No.	Title
8.1	Assignment of Plant and Equipment
8.6	Property Management of Tools

Preliminary unitization reports were prepared for projects:

CA-681	Hanford Equipment In the ETR	$1,038,038
CG-785	In-Reactor Studies Equipment-105KW	321,110

Cost in connection with Project CA-681 was transferred to classified accounts during December and costs in connection with Project CG-785 will be transferred in January. The unitization report will not be issued in final form until final billing is received from the AEC on CA-681 and accruals are cleared on Project CG-785.

A unitization report was issued on Project CA-747, Plutonium Fabrication Pilot Plant - 308 Building, and costs ($4,198,609) were transferred from unclassified to classified accounts.

Eighty-eight items valued at $37,121 were received at the Laboratory Central Storage Pool during the month of December. Thirteen items valued at $7,323 were loaned or transferred in lieu of placement of requisitions, three items valued at $1,465 were temporarily withdrawn by custodians and six items valued at $171,621 were withdrawn permanently by custodians. The permanent withdrawal includes two items of Hastelloy-X @ $995 and UO_2 powder valued at $169,873. There are currently equipment items valued at $481,962 physically located in the storage pool. Thirty-two items valued at $54,198 are held for the convenience of others. Reactor and Other Special Materials on hand at the pool at month-end totaled $181,901 and materials held for the convenience of others totaled $109,548.

For the first six months of Fiscal Year 1962, 129 items valued at $57,768 were loaned or transferred in lieu of placement of requisitions. Operating costs for the same period were $5,925.

A net reduction of $742,000 was noted in the SS Material Research and Development account 0530 during November 1961. This reduction can be summarized as follows:

	Amount in Thousands
Receipts from Off-Site	$ 11
Transfers to Other Departments	(420)
Material Consumed in Research	(348)
Material Produced in Reactor	16
Transfers between HLO Inventories	(1)
	$(742)

Receipts were in the form of Enriched Uranium and transfers out were primarily plutonium to CPD. Of the material Consumed in Research, $310 thousand is attributable to Plutonium representing the difference in value between HLO and return price to CPD, and the fact that the value of material transferred to the PRTR is less than the regular HLO R & D inventory value with the difference being charged to Consumed in Research. The PRTR has produced plutonium valued at $16 thousand which was added to inventory with an off-setting entry to a new account, Nuclear Materials Revaluations, Account No. 2960. C&AO Product Cost requested and obtained a unit cost from the AEC for this material.

Nuclear Material Work in Process inventory account 0510 showed an increase of $536,000 over October 1961. Receipts from other departments were $1,027,000 and Returns were $491,000. The November 30, 1961 balance in this account is $8,618,000.

Three-hundred eighty-three Extended Property Passes covering the routine movement of specific categories of property within the plant areas were prepared and issued during the month of December. These passes will expire on December 31, 1962 or upon termination, transfer, etc., of the person to whom it was issued.

Heavy Water losses chargeable to operating cost for the month of December amounted to $7,583 representing $355 scrap material and $7,228 BPID loss. At December 31, 1961, accumulated scrap for return to Savannah River totaled 15,199.88 pounds, valued at $184,979.33.

Four suggestions were evaluated. An award of $25.00 was recommended and paid for one while the other three were rejected.

Contracts processed included:

DDR-138	Battelle Memorial Institute
MRO-145	Fisher Scientific Company
CA-313	Henry Eyring
SA-195	Sperry Product, Inc.
SA-193	Reactive Metals, Inc.
SA-191	The Swedish Hospital

The following two revised OPGs were issued:

5.2	Authorization for Construction or Acquisition of Plant and Equipment
22.1.10	Technical Administration Operation Organization

Five new forms were established during December. In addition, several forms were revised or renumbered for processing by Central Printing due to the reduction of services provided by 300 Area Printing and Duplicating.

There were nine revisions to the AEC Manual, the contents of which are covered in a separate report to the Section Managers.

Action as indicated occurred on the following projects during the month:

New Money Authorized HLO

CAH-842	Critical Facility	$10 000
CAH-927	Waste Demonstration Facility 271-CR	(1 800)

Physical Completion Notices Issued *AEM Services Only

CGH-834	Modifications and Additions to High Pressure Heat Transfer Apparatus - 189-D Building
CAH-914*	Rattlesnake Springs Radioecology Facility
CAH-919*	Air Conditioning 314 Building

Construction Completion and Cost Closing Statements Issued

CAH-935*	Metals Storage Building

COST ACCOUNTING OPERATION

A summary of final project costs versus original cost estimates was prepared on Project AEC-167, "Plutonium Recycle Test Reactor", for use by Reactor Engineering Development for inclusion in a report on project construction and startup history.

Special requests established during the month were as follows:

Accounting Code	Activity
.1Y	E. D. Clayton - To consult with Vitro International at New York. He will review the criticality hazards aspects of design of a reprocessing plant to be constructed for the Italian Nuclear Energy Commission.
.3A	J. C. Fox - To consult with Carolina-Virginia Nuclear Power Associates, Inc., concerning pre-operational tests and associated problems.
.4C	An additional authorization in the amount of $3,000 was received from Union Carbide Nuclear Company to perform X-ray diffraction studies on five additional UO_2 specimens. Total authorization is now $8,000 for this job.

Considerable time was devoted to recasting FY 1962 fiscal year-to-date billings for irradiation unit charges from IOO-AEC. The number of irradiation units and associated costs were related to each experiment and GEH license as well as the appropriate AEC program code. This information was transmitted through Contract Accounting to HOO-AEC who will make a formal adjustment in the irradiation unit charges' billing.

W Sale
Manager - Finance

W Sale:whm

TEST REACTOR AND AUXILIARIES

MONTHLY REPORT

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

The reactor was down throughout the month of December except for a 4-hour period of 0 power level operation for the performance of PRTR Test No. 12, Measurement of Effective Delayed Neutron Fraction. Total energy generated has been 3646.5 MWD. Maximum fuel element exposures are 53.3 MWD for UO_2 (~1080 MWD/T) and 53.2 MWD for Pu-Al (~31.5% burnup). In-reactor inventory remains at 39 Pu-Al and 46 UO_2 elements. Three fresh Pu-Al elements equipped with flux monitors were charged in ring 5.

D_2O loss for December was 517 pounds as determined by inventory method. Helium loss for the month was 162,000 scf.

In order to permit repairs to the primary pump #2 discharge check valve (P-2), several tests were performed and special equipment and procedures developed to permit draining of the primary system between the pumps and the process tubes while maintaining adequate cooling of fuel elements. After approximately 19 days of outage, the water temperature just above the fuel element in a stagnant process tube containing a maximum power Pu-Al fuel element rose from 70°F to 114°F in four hours and leveled off. Another test demonstrated leak tightness of angle valves separating the process tubes from the part of the system to be drained. Fifteen tubes exhibited leakage rates of 2 inches per 10 minutes or greater. These tubes were discharged prior to performing repairs on P-2. The procedures developed for the P-2 valve repair work saved an estimated three days over the time estimated to completely discharge the reactor and drain the entire primary system.

Tests were performed to induce vapor locking of a primary pump. The low flow instrumentation demonstrated an ability to signal the resulting no-flow condition and procedures were developed to relieve the vapor lock situation.

The Planning Engineer function was filled and two additional non-exempt personnel were assigned to planning for operation of new facilities. A major portion of PRP-CF operating procedures are in draft form and orientation for the Gas Loop was started.

Equipment Experience

Two primary pump mechanical seals were repaired, one from a September failure. In addition one pump was completely overhauled to replace bearings.

Check valve P-2 stuck open which was caused by an internal bolt on the counterweight arm working loose and binding. The bolt was welded in place. The nuts had been welded prior to installation. Bolts on valves P-1 and P-3 were also welded in place. One of these was slightly loose.

A valve (E-9) on the heavy water side of the light water injection system leaked heavy water. This valve was supposed to be a single port valve. Upon inspection it was found that it was a double port valve with a plug screwed into the bottom port. After much difficulty, it was found that leakage was past the threads of the plug and the threads of the valve seat. Tightening of these pieces corrected the leakage.

Helium leaks on the top face were reduced by replacing the top shield shim gaskets and cleaning and retorquing the nozzle holdown bolts. However, it was found that only nominal forces on the side of a nozzle would cause leakage again.

Helium leakage past gaskets on the lower face were observed. A program was established to regasket as the respective tubes are scheduled for discharge.

Improvement Work

The following work was completed:

Installation of low flow indicator and alarm.
Criticality - Evacuation Alarm Inter-tie.
Relocation of Primary Injection Line.
Thermal Barrier Relief Valve.
D_2O Storage Tank Level Instrumentation.

The following work was partially completed:

Experimental Capacitance Probe (Moderator Storage Tank Level)
Primary Pump Vents (complete on pumps 1 and 3)
Modifications to Fuel Transfer Unit
Moderator and Reflector, Top and Bottom Shield Pump seal replacement.
In-Line Gas Sampling.
Leakage collection on primary valves.

Engineering work was performed on the following:

Relocation of pressurizer level transmitters - complete
Series block valves in helium lines to stack - issued for approval.
Automatic return of D_2O condensate from HX-7 - partially complete.
Remote oil addition to the High Pressure Helium Compressors

Technical Planning

Preliminary results from PRTR Test No. 12 show that the ratio of the effective delayed fraction to the prompt lifetime has dropped from approximately 7 for the initial reactor loading to about 6 for the present loading. This reflects a reduction in the effective delayed fraction and hence, the value of the dollar of 0.001. The data were partially obscured by an

oscillation, most probably in the gas balance - moderator system.

The first draft of the proposed experimental program for 1962 was prepared.

Final reports were prepared for the following PRTR tests:

Secondary System Shutdown Water Quality Limits

Deoxygenating by Use of a Sulfite Regenerated Anion Resin

Practice Charge-Discharge - Nested Tubular Element

Test of Reactor Cooldown and Depressurization in the Event of an Evacuation - Check of Depressurization Method

Safety

One Operating Standard was revised and two others combinding several standards were issued for approval. Four Maintenance manuals were issued for comment. Four Maintenance Procedures were issued for comment. Five Preventive Maintenance Procedures were issued. Twenty-five drawings were approved for as-built.

Project Management and Design

Plutonium Recycle Critical Facility (Project CAH-842)

The project is 88% complete (adjusted to revised scope). A directive authorizing a total of $400,000 and a completion date of March 15, 1962, was received. The vendor continues to have difficulty with the cadmium cladding for the control rods. Instrument check out and calibration was nearly completed.

Fuel Element Rupture Test Facility (Project CAH-867)

Construction is 67% complete versus 75% scheduled. The motor control center, received during the month, did not meet specifications. Negotiations are under way to settle costs of modifications. Design criteria were modified to provide a second rupture monitor sampling line and basin clean-up facilities.

Testing at 314 Building demonstrated the planned method of cooling fuel elements during reactor discharge to be inadequate.

GAS COOLED POWER REACTOR PROGRAM

Gas Loop Project Management (Project CAH-822)

Installation is 92% complete. Electrical design tests have been completed and instrument design tests are in progress. Pressure testing of piping is under way. The latest estimate for completion of the blowers is late March 1962.

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
R. G. Wheeler and F. B. Quinlan	Bonding Iron and Iron Alloys to Uranium Using a Molybdenum Interface
R. H. Moore	A Process for Decontamination of Molten LiCl-KCl Solutions Containing Uranium and Plutonium Chloride Preliminary to Incorporation in the "Salt Cycle" Process (HW-72151)

W. K. Rea

for Manager, Hanford Laboratories