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HANFORD LABORATORIES OPERATION
MONTHLY ACTIVITIES REPORT
NOVEMBER 1956

Compiled by
Operation Managers

December 21, 1956

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

Work performed under Contract No. W-31-109-Eng-52 between the Atomic Energy Commission and General Electric Company

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HANFORD LABORATORIES OPERATION
FORCE AND PERSONNEL STATUS CHANGES
NOVEMBER 1956

	At Close of Month			At Beginning of Month			Additions		Separations		Sep. Rate	
	Exempt	Non-Exempt	Total	Exempt	Non-Exempt	Total	Exempt	Non-Exempt	Exempt	Non-Exempt	M	F
Chemical Research and Development	127	94	221	126	94	220	2		1		.694	
Reactor and Fuel Research and Dev.	139	82	221	135	83	218	4	3		4	.529	3.125
Physics and Instr. Research and Dev.	59	23	82	60	20	80	1	1	2		2.817	2
Biology	30	41	71	31	42	73		1		2		7.143
Operations Research and Synthesis	12	3	15	12	3	15						
Radiation Protection	41	203	244	41	209	250		2		7	.781	5.172
Lab. Auxiliaries	36	199	235	34	196	230	1	9		2	.740	1.000
Financial	15	29	44	15	30	45				3		15.000
Employee Relations	13	12	25	13	11	24						
General	1	1	2	1	1	2						
Total	473	687	1160	468	689	1157	8	16	3	18	.789	3.250
Composite Separation Rate											1.724	
The separation rate for other HAPO components is determined on the basis of their personnel who leave HAPO employment. The HLO separation rate determined in this manner is -											1.466	

HANFORD LABORATORIES OPERATION
PROMOTIONS AND TRANSFERS[1]
NOVEMBER 1956

Component	Promotions			Exempt Transfers[2]				Non-Exempt Transfers	
				To HLO		From HLO			
	Exempt	Non-Exempt to Exempt	Non-Exempt	From Other HAPO	From Other G.E.	To Other HAPO	To Other G.E.	To HLO	From HLO
Biology	1	0	1	0	0	0	0	1	0
Chemical	4	0	3	0	0	0	1[4]	0	0
Laboratory Aux.	1	0	11	0	0	0	0	1	1
Operations Research and Synthesis	0	0	1	0	0	0	0	0	0
Physics and Instr.	0	0	3	0	0	0	0	1	0
Radiation Protection	1	0	3	0	0	0	0	1	0
Reactor and Fuels	4	2[3]	3	0	0	0	0	3	0
Employee Relations	0	0	1	0	0	0	0	0	0
Financial	0	0	2	0	0	0	0	0	0
Total	11	2	28	0	0	0	1	7	1

(1) Data through 11/30/56
(2) Transfers within HLO not included
(3) W. J. Gruber and F. J. Worzala
(4) J. L. Murray to APED

TECHNICAL ACTIVITIES

METALLURGY AND REACTOR FUELS

Uranium Metallurgy

Uranium Grain Size. At the meeting of the Committee on Metallography held at Savannah River, concurrence was received for the changes made in the grain size chart originally prepared at HAPO and recently amended by the substitution of certain new photomicrographs. The chart was retitled "Alpha Treated Uranium Grain Size Chart" to indicate that it was acceptable for the estimation of uranium grain size in all conditions of heat treatment except beta transformed.

The estimation of the grain size of uranium heat treated in the beta region is difficult for several reasons. It is characterized by grains of very non-uniform size and by non-equiaxed grain shapes. For these reasons a chart consisting of several photomicrographs in each range of grain size from 0.065 mm to 0.500 mm average grain diameter might be more applicable, since more grain shapes would be introduced for comparison and it would be correspondingly easier to estimate unknown sizes. The first chart using this approach to the problem was prepared at HAPO and introduced at the meeting held at SRL. Favorable comments were received and the method shows promise as one approach towards determining beta treated grain sizes. A trial period was recommended by the members. Grain size charts are now being prepared for the other members of the committee. A small group of uranium specimens will be prepared and distributed among the committee members and grain size estimates made and evaluated to determine the feasibility of this method.

Low Temperature Irradiation of Uranium Tensile Samples. Six tensile samples from PT-105-528-SI have been tested this month. These samples were prepared from beta annealed uranium sheet and were irradiated to maintain sample temperature below 50 C. The samples were irradiated in two groups of three each with exposures of 10 minutes and 1-3/4 hr (approximately 10^{16} and 10^{17} nvt respectively). After irradiation the samples were kept under refrigeration for the purpose of preventing self-annealing. The results of the tests indicate that even with low exposures the uranium suffers extensive damage. The changes observed are indicative of the "keying" of the deformation processes rather than of damage produced by fission products or microcracks. The remaining samples from the test will be annealed to determine whether these short-term effects can be reduced by thermal treatment.

Effects of Short-Time Exposure on Uranium Tensile Properties

Sample	Exposure	Ult. Strength	0.2% Yield Strength	Elong.	Mod.
2-2	10^{16} nvt	72,700 psi	39,200 psi	7.0%	24.8×10^6 psi
2-3	"	63,400	34,800	4.4	30.4×10^6
2-6	"	72,200	33,200	6.6	28.7×10^6
3-2	10^{17} nvt	64,700	43,400	4.1	27.6×10^6
3-3	"	71,200	41,300	4.9	33.6×10^6
3-5	"	73,900	42,200	6.6	26.6×10^6
Unirradiated Control Average		70,550	26,800	7.7	20.6×10^6

Thermal Stresses. To extend the temperature ranges in which the thermal stresses may be calculated by the method reported in HW-46125 "General Solution for the Thermal Stresses in a Cylinder", an analytic expression describing the creep of uranium was obtained. Creep data from BMI were analyzed by a method similar to that reported in HW-40494 "A Model of Mechanical Behavior Evaluated with Creep Tests Applied to Alpha Uranium". The previous analysis was generalized and an expression covering the temperature range 100 C to 600 C has been obtained.

The coefficients previously called the strain hardening coefficients and activation energy are now temperature dependent. The stress dependence is exponential and the coefficient in the exponent is not temperature sensitive.

The thermal stresses in four types of cylindrical fuel elements will now be calculated using this material law and the method described in HW-46125. These cases are (1) solid cylindrical fuel element externally cooled (2) hollow cylindrical fuel element externally cooled (3) hollow cylindrical fuel element internally cooled and (4) hollow cylindrical fuel element internally and externally cooled.

Diffusion Studies. Diffusion in non-irradiated U/AlSi and U/Zr couples is being investigated to provide a basis for interpretation of the results from irradiated couples. The results from eight couples that were analyzed this month indicate a penetration of the uranium into the AlSi of about 2.5 mils at 250 C after 45 days, about 6 mils at 300 C after 30 days, and approximately 6.5 mils after 45 days at 300 C. These couples were analyzed by the method described in the June 1956 Metallurgy Quarterly. All the values for couples annealed at 300 C were lower than expected. Chemical analysis of the AlSi used in one of the couples indicated a lower than eutectic silicon content. Six additional U/AlSi couples are now being annealed at 300 C to recheck the rate of diffusion at this temperature. Analysis of these couples should complete the work on the diffusion of uranium into AlSi in non-irradiated couples. A terminal report is being written.

UO_2 Particle Size Measurements. Extruded UO_2 rods, approximately 93 per cent of theoretical density, were examined by electron microscopy techniques. Chemical etching of the surface of the specimen did not produce a satisfactory microstructure. Cathodic etching of the samples is being investigated. Measurement of the particle sizes of UO_2 powders by electron microscopy techniques revealed that the average diameter of Mallinckrodt UO_2 particles was 3 microns after ball-milling with uranium rods for twenty-four hours. An apparatus for measuring the apparent porosity and size of the pores in UO_2 samples is being developed. A literature survey of solar furnaces and their possible application to studies of UO_2 was completed.

Plutonium Metallurgy

Design work for experimental casting fabrication of a new shape is complete and the mold drawings are in the shops. Molds for as-cast tensile test specimens, wrought tensile test specimens, fluidity test, thickness vs. density and chemical analysis standards have been completed. The chemical analysis standards were cast. Mold temperature was about 450 C, metal about 850 C and pressure was about one micron of mercury. Metal froze up in the cup on the first attempt at pouring the ingots for the wrought tensile test specimens. Mold temperature was 240 C, metal temperature 860 C. Sketches of graphite molds for casting of standard creep specimen,

cantilever notched impact specimen, and tension impact specimen type V were submitted for fabrication. The design of these molds will allow for metal shrinkage without fracture of the metal for damage to the molds.

Analytical results from plutonium samples melted in zircon, zirconia, magnesia and graphite have been completed. The results show only carbon pickup and possibly some iron contamination from the graphite.

Copper thickness standards to be used in developing a non-destructive thickness testing gage were electroplated in the Unichrome copper plating bath. A new plating tank has been fabricated in the shop. This tank will incorporate cathode rod agitation, both horizontal and vertical. Plating mandrels and plating racks for the electroformed shipping cans have been designed and the miniature barrel plater has been redesigned. It is hoped that this new design will enable successful electrodeposition on Pu-Al alloy discs.

Experimental dies for obtaining pressing data have been received. An experimental pressing die in the same size range as the proposed prototype devices has been designed and is being fabricated. This die will be used in the 440,000 lb Universal testing machine.

A number of unsuccessful attempts have been made to etch plutonium by the cathodic vacuum technique. It is clear that one of the major problems is thermal contact between specimen and cathode.

The titanium metal Evapor-Ion pump and high vacuum evaporation equipment and techniques have been investigated. With slight modifications, the Evapor-Ion pump could be used for metal coating equipment, hot stage metallography, etc. Very little work has been done on thick (>1 mil) metal coatings. Techniques for holding metal being evaporated and methods for controlling the geometry of the evaporated coating have been learned. Conventional coating equipment could be modified for vapor deposition of plutonium metal.

Zirconium Metallurgy

Recrystallization and Recovery of Zircaloys. The kinetics of recrystallization and recovery in Zircaloy-2 and Zircaloy-3 are being determined to establish optimum conditions of heat treatment during fabrication operations. Per cent cold work, temperature, time and heat treatment atmosphere were selected as variables. Hardness data were obtained for three cold work levels for both materials annealed in air and helium at temperatures from 300 to 800 C for 10 to 1000 minutes. Metallographic examination has been started for approximately 200 specimens. Initial 680 F water corrosion results for as-worked specimens of these materials were obtained from tests conducted at the Bureau of Mines. Additional heat treated specimens were placed in test during the month.

Tensile Properties of Irradiated Zircaloy-2. Post-irradiation examination was continued of Zircaloy-2 tensile specimens exposed two operating cycles in the L-47 position of the MTR. The estimated neutron exposure of the material was 1.4×10^{20} nvt (>1 Mev) and 7×10^{20} nvt (thermal). Annealed and cold worked conditions were investigated. Changes occurred in the tensile properties in both conditions. Irradiated annealed samples increased in 0.2 per cent yield strength in excess of 50 per cent, the ultimate strength increased 10-20 per cent, while

elongation decreased about one-third of the initial value. In the cold worked specimens, a lower increase in yield occurred. Ultimate strength also increased and elongation was halved. The changes observed compare favorably with the maximum change determined for annealed and cold worked zirconium irradiated at Hanford to an exposure of 2.4×10^{20} nvt (thermal).

X-Ray Diffraction Analysis of Irradiated Zirconium. Preliminary but precise measurements of line broadening have been made on zirconium irradiated to 6.0×10^{19} nvt. Contrary to previous conclusions, structure damage is indicated by an increase in line breadth of approximately 20 per cent. Further studies are in progress to determine the extent of this damage.

Static Corrosion Tests in Organic Coolants. During the past month, tests were performed to determine the effect of MIPB on zirconium at temperatures of 425 and 450 C. The zirconium contained approximately 16 ppm of hydrogen. After 80 hours at 425 C, the zirconium contained about 1000 ppm of hydrogen and was very discolored. After 6 hours at 450 C, the zirconium contained about 300 ppm of hydrogen. After 150 hours at 450 C, the zirconium was very brittle, indicating a very extensive reaction with hydrogen.

Zirconium Hydride Survey. A literature survey on the properties of zirconium-hydrogen alloys was continued. Emphasis has been placed on the accumulation of any data regarding changes in mechanical properties with temperature and hydrogen concentration. Adequate impact strength data of this nature have been reported and impact transition temperatures for zirconium and Zircaloy-2 containing up to 400 ppm hydrogen have been clearly bracketed. However, it appears that only limited data of this type are available on tensile strength, yield strength, or ductility at room temperature and 260 C and none on the tensile transition temperatures based on these properties.

An extensive experimental program is needed to obtain this data if zirconium or its alloys are to be utilized with confidence in pressurized reactor tubes. Establishing the behavior of each of the above mentioned mechanical properties as a function of temperature and hydrogen concentration calls for the preparation and testing of a multitude of samples unless the extent of the work can be narrowed in a manner explained by Forscher at WAPD. Forscher hypothesizes that strain induced porosity (and decreased ductility) is a manifestation of a precipitated hydride in zirconium. Furthermore, he notes that strain induced porosity does not appear at test temperatures high enough that all hydrogen is taken into solution, i. e., temperatures above the terminal solubility temperature for a given hydrogen content.

If changes in the mechanical properties of zirconium with temperature and hydrogen content are dependent on the presence or absence of a precipitated hydride, then it would appear that a knowledge of terminal solubilities of hydrogen in alpha zirconium should be particularly useful in predicting transition temperatures.

Terminal solubilities of hydrogen in alpha zirconium have been experimentally determined by Gulbransen and Andrew in the temperature range 425 to 600 C. An extrapolation of their results follows.

Temperature C	Terminal Solubility ppm Hydrogen
400	209
300	63
200	12
100	0.8

Zirconium Process Tubing for HAPO Reactors. A zirconium billet was successfully pierced by Allegheny Ludlum Steel Corporation. The pierced hole was 3-1/2" in diameter and 15" long, and exhibited excellent concentricity and a uniformly smooth surface. A material savings of about 20% should be realized by piercing rather than drilling the extrusion billet. The pierced Zircaloy billet was extruded into a ribbed tube blank with good results.

The zirconium and Zircaloy strip to be roll-formed and welded by the Trent Tube Company was found to contain areas that were undersize or that had edge cracks. The amount of usable strip will be sufficient for approximately 4 tubes from each alloy. The rod for the tube rib was rolled into the rib contour and will be vacuum annealed prior to shipping to Siacky Brothers where the ribs will be welded to the strip.

Mechanical Properties of Aluminum Alloys

The results of elevated temperature tests of the M-257 aluminum alloy revealed that the ultimate and yield strengths are approximately three times greater than for alloys 1100 (2S) and 6063 (63S) at temperatures above 200 C. At test temperatures of 100 C and above, fracture occurred through the extensometer punch marks or through the shoulder of the specimen. This behavior indicates a notch sensitivity of the alloy at elevated temperature and warrants further study.

The mechanical properties of the M-457 alluminum alloy (aluminum powder compact plus 1% nickel) were determined at temperatures up to 300 C. The ultimate and yield strengths were not quite as high as M-257. However, they were about two times greater than those for 1100 and 6063 alloys at temperatures above 200 C.

Aluminum Corrosion Studies

Corrosion-Resistant Films Produced by Autoclaving. Films having varying degrees of corrosion resistance may be deposited on aluminum by autoclaving in different media. For example, it was shown by a previous production test that fuel elements autoclaved in a 1 per cent CrO_3 solution and exposed in the reactor along with standard autoclaved fuel elements had less than half as much corrosion (weight loss) as the standard pieces. Further tests are being made to check these results as this appears to be a simple, inexpensive method of reducing in-reactor corrosion. The best method for checking these results is an in-reactor exposure of several tubes up to the present goal exposure. The preliminary laboratory tests to determine the best autoclaving procedure are now in progress.

Recent experiments have shown that small concentrations (about 5 ppm) of phosphate added to the water decrease the corrosion rate of aluminum. Consequently, it was concluded that a corrosion-resistant phosphate film might be applied to aluminum by autoclaving in a solution containing phosphate ion in suitable concentration and at a suitable pH value. Various solutions of phosphate have been used as autoclaving solutions. The phosphate ion concentration was varied from 100 to 30,000 ppm and the pH was varied from 1.3 to 7.0. Previous results had shown that solutions at pH values above 7.0 deposited a black flaky, non-adherent film of no value for corrosion resistance. The results from the present series of experiments were discouraging. The films deposited from solutions containing more than 1000 ppm were chalky and non-adherent. The films deposited from dilute solutions (less than 1000 ppm $PO_4^{\equiv}$) were thin and appeared to be about as protective as a water autoclave film. Based on all the laboratory tests, it is concluded that films deposited from phosphate solutions are not as protective as the film deposited from deionized water.

Phosphate Inhibition of Aluminum Corrosion. The effect of phosphate additions to reactor process water on the corrosion of M-329 aluminum was measured in laboratory tests. Samples were exposed for 308 hours at 92 C under conditions of low flow. The process water was obtained from 1706-KE and contained 2 ppm dichromate at pH 7. Phosphoric acid was added to give concentrations of from 0.5 to 100 ppm phosphate ion. Phosphate concentrations of less than or equal to 1 ppm had no corrosion inhibiting effect, while phosphate concentrations in the range from 2 ppm to 100 ppm were beneficial in reducing the aluminum corrosion rate. Further tests are being made to determine the effect of varying the phosphate concentration between 2 and 5 ppm.

Corrosion Product Formed in Phosphate Solutions. A sample of the corrosion product formed on aluminum during exposure to water containing phosphate ion was identified by x-ray diffraction analysis to be augelite $Al_2PO_4(OH)_3$ or $2Al_2O_3 \cdot P_2O_5$, $3H_2O$. Boehmite ($Al_2O_3 \cdot H_2O$), which is the corrosion product usually formed on aluminum in aqueous media, was not detected in the sample. The sample of corrosion product was obtained from an aluminum sample which had been exposed in H-Loop for 24 days at 195 C to pH 4.5 demineralized water containing 5 ppm of phosphate ion.

Effect of Process Water Purges on Corrosion Rates. In-reactor deionized water cooling systems usually are backed up by process water supplies. These process water supplies may be used when the reactor is down. An experiment was conducted to determine whether such intermittent uses of process water adversely affect the corrosion rate as measured in such loops.

Two autoclaves were charged with corrosion samples. One was operated on a 24-hour cycle of (1) 16 hours at 170 C on deionized water (2) 2 hours at 25 C on deionized water (3) 3 hours at 25 C on 300 Area process water and (4) 3 hours at 25 C on deionized water. The companion autoclave was operated solely on deionized water, with 16 hours at 170 C and 8 hours at 25 C. The condensed results are given in the following table.

Effect of Alternating Process and Deionized Water on the Corrosion of Aluminum. Penetration in Milli-inches.

Alloy	Deionized Water		Alternating Water	
	One Month	Three Months	One Month	Three Months
M-329 & M-388	0.07	0.09	0.21	0.30
X-2219	0.08	0.12	0.29	0.46

The effect of alternating water is somewhat exaggerated in this test because alternations were made more often than they would be in the reactor. Furthermore, the 300 Area process water is known to be more corrosive than 100 Area process water.

High Temperature Corrosion of Aluminum. A vigorous test in 363 C water has shown alloys M-388 and M-400 to be significantly more resistant than other special alloys tested. This is in contrast to tests at lower temperatures. The alloys of the same group tested at 363 C were nearly indistinguishable in corrosion rate at 350 C and lower temperatures. Continuing tests of M-329 (can stock) in 200 C water indicate a greater resistance to intergranular attack for annealed samples than for those in the as received condition. This is in agreement with results from previous tests.

Corrosion Testing of Harvey Cans. Fifty standard 8-inch, M-329 fuel element cans produced by the Harvey Machine Company were corrosion tested. Their performance in aqueous sodium chloride-hydrogen peroxide solutions and in aqueous sodium chloride-gelatin-aluminum reagent solutions was similar to results previously obtained on ALCOA cans.

Twenty-three of the 50 cans were spot-tested for iron and copper contamination. No copper contamination was found. Several of the cans contained minute particles of a ferrous substance.

Effect of Iron Additions on Resistance of Can Stock to Intergranular Attack. The continuation of hot spot failures under current reactor conditions has suggested revision of can stock chemical specifications. Since past work has pointed to the importance of iron concentration, samples of aluminum melts differing only in iron concentration have been prepared and tested for resistance to intergranular corrosion in high-temperature water. Weight gains were approximately the same as for M-388 which is not subject to intergranular attack under the test conditions. From the preliminary results obtained it appears that a certain minimum concentration (greater than 0.5%) of iron is required for the greatest corrosion-resistance. Further tests are planned and may result in a recommendation for changing the chemical specifications for aluminum can stock.

X-Ray Diffraction Analysis of Irradiated Molybdenum

A study of the high angle x-ray diffraction lines obtained from irradiated molybdenum has revealed an increase in line broadening and lattice parameters with increasing exposures to fast flux. The broadening of the lines began between 10^{18} nvt and 10^{19} nvt fast and increased rapidly up to the maximum exposures of 1.2×10^{20} nvt. A measure of the unit cell size gives a maximum increase of

0.047 per cent in the a_o value at an integrated flux level of 5.0×10^{19} nvt. For exposures greater than this the exact position of the line maxima has not been determined because of the broadening interference between the α_1 and α_2 peaks. Accurate means of separating these peaks are being studied so that lattice expansion measurements can be made on material irradiated to 1.2×10^{20} nvt.

Graphite Studies

Graphite Burnout Monitoring. Weighed graphite samples charged into the bare process channel 3478-D on February 12th, 1956, were discharged on November 18th. The samples were exposed to the prevailing conditions of moderator temperature and reactor gas composition to determine the current rates of moderator oxidation. The results from each group of five samples are presented in the following table as the average per cent oxidation along with the standard deviation of the mean.

Distance from Front Van Stone Flange, Feet	Average Per Cent Oxidation
11.0	0.021 ± 0.002
20.8	0.340 ± 0.032
30.5	0.013 ± 0.001

Considering the length of exposure, the extent of oxidation is not excessive if the effective exposure period is assumed to be 200 operating days, the central burnout rate would be about 1.7 per cent/1000 operating days. This rate is somewhat higher than normally observed (approximately 1%) and indicates that the oxygen or water vapor concentration in the reactor gas may be higher than normal.

High Temperature Graphite Irradiations. During a MTR shutdown, GEH-9-3 test thimble was discharged from the reactor and GEH-9-4 was charged. This irradiation will last for six weeks with goal temperatures of 750 C.

New startup and operating instructions were devised to gain information that might explain the phenomena experienced in previous irradiations in which the temperature dropped despite increased heater power. As was expected, GEH-9-4 experienced a dropping temperature in positions 2, 3, and 4, the highest flux positions. Analysis of all available evidence indicates that the temperatures are at goal or higher. Startup tests with zero heater power showed that samples in these positions reached almost to goal temperatures and that the goal temperatures were reached with only moderate heater powers. This is not the observed case after several days irradiation which places more suspicion on faulty thermocouples as the reason for failure to hold the control point.

Controlled Medium Temperature Irradiation of Graphite. Tests on a prototype heater for this in-reactor experiment are continuing. The latest heater design uses a Calrod type element wound around a standard sample casing. This should eliminate some of the difficulties encountered in prior experiments.

Reactor Graphite Thermocouple Development. Prior to further reactor testing of thermocouples, an attempt is being made to determine the cause of reactor thermocouple failure by examination of failed thermocouples removed from reactors. Resistance measurements were made on stringer RS-6 at the KE Reactor. Two of

the eight thermocouples seemed to be working properly and the rest were either grounded or shorted. The RS-6 stringer at KE will be removed after December 12th, by pulling it into a rod cave east of the 105 KE Building and it will be allowed to decay for a month before sampling.

Stainless steel sheathed thermocouples have been welded and radiographed in preparation for laboratory life testing. Initial tests will be made in tube furnaces at 600, 800, and 1000 C with 100 per cent CO_2 atmosphere.

High Temperature Thermal Conductivity Apparatus

A new terminal strip of Micalex and a molybdenum radiation shield have been designed, constructed, and installed in the high temperature thermal conductivity apparatus. These are for the purpose of improved temperature control and smoother thermocouple behavior during the measurements. Comparable problems are being experienced at BNL and the exchange of ideas should lead to an earlier solution at both sites.

Evaluation of Plastics and Elastomers

Exposure of plastics and elastomers to a variety of environments is continuing. The effect of gamma radiation dose rate now being studied under GEH-5-2 at the MTR will be completed about the first of the year. Rates from 10^4 to 10^7 r/hr are being used to determine the existence of a rate sensitive effect. Preliminary results indicate that materials such as Teflon which are cleaved by irradiation damage are affected to a varying degree by different dose rates, while damage to cross-linking materials such as polyethylene is independent of rate. The combined effects of temperature and radiation on promising materials is also being studied.

Tests have been completed to show that there is no post irradiation recovery of properties in the range from 12 to 72 hours after removal from the gamma field. Shorter periods of time are being investigated. Similarly, observation of what appears to be stress cracking in some irradiated elastomers is being investigated by irradiating elongated samples.

Standard Uranium Fuel Elements

K-Reactor Split Failures. The examination of Failure No. 739 from tube 4573-KE continued. The presence of intergranular corrosion of the jacket in the hot spot zone was confirmed metallographically. There was no evidence of jacket melting in this area, but indications of uranium diffusion into the aluminum were observed.

100-H Hot Spot Rupture. Metallographic examination of the 8-inch uranium slug which failed in tube 1587-H, on September 6, 1956, after an exposure of 376 MWD/T disclosed several non-standard conditions. In addition, extensive intergranular corrosion had occurred in the heat-affected zone. Other areas of the sample are being checked to determine whether the intergranular corrosion occurred outside the heat-affected zone. Extensive diffusion of uranium or silicon into the can wall was observed in the heat affected zone. The compound-layer formation was found to be broken into many phases with some evidence of retained lead, and a few large voids were found in the aluminum can wall which may point to poor can quality.

Compound Formation Between Uranium and AlSi. A study is being made of a standard canned fuel element in an attempt to identify the compounds formed between the uranium and the AlSi bonding material. Metallographic results and a diamond pyramid hardness traverse of the zone indicate that there are at least three compounds in this system. An attempt is being made to obtain x-ray identification of each layer.

Hot-Press Canned Fuel Elements

Nickel Plating. The shakedown runs have been completed and uranium slugs are being nickel plated for the hot-press production test. Spot checks of the hot-press pieces have been made to determine their undercutting resistance and the characteristics of the different bonding layers. In the undercutting tests, the Hanford product was subject to attack in areas remote from the point of water penetration of the jacket, but the Sylvania product exhibited only localized attack. This indicates that the nickel-uranium bond of the Sylvania pieces was much stronger than that for the Hanford pieces. Microscopic studies of the two products show the uranium surface on the Sylvania piece to be quite smooth with little foreign material between the uranium and nickel. The uranium in the HAPO piece is extensively etched with foreign material present beneath the plate. The foreign material is supposedly oxides, hydrides, and/or residual oxidized uranium not removed after the etching operation. Microscopic examination of the surface of uranium which had been anodically etched for varying lengths of time in the HAPO light etch bath, indicated that a 2-minute etch would produce a lightly etched surface to which the nickel could key yet virtually eliminating inclusions within or under the plate. Undercutting tests are now being made to determine whether this 2-minute etch produces an improved bond. If it does, all pieces for the production test will be etched in this manner.

Measurements of the thickness of the nickel plate have shown that the racks which had been used gave unsatisfactory distribution. To ensure a 0.5 mil minimum plate it was necessary to plate on an average thickness of 1.2 mils. During the past month, racks have been redesigned to increase the distance between pieces and minimize excessive shadowing. The pieces plated with the new rack show improvement, with the average-to-minimum ratio decreased from 2.4 to 1.6.

Plating Bath Control. The nickel plating bath is extremely sensitive to contamination by nitrates which are present as nitric acid in the pretreatment processes. Apparent nitrate contamination, which is first evident on the Hull Cell test plate at concentrations near 0.09 g/l is being controlled by low cathode current density electrolytic purification with continuous charcoal purification. The cathode current density is 1 to 1.5 amps/sq. ft. with electrolysis totalling approximately 0.2 ampere hour per liter of bath. This permits purification during 16 hours and plating during the regular work day.

Inspection. It has become increasingly evident that some of the more important quality tests used in the inspection of standard hot-dip canned production slugs are not suitable for evaluating the quality of hot-press canned or vacuum-canned slugs. The results of alternate tests, both destructive and non-destructive, often fail to correlate with those of the standard tests. In some cases it has been possible to modify the standard tests for more appropriate application to the new types of fuel element. For example, the inner-tube autoradiograph method developed for production use was adapted to require a minimum of storage space during autoradiographic exposure, thus permitting a more comprehensive examination of the

product. For the most part, however, pending the development of more appropriate non-destructive tests, it is still necessary to rely largely on destructive metallographic examination of representative samples.

Unbonded Uranium Fuel Elements

The heat transfer coefficient of the uranium to aluminum contact zone was measured in an unbonded four inch slug. The slug was heated axially by a graphite resistance heater running through the cored center of the slug. It was cooled externally with flowing water. Coefficient values (h) of 1000 to 3000 BTU/hr/ft^2/^{o}F were obtained at slug powers of one to seven KW/ft. This range of values represents scatter of data and not an observable power dependence. A value of 6000 BTU/hr/ft^2/^{o}F is currently used as the best available approximation of the "bond" coefficient of an unbonded slug. The values of 1000 to 3000 quoted are only tentative and subject to confirmation by additional tests and evaluation of sources of error.

I and E Fuel Elements

Experimental work in canning process improvement and fabrication facilities development has been carried out on four different types of fuel element. These include (1) the hot-press canned I & E slugs (2) the vacuum canned AlSi bonded I & E slugs (3) the hot-dip canned, AlSi bonded segmented I & E slug and (4) the self-supported slug with welded-on supports. With the goal of getting four tubes of each of these fabricated, inspected, and ready for reactor evaluation by the end of the third quarter of FY-1957, the processes for the first two types were frozen as of November 1, and tne fabrication and testing of the four-tube Production Test lots were begun. Production of these two types of fuel element is proceeding on schedule.

The irradiation of an eight-inch I & E wafer fuel element in the HAPO fuel element testing facility in the MTR continues without incident. Discharge is scheduled for December 17, at which time the exposure is expected to be approximately 700 MWD/T. The irradiation began September 28. Thermocouples in the cooling water immediately above and below the specimen indicate a temperature rise of about 9 C which, at the measured flow rate of 19.7 gpm, corresponds to a specific power level of 70 KW/ft.

Wafer Fuel Elements

Experimental process development is still under way on the vertically segmented or "wafer" slug and on the self-supported slug. With the former a current objective is to develop procedures for producing uranium washers of suitable geometry and crystallographic structure at a reasonable cost, preferably competitive with that of solid I & E cores. FMPC has supplied a lot of stamped washers that appear satisfactory upon preliminary examination. Further efforts are being devoted to the development of improved tools to facilitate the canning assembly of the wafer slug components.

Cored Fuel Elements

Beta-Phase Transformation of Cored Enriched Slugs. The examination of three 4-inch cored enriched slugs (irradiated under PT-105-607-A-57 MT) to determine the degree of beta phase transformation continued. There was no evidence of transformation to the beta-phase of U-Al diffusion. However, a positive check for diffusion could not be made because a large portion of the uranium surrounding the aluminum plug had corroded away during storage in the 105-B basin.

Axial Temperatures of Insulated and Unbonded Cored Slugs. Previous experiments in which axial temperature measurements of insulated and unbonded, cored uranium slugs have been made has provided information on performance of such fuel elements under MTR operating conditions. A similar thermocouple has been installed in an unbonded, cored, uranium slug to determine temperature asymmetries in the core and, hence, power or flux asymmetries in the operating fuel element. Two thermocouples at 180° from each other in the cored slug should record the ΔT from one side of the fuel element core to the other that may be caused by a flux gradient. This experiment will be run in cycle 80 in the MTR.

Thermocouple-Uninsulated Slugs

Metallographic examination of a wafer from the bonded area of the thermocouple slug (GEH-4-12), showed the bonding to be the result of mechanical interlocking of the can and slug surfaces during the sizing operation. No diffusion bonding was evident.

Cluster Fuel Elements

Fabrication of two lucite dummy seven rod cluster elements has been completed and preliminary flow line determinations started. From these preliminary tests, it may be necessary to pin the fuel elements to assure alignment of rods in order to assure ample flow in the KE thru-hole facility. These tests are being performed by the means of a Fluid Polaroscope. Pressure drop tests should be completed by the end of December.

Fabrication of welding jigs and fixtures has been completed in preparation for the assembling of seven rod cluster fuel elements. Inert arc spot welding was proven to be a better method of fabrication of clustered elements than the previously used resistance spot welding. A more efficient use of this new method will be explored during the coming month.

Fuel elements to provide thirty feet of charge for KER operation were completed. These fuel elements are of four rod cluster design, clad in stainless steel. As fabricated, they will fit into a ribbed tube. Supports have been made so that these fuel elements may be adapted to ribless tube operation if desired. Fuel elements of external ring four rod cluster design will be completed in December.

Determination of the space-filling characteristics of rod cluster fuel elements has continued. Contour maps of the fraction of the tube occupied by cladding and by fuel, as a function of tube size, cladding thickness and rod separation have been prepared for four, seven, thirteen and nineteen rod cluster elements.

A comprehensive mathematical study of surface heat distribution in inverted cluster fuel elements is being processed on the IBM 650 computer. The first of two portions of the cases without a co-axial cooling channel is partially completed.

Self Supporting Fuel Elements

The feasibility of attaching supports of several types to previously canned and inspected slugs by various welding procedures has been the subject of considerable experimentation. The results of the various charging, handling, and failure-rate tests lead to an inclination toward collapsible supports which will fold up under excessive stress (such as an underlying swelling due to jacket leakage), rather than the solid type of support which permits only slight slug swelling before seizing in the tube. Document HW-46340, "Status Report on Projection Fuel Element as of October 10, 1956", describes the latest developments in this field. Canning of uranium cores in aluminum cans with integral ribs has also been accomplished.

Coaxial-Tube Fuel Elements

Coaxial-tube fuel elements are of interest for present HAPO reactor application because they offer promise of having the increased split failure resistance of the larger ID cored element without the accompanying reactivity deficiency. A specimen is being fabricated for irradiation testing in the HAPO fuel element testing facility in the MTR. The specimen will consist of a uranium tube of 1.336" OD and 0.750" ID with a 3/8" ID tube inserted in it (a press fit). Uranium end plugs will be welded in place and the uranium assembly dip braze canned in the conventional manner. The dimensions were picked so that at the power level of the testing facility, the outer tube will operate in alpha phase and the inner tube will operate in the low gamma phase. The irradiation is currently scheduled to begin in January 1957.

PHYSICS AND INSTRUMENTATION

Reactor Lattice Physics

Physical Constants Test Reactor. A dry 7-1/2-inch lattice containing 1.68-inch solid slugs was studied. Results indicate that the infinite multiplication factor of this lattice is about one per cent greater than unity. More experimental data will be required before a quantitative result can be given.

Extensive measurements were made with 1.36-inch slugs in both 6-1/2-inch and 9-1/2-inch lattices. These studies were made to give further insight into the interpretation of the k_∞ measurements performed in the PCTR. They also include one series of experiments in which U-235 aluminum alloy elements were used in the buffer.

Exponential Experiments. Buckling measurements were completed with 0.94 per cent enriched uranium. This uranium was in the form of solid slugs, 1.34 inches in diameter and 6 inches in length. The canned diameter and length was 1.44 and 6.375 inches, respectively. The current results for the 0.94 per cent material are given below.

Buckling of 0.94% Enriched Uranium
(1.34-inch, solid, fuel elements)

Lattice Spacing (inches)	Al/U	C/U	Cooling Annulus Condition	H_2O/U	Buckling (10^{-6} cm^{-2})
5-3/16	0.61	32.3	dry	-	112
	0.61	32.3	wet	0.198	210
10-3/8	0.61	138.5	dry	-	258
	0.61	138.5	wet	0.198	203

Theoretical calculations have indicated that as the uranium enrichment is increased the lattice spacing at which the dry and wet buckling values are equal is increased. Information of this kind is of interest for possible application to enriched, fringe-driven, reactor types. Measurements with 1.44 per cent enriched uranium were undertaken to check the theory on this point.

The 1.44 per cent uranium was in the form of I and E slugs, 1.37-inch O.D., 0.48-inch I.D. and 8 inches in length. The canned dimensions were 1.47-inches O.D., 0.375-inch I.D. and 8.63 inches in length. The measured bucklings with the I and E slugs, together with the atom ratios for the cell, follows.

Buckling of 1.44% Enriched Uranium
(1.37-inch O.D., 0.48-inch I.D., I and E elements)

Lattice Spacing (inches)	Al/U	C/U	Cooling Annuli Condition	H_2O/U	Buckling (10^{-6} cm^{-2})
8-3/8	0.60	98.2	dry	-	588
	0.60	98.2	wet	0.341	585
5-3/16	0.60	35.5	dry	-	459

The amount of water in the coolant annuli is 3.76 cc/cm of tube length, with 81 per cent of the water in the external annulus. The H_2O/U atom ratio is 1.78 times larger than that for a 1.36-inch solid slug in B, D, and F type process tubes. The small difference between the wet and dry measurements seems to indicate that the cross-over point shifted to higher lattice spacings, but further measurements will be required to affirm this. The lattice spacing at which the wet and dry buckling values are equal for a 1.36-inch, solid, natural uranium slug has been estimated to be $\simeq$6.9 inches for a H_2O/U of 0.19.

High Temperature Exponential Experiment. The 7-1/2-inch lattice pile, which had been heated to 460C, has been cooling down. Buckling measurements were 78.6×10^{-6} cm^{-2} at 200C, and 87.6×10^{-6} cm^{-2} at 100C. These values are several microbucks (10^{-6} cm^{-2}) lower than those obtained at the same temperature while the pile was being heated. This is not unreasonable, because as the pile cools down voids will occur in the lattice because of the expansion.

Light Water Moderated Lattices. A series of buckling measurements has been performed in the water-moderated lattices using enriched cored fuel elements with granular bismuth in the core. These experiments were carried out to investigate the effect of neutron streaming in the assemblies previously measured with air in the core.

The 1.44 weight per cent U-235 fuel elements used in these studies had an O.D. of 1.37 inches and an I.D. of 0.48 inch. The elements were positioned in Type 3-S aluminum tubes (1.50-inch O.D. and 0.049-inch wall thickness) for insertion in the lattice assemblies. The table below compares the values of the buckling for the cases of the air core and bismuth core at two water-to-uranium volume ratios. The 2.2-inch lattice is near the region of maximum buckling. The bismuth in the core had an effective density of 5.5 gm/cm^3.

Lattice Spacing (hexagonal geometry)	H_2O/U (by volume)	Buckling (10^{-6} cm^{-2}) Air core	Bi core
2.0"	1.31	5203	5253
2.2"	1.87	5741	5730

On a relative scale the uncertainty of each buckling is about $\pm 30 \times 10^{-6}$ cm^{-2}. These results show the bismuth in the core to have little effect on the buckling. The calculated change in the thermal utilization is negligible between the air and bismuth core cases. Unless the age (ν) or fast effect (ϵ) are appreciably changed by inelastic scattering in the bismuth core, actual streaming effects on (ν) are small, i.e., the bismuth should have prevented any streaming which would otherwise have taken place.

The possibility remains that a reduced ϵ and P (resonance escape probability) are just compensated for by a decrease in ν. Further theoretical analysis of these measurements will be done.

Reactor Theory. Disadvantage factors (ratio of flux at the slug surface to the average over the interior) have been calculated for the enriched fuel elements utilized in the exponential program. This was accomplished with the P_3 program on the 702 computer. The results of these calculations are given in the table below.

Fuel Element O.D.	Fuel Element I.D.	Weight % U-235	Disadvantage Factor
1.336"	-	0.94	1.353
1.660"	-	1.007	1.530
1.660"	0.94"	1.007	1.245
0.926"	-	1.007	1.201
1.370"	0.48"	1.440	1.414

The thermal utilizations have been calculated for a series of water lattices with a 1.37-inch O.D., 0.48-inch I.D. fuel element. The fuel element was enriched to 1.44 weight per cent U-235. The fuel elements were loaded into 1.50-inch O.D., 0.049-inch thick, Type 3-S aluminum tubes for positioning in the lattice assembly in the water tank. The tubes were filled with water which resulted in a thin water annulus between the fuel element and the tube wall. The core of the fuel element was filled with water. The results of these calculations are given in the following table.

Lattice Spacing (hexagonal geometry)	H_2O/U (by volume)	f_u	f_w	f_{al}
2.0"	1.51	0.8967	0.0973	0.0061
2.1"	1.79	0.8768	0.1171	0.0060
2.2"	2.07	0.8557	0.1384	0.0058
2.4"	2.69	0.8092	0.1854	0.0055
2.6"	3.36	0.7600	0.2347	0.0052

A mathematical model for the PCTR has been set up using small source theory. This model (the critical equation for which is a 3 x 3 determinant) can be manipulated in exact analogy with the physical PCTR. It is expected that results will be obtained exhibiting the shapes of experimental PCTR curves and cataloging the contributing effects.

Concern has recently been expressed by several sites on the disagreement between calculated and measured xenon poisonings. It has been shown that much of the discrepancy can be removed by accounting for self-shielding effects.

Experimental Nuclear Physics

Measurements were continued of the fission cross section of Pu-241 on the neutron spectrometer.

Development of a magnetic balance for the Plutonium Metallurgy Operation was initiated. A balance is desired which will weigh a plutonium sample surrounded by a controllable atmosphere.

Specifications were drawn up for a plutonium mass spectrometer suitable for use with the critical mass program.

Experimental Reactor Operations

Physical Constants Testing Reactor. Operation of the Physical Constants Testing Reactor continued routinely during the month. Six unscheduled shutdowns occurred. Investigation of the effects of varying the number of buffer rods surrounding the central lattice cell was continued. The first measurements were made on a 7 element cluster fuel rod in a 7-1/2-inch lattice.

Reactor improvement items provided during the month included (1) installation of the photoelectric vertical safety disk down switches, and (2) plans for partitioning the control room so that personnel in the control room can be limited to three during reactor operation.

Thermal Test Reactor. The change in reactivity caused by placing graphite or polyethylene plugs in the top of the fuel tubes was measured. A loss of 13 cents occurred when 20 polyethylene plugs 2 inches long were placed in the fuel tubes. In contrast, a reactivity gain of 70 cents occurred when 20 graphite plugs 5 inches long were placed in the fuel tubes.

Since it is planned to keep the graphite plugs in the fuel tubes, the 20 top fuel disks containing 45 grams of U-235 were removed, reducing the reactivity 46 cents. The excess reactivity of the TTR is now 40 cents, equivalent to 1/3 the strength of the two controls rods.

A 6-cent gain in reactivity was obtained when the reflector on the south side of the reactor was increased 4-3/16 inches in thickness. This indicates that the reactivity of the TTR can be changed by a change in the geometry of any external reflectors.

Radiological Physics

A shield was made of aged lead obtained from the National Lead Company to determine if it would provide a lower background for scintillation counters. There was no improvement over a shield made of ordinary lead. This may mean that the principal remaining sources of background are within the materials of the scintillation counter itself.

Insertion of a new ion source into the Van de Graaff improved the intensity of the analyzed beam considerably and apparently removed some of the troubles which had been plaguing the accelerator.

The ratio of the counts obtained with the two parts of the double moderator neutron fluxmeter-dosimeter was measured as a function of neutron energy in the range 0.025 to 5.1 Mev. Below 1 Mev the ratio is linear with energy. In this range a simple ratio measurement will determine the average energy of a complex spectrum. For a complex spectrum containing higher energies the ratio will determine a weighted average in which the lower energies are emphasized.

The scattering chamber for use in proton w measurements was installed on the Van de Graaff and tested. The chamber operated satisfactorily and exposure to the beam did not appear to damage the gold scattering foil.

A constant temperature water bath was installed at the electron Van de Graaff so that calorimetric experiments can be performed. Modifications of the accelerator shielding were necessary but will not interfere with routine operations. Controls of the bath were tested and worked sufficiently well to permit experiments to begin. Design of a calorimeter to be used in this system to make differential measurements of w for electrons was completed. Infra-red heat lamps were substituted for the knife blade heater on the water bath used in the Co^{60} calibration to see if using a distributed heat source would improve bath temperature controls. Use of the lamps and an improved stirring system reduced the bath temperature variations to one-fifth their former value, to 0.0004C. It was verified experimentally that the temperature variations in a calorimeter due to variations in the bath can be computed using an electric circuit analogy. Both amplitude and phase are given correctly.

The current from a carbon wall ionization chamber being irradiated with X-rays was measured as a function of air pressure in the chamber. These data will be combined with similar data for aluminum and copper chambers to demonstrate the lack of effect of chamber size for air wall ionization chambers.

The difference in response to beta rays of film badges and the modified CP was shown to be due to the difference in the amount of absorbing material between the source and the sensitive elements in the two devices.

Atmospheric Physics

Meteorological and climatological summaries for the 100-F and 300 Areas were prepared. These summaries contained meteorological data pertinent to reactor hazards appraisals and were designed to permit comparison of these Areas as prospective sites for the PRPR. Since temperature stratification and precipitation data are not normally collected at these sites, this information was drawn from observations made at the Meteorology Tower.

A field test designed to measure the dust loading and the temperature and wind stratification up to 40 feet above the ground was completed. This experiment was conducted at the Benson Ranch site using the Portable Mast equipment. The data have been reduced and prepared for machine analysis.

Operation of the wind station network and reduction of the data collected from this network continued on a routine basis.

Weather Summary

Type of Forecast	Number Made	% Reliability
24-hour General	60	85.2
8-hour Production	90	86.0
Special	53	88.7

Barometric pressure averaged higher and 50-foot wind speed averaged lower than during any previous month in 12 years of record. The temperature average of 36.6, 3.3% below normal, was considerably higher than in November of 1955. Although there was considerable cloudiness during the past month precipitation totaled only 0.15 inch. This is less than 20% of the normal amount for November.

Instrument Research and Development

The power supply and beam deflection tube amplifier in the ten-channel analyzer were modified to provide greater stability and higher sensitivity. Using a pulse generator with synchronous motor-driven amplitude adjustment, it was found that the channel widths could be equalized to within 3.0% with little difficulty.

Tests with three RCA 6810 fourteen-stage photomultipliers showed that pulse-height spectra of gamma emitters with energies above 100 Kev could be resolved without the aid of an amplifier preceding the pulse-height analyzer circuits. By making use of an amplifier both the 17 Kev Pu^{239} X-ray peak and the 5 Kev peak of Fe^{55} were determined.

A change from powdered anthracene to a one millimeter thick NaI crystal has multiplied the sensitivity of the Scintillation Dose-Rate Meter by a factor of about fifty. The low-range full-scale reading is now 0.5 mr/hr. By inserting a silicon diode in the amplifier input, the meter was made to indicate logarithmically over a range of 0.5 mr/hr to 5.0 R/hr.

The electrical part of the Water-Direction Indicator is completely designed including the directional gyro drive supply. The apparatus for introducing electrolyte into the water volume under test will be fabricated.

The Automatic Scanner mechanism was assembled and provided with slide-wire voltage dividers. An "Autograf" X-Y recorder actuated by the slide-wire brush voltages charts continuously the movements of the scanner carriage upon which a scintillation detector is to be mounted. Scanning speeds may be adjusted over the range 0.5 to 15 inches/minute.

The problem of installing a lead glass window at the location of the presently unused Fly Eye in 105-F Bldg. is being considered. Among the suggestions being considered are (1) a thick lucite and lead glass assembly (2) a minimum thickness lead glass requiring removal of a part of the concrete shielding (3) a wide-angle periscope and (4) a modified form of the original Fly Eye.

Evaluation tests were completed on a bench model of the completely transistorized battery scintillation poppy equipped with a counting-rate meter circuit. Evaluation tests were begun with a fast and slow neutron detector and an investigation was started comparing the characteristics of halogen quenched with organically quenched mica window tubes.

REACTOR TECHNOLOGY

Reactor Optimization Program

An IBM 650 program is being prepared to furnish input punched cards for the current reactor study employing the IBM 702. This study is concerned with the effect of fuel element variables on reactor performance and is a part of the reactor optimization progam.

Fuel Element Test Reactor

A preliminary study of applicability and feasibility of a fuel element test reactor is near completion. Additional prescoping is necessary before recommendations can be made as to the reactor type which would be most beneficial.

Plutonium Recycle Program - Reactor Design and Engineering Studies

Schedule. No changes were made in the PRPR schedule during November. Design and Research and Development are on schedule.

Site Study. The site study for the location of the Plutonium Recycle Program Reactor was completed. A study stage cost estimate of those factors which would be affected by the site location revealed that the cost at a 300 Area site, with a reactor building incorporating containment features, would be about the same as the 100-F Area site, with a non-containment reactor building.

Building Structure and Arrangement. The basic concept of using two cylindrical sections for the process equipment systems remains basically the same. The number of compartments has been reduced from eight to six while the building area has been increased. Preliminary cost indicates the later arrangement may be more expensive, due primarily to the additional 4000 square feet of floor space. Definite building area requirements will be refined further when more equipment sizes are determined. Two arrangements for the service area of the reactor installation have been made. The first scheme distributes the service area around the containment vessel, whereas the second scheme combines all the functions in a rectangular building on one side of the containment vessel.

Reactor Core and Shielding. Two preliminary scope drawings of the reactor core and shielding have been completed. Drawing SK-1-6171 is a horizontally sectioned drawing through the reactor and shows the calandria, graphite reflector, and primary shielding. A vertical cross section through these reactor components is incorporated in Drawing SK-1-6170.

The design of the core and shielding presents an economical and workable arrangement which should require a minimum of development work. Pile testing of the ferrophosphorous and water top and bottom shield material is required. Although considerable data is available concerning the use of barite in reactor shields, samples of barite will be procured and experimental work initiated at HAPO to obtain design data and experience under local conditions. From these data, refinements in shielding computations will be performed to evaluate the optimum thickness of the various shield components.

Calculations have been made to determine the thicknesses for the various types of shields. Calculations have also been completed for the primary shield for the sides and top of the reactor. Preliminary calculations have been made for the secondary shields around the piping and the cask for unloading fuel elements.

Because radiation levels are dominated by water activity wherever pipes or pumps are present (top and bottom halls, pump rooms, etc.) savings in shield costs can be realized by having the primary shield only thick enough to prevent activation of the materials (primarily steel) in the shielded area. Present thinking is that the flux level must be dropped to 10^4 neutrons/cm^2-sec to prevent activation of the materials in the shielded area. However, there are indications that the flux levels can be 10^6 without causing significant activation. If attenuation to 10^6 rather than 10^4 is all that is necessary, then shield thickness for these areas can possibly be reduced by one to two feet. Measurements in the DR test well will be made to establish the permissible flux level.

Primary Cooling System. A steam generation system has been selected as the means of dissipating primary reactor heat. This system will be composed of two parallel units operating on the secondary side at 425 psig. Auxiliary heat exchangers, such as the moderator cooler and the purge cooler, will be used to preheat the feedwater.

Standby power will be (1) BPA and (2) power generated from steam turbine driver generators supplied by a steam boiler. Turbine exhaust will serve as building heat with the heat exchangers piped to the river. Work is continuing on the inert gas system and the deuterium oxide treatment system.

Preparation of technical bases for the PRPR coolant system continued. The primary areas under study are (1) water quality requirements (2) system materials compatibility and (3) radioactivity and decontamination problems.

All necessary equipment for installation of a vertical test section in ELMO-7 has been placed on order. Addition of heaters to provide for corrosion tests under boiling conditions is being considered.

Reactor Coolant Piping. The inlet and outlet headers will be single-piece carbon steel rings about twelve feet in diameter, each connected to the primary cooling loop by two lines on opposite sides of the ring. The inlet header will probably be located against the wall of the lower reactor room, outside the ring of moderator dump pipes. At present, it appears necessary to provide a valve in the jumper to each process tube so the tubes can be isolated and drained for replacement. A venturi flowmeter will also be located in each inlet line. Since all charge-discharge operations will be performed from the top face, the inlet end of the process tube has been tapered to reduce D_2O inventory and process tube costs, making it possible to connect the inlet jumper directly to the process tube without benefit of a nozzle. Several bottom face gas seal arrangements are proposed, each permitting removal of the process tube and fitting through the top of the reactor. The outlet nozzle is fastened to a support tube keyed to the top biological shield to allow for the high torque required in tightening nozzle nuts and caps. Outlet jumpers are connected between the nozzles and outlet header. It is planned to make the outlet nozzles and all screwed connectors of stainless steel for ease and safety of assembly.

Control Systems. Analysis of the dynamics of moderator dumping indicates that the reactor can be shut down easily within one second of receipt of a scram trip signal. In the analysis of the dump behavior, an initial period of 0.25 second was assumed as the "delay time" entailed in instrumentation and mechanisms. Following this initial period, 0.05 second is required to open the fast-acting dump valves and equalize the gas pressure above and below the moderator.

The D_2O moderator in the tank begins to spill immediately into the annular weir box at the base of the calandria. After 0.2 second (a total lapsed time of 0.5 second) the level of D_2O in the tank will have fallen 0.6 feet, and the rate of fall will be about five feet per second. The moderator will fall two feet in a total lapsed time of about 0.7 second. Under normal operating conditions, this represents a reactivity decrease of about 250 in-hours.

When the moderator level has fallen two feet, the weir box will have filled, and the discharge rate will be governed by the flow through the lines draining the weir box. The rate of fall of moderator level will then be 1.5 to 2 feet per second, decreasing the reactivity by 300 to 400 inhours per second.

Bead chains of zirconium-hafnium alloys are being considered as an alternate to braided tapes or cables for use as shim control elements. Evaluation of tape construction and of the suitability of the alloys for the control element is being performed.

An ink system for shim control has been under consideration but preliminary cost estimates showed such a system to be considerably more costly than a mechanical tape or chain system.

Moderator Purification. Engineering study of the heavy water purification (light water removal) problem for the PRPR was continued. Hydrogen exchange, amine extraction and electrolytic methods are being studied. The electrolytic method appears most favorable.

Heat Transfer. The dimensions of a proposed fuel element were calculated. The fuel element consists of a central cylindrical rod surrounded by two annular hollow rods, each rod being of UO_2 canned in zirconium. The design is chiefly novel in that it requires simultaneous balancing of heat flow and water flow in each of the three flow passages.

Development work is under consideration that will determine the water temperature distribution in cluster elements and provide some information on heat transfer coefficients in the system.

Charge-Discharge and Material Handling. Fuel elements will be discharged normally into a lead cask which will transport them to a water pit at the edge of the reactor hall. A chain conveyor will carry them through an under water tunnel into the storage basin outside the containment vessel. In event of a stuck fuel element, (either initially or part way out of the tube) an auxiliary saw will be lowered from the cask to cut off the nozzle elbow and permit removal of fuel, tube, outlet nozzle and support tube as a unit.

Mechanical Development. The proposed use of a conventional type pump having double shaft seals instead of a canned motor pump in the PRPR flow system has necessitated a study of mechanical seals capable of operating at the required temperature and pressure of 275 C and 1000 psi. A testing program expanding the original program of mechanical seal testing for organic collants is underway.

A proposed program covering the development of the mechanical aspects of the PRPR for the remainder of FY 57 and projecting into FY 58 was prepared. This program included the testing of the outlet nozzle in FY 57, the gas seals in FY 57 and 58, the inlet fittings and connectors in FY 57, the complete process tube assembly in FY 58, the charge-discharge procedure in FY 58, the flexible control tapes in FY 58, and the moderator controls in FY 58.

Out-of-Reactor Prototype Tests. The scoping of a prototype facility containing a complete process tube assembly and a prototype of the proposed double shaft seal pump has been completed. This facility will be used not only for tests on the individual components of the flow systems, such as the primary pump, process tube, nozzles, seals and caps, and connectors, but for the testing of the entire system to insure compatibility of the components. The location of the process tube in a pit beneath the floor level of the 314 Building will permit the use of the facility for charge-discharge tests to accomplish the development of the PRPR charge-discharge system.

Delivery of the prototype pump is estimated during December, 1957, thereby allowing the facility to be in full use by the end of 1957.

Separations Facility. Preliminary scoping of a separations facility is underway. This facility could be constructed in close association with the reactor facility if deemed desirable or necessary at a later date.

To permit simpler separations facilities in the PRPR, suggestions were made relative to the preferred composition of the recycled fuel element. The use of a mixture of plutonium and uranium oxides instead of the presently considered aluminum-plutonium alloy was suggested. This suggestion is supported by the following factors (1) an oxide element should be cheaper and simpler to fabricate from separated plutonium than the alloy element (2) chemical dissolution should be simpler by avoiding the gas evolution and treatment problems associated with alloy dissolution (3) the incorporation of partially depleted uranium provides for most efficient use of uranium and more complete burnout of the U-235 (4) waste treatment and disposal should be simpler and less costly and (5) a mixed oxide (Pu-U) fuel will provide more uniform and stable thermal conditions for reactor operation because of the adjacent uranium and residual U-235.

Plutonium Recycle Program - Reactor Physics

Shim System Reactivity. A two-group calculation of shim control strength was begun and partially completed. As an example of the order of magnitude of the results, a lattice of 24 black cylindrical rods with diameters of 0.53 inch or black strips or tapes with widths of 0.56 inch were found to be worth 6.3 % k. This is slightly higher than the optimum shim strength based upon estimates of

requirements for xenon override, experimental flexibility and reactor safety. Calculations are proceeding to determine specifications of systems with smaller or weaker elements which will provide a maximum strength of the order of 3-4% k.

Vertical Flux Distribution. The effect on the vertical flux distribution of varying the top reflector thickness is being determined by using the two group method of calculation.

Moderator Temperature & Dilution Studies. The effects resulting from a change in moderator temperature or from dilution of the D_2O with small amounts of H_2O are discussed in HW-46679. The results show small reactivity effects resulting from a less than 5% dilution by H_2O. There is a strong negative moderator temperature coefficient primarily as a result of the density coefficient for the D_2O.

Material Balance in Uranium-Recycled Plutonium Systems. Analytical expressions for the isotopic concentrations as a function of exposure have been derived for U-235, U-236, Pu-239, Pu-240, Pu-241, and Pu-242. U-235 is assumed constant for purposes of the present study. Cases under investigation are (1) fresh natural uranium feed, enriched with plutonium; uranium and all fission products rejected at completion of cycle (target exposure); part or all of the plutonium derived from the rejected uranium fed back as enrichment for the subsequent cycle; recycled plutonium rejected at a predetermined exposure (2) same as case (1) except uranium is also recycled and (3) same as case (1) and (2) with blending of the plutonium.

To maintain the same heat generation rate, within limits, as adjacent uranium, the plutonium must be re-concentrated many times during a cycle as defined in (1). However, if there are no resulting losses of material, this does not affect the calculation.

As a starting point, the isotopic concentrations in case (1) are being calculated as a function of exposure for 4000, 5000 and 6000 MWD/T plutonium. The reject exposure level for plutonium is taken as three times the reject exposure level for uranium, but this will later be varied.

The next step will be to relate the isotopic concentrations to reactivity in each of the above cases.

Plutonium Recycle Program - Fuel Element Studies

Ceramic Fuel Studies. Preliminary experiments were conducted to determine the feasibility of cladding uranium oxide tubular fuel elements with zirconium alloy by swaging methods. As a first attempt, alumina was used as a stand-in for UO_2 and concentric stainless steel tubes as the cladding material. The results appeared sufficiently favorable to warrant further pursuit of the method. Future work will involve using thinner walled tubes and swaging over a series of mandrels to attain the desired degree of compaction.

Uranium oxide tubes 1. 22" ID, 2. 29" OD and up to 2" long were prepared by compaction in a die. Densities up to 92 per cent of the theoretical density of uranium oxide were obtained by sintering the tubes. A technique was developed for preventing the cracks which previously were observed in compacts prepared

by cold pressing. The technique consists of alternately applying and releasing a gradually increasing pressure until the desired maximum pressure of approximately 40,000 psi is reached. This procedure presumably allows trapped air to escape from the piece before the maximum pressure is attained, resulting in a more compact, stable product.

Uranium oxide as received from the Mallinckrodt Chemical Works was ball-milled twenty-four hours using Burundum balls and a rubber-lined mill. Carbowax 20 M in methyl alcohol solution was added to give a 1.5 w/o mixture. The alcohol was evaporated leaving a uniform mixture of uranium oxide and binder. Compacts having higher green and fired densities were obtained when uranium balls were substituted for the Burundum balls. The powder obtained by using the uranium balls was free-flowing. Densities obtained by compacting the powder milled with Burundum balls ranged from 80 to 84 per cent whereas powders milled with uranium balls ranged from 89 to 91 per cent. Only a small decrease in fired density was observed when the length of the pressed tubes was extended from 1/4 to 2 inches. No serious warping or sagging of the pieces occurred upon sintering at 1750 C in hydrogen for three hours. In general, pieces containing no visible flaws before firing were equally satisfactory after firing. Small cracks which occurred in some of the pieces pressed before a satisfactory pressing technique was developed did not grow larger during sintering, nor did they appear to heal. Inside and outside diameter shrinkage of approximately 11 per cent occurred during sintering of tubes fabricated with the uranium-milled powder. Length shrinkage was proportional to length. It was 8 per cent for 0.5 inch pieces and 9.5 per cent for a 1 inch piece. Additional work will be done to determine more accurately the sintering shrinkage and to establish the optimum length for tubes prepared by compacting.

Extrusion is being investigated as an alternate method for the fabrication of UO_2 fuel elements. Previously extruded UO_2 rods had sintered densities approximately 89 per cent of the theoretical density. During the past month UO_2 having a density 93 per cent of the theoretical after sintering was extruded in the form of a rod, using 1.7 per cent Carbopol 934 (a water sensitive gum) as a plasticizer. Although this UO_2 had a higher sintered density than material previously extruded, it was difficult to extrude and sinter. The effects of die design, extruding pressure, particle size, and various plasticizers on the appearance and density of extruded and sintered rods will be investigated further.

Slip casting of UO_2 fuel elements was investigated as a means of producing UO_2 bodies having uniform densities, a variety of shapes, and requiring low initial capital outlays. Although a slip with good casting properties was developed, numerous fractures occurred during sintering. Addition of one per cent TiO_2 reduced fracturing and improved the sintered strength of the material but bloating decreased the density of the ware. This bloating may have been caused by a liquid eutectic phase of TiO-Ti_2O_3. The water content of the slip before casting was observed to have little effect on the sintering properties of the oxide. Much effort would apparently be required before satisfactory slip cast fuel elements could be produced. This work is being temporarily discontinued in order to apply increased effort on extrusion and other phases of UO_2 fabrication which appear more promising.

Plutonium Fuel Studies. Study continued on various types of plutonium fuel elements for the PRPR. Consideration was given to (1) liquid systems (2) fused salt systems (3) systems employing plutonium carried on various solids (solid and granular) and (4) plutonium metal alloy systems. Little information is available for specific plutonium-aqueous solution systems although data for uranium and thorium has permitted analogies of value to the plutonium case. Each of the cases enumerated above have individual characteristics that may be advantageous in specific plutonium recycle reactors.

Two high temperature autoclaves for testing plutonium-bearing fuel rod samples up to 30" long have been ordered. This equipment will permit evaluation of jacket integrity and corrosion characteristics of fuel capsules at 300 C.

A steel mold for simultaneously casting six aluminum-plutonium fuel cores, 12 inches long will be ready for use in December. Experiments will be made with aluminum and U-238 alloys before working with the Pu-Al castings.

Two fifty-ton capacity, hydraulic presses with accessories were acquired. One of the units will be used to conduct preliminary extrusion experiments. It is hoped that extrusion techniques can be used for fabricating the PRPR reactor fuel elements.

Process Specification and Control

I & E Thermal-Hydraulic Tests. Study of flow instability characteristics of I & E slug columns continued on the low pressure electrically heated mockup. Testing was limited by shutdown of generators for necessary repairs.

The tests pertained to an eccentric geometry wherein an over-sized heater rod in a C reactor process tube was used. The eccentricity led to initiation of regional local or bulk boiling at the downstream end on top of the annulus. This tended to cause increased pressure drop at higher annulus flows than with a concentric geometry. Tests were run simulating selective hold and annulus plugging as well as general plugging of both channels.

Of greatest interest are tests simulating selective stream plugging, since Panellit sensitivity to these is reduced and represents the major novelty concerning I & E slug flow monitoring. The eccentricity effects do not appear to alter instability characteristics, at least up to tube powers of 1000 kw. As had been found with the concentric tests, heat shift from the plugged to the unplugged stream lessens the tendency of the plugged stream to become unstable. Thus, stable flow regimes exist wherein flow in the plugged stream is boiling. For example, at tube powers below 1000 kw flow remains stable at all degrees of hole flow reduction, including complete shut-off.

From a flow monitoring standpoint this indicates that trips are probably not obligatory for conditions where one channel is at incipient bulk boiling. Analysis has not proceeded to the point of verifying the boundaries of adequate cooling under conditions where bulk boiling is occurring in just one stream. Verification of this effect at higher tube powers is in order because at high tube powers smaller flow reductions will lead to bulk boiling.

Tests wherein both streams were plugged to the same extent indicated a modest distortion of typical pressure drop versus flow relationships. The distortion is power dependent, being greater at high tube powers.

The services of a manufacturers' engineer were procured and led to correction of recent difficulties with the direct current generators. By effecting recommended changes it was possible to operate the generators at significant overload. Sustained output of 1350 kw was demonstrated. With alteration of heater rod design, higher power outputs are expected for brief but practical working periods. While the rated capacity of the equipment is only 1125 kw, tests at 1250 kw are now possible with present heater rods.

Reactor Process Assistance

When cavitation occurs within an orifice assembly, tube flow rate is not sensitive to changes in downstream flow resistance. If cavitation is severe, it can cause failure of pigtails. Furthermore, cavitation affects Panellit response to flow changes, but in a manner which depends on the location of the point of cavitation with respect to the Panellit pressure tap. A potential hazard exists for the case of pigtail or front nozzle cap failure for reactors such as K and those undergoing CG-558 modifications. A similar situation would appear to exist at the older reactors for the case of front nozzle cap failure. If the tube were normally operating with or near incipient cavitation in the secondary orifice and the front nozzle cap failed, flow through the orifice assembly would increase only to that controlled by cavitation in the secondary orifice. The Panellit pressure will decrease as the flow increases. The flow increase can be small if the tube is initially near cavitation and may not be sufficient to initiate a reactor scram by a Panellit low trip. Although generally confined to extreme fringe tubes, this is an undesirable situation.

Using solid aluminum dummy slugs to suppress cavitation appears possible but requires some means of predicting when cavitation will occur. Cavitation in orifice assemblies has been studied. Experiments were performed with CG-558, C and K Reactor orifices and tests will be run with old reactor orifices. Review of experimental results led to derivation of formulas enabling approximate prediction of Panellit pressures corresponding to incipient cavitation. Cavitational tendencies depend on geometry of the leading edge of the orifices. Of all flow tube geometries tested, those with conical entries are superior to those with rounded entries. The conical entry design is in common use.

For the CG-558 reactors, tests indicated the configurations now installed in B Reactor to be sufficiently cavitation resistant for interim use but probably undesirable for film free or seriously corroded fringe tubes. A configuration devised for the DR Reactor is more resistant to cavitation and makes use of rounded entry orifices. A conical entry configuration is suggested for use at other reactors. It is more resistant to cavitation than the rounded entry configuration used for DR Reactor and can more easily be drilled to the new diameters as needed for orifice changes resulting from new equilibrium filming tendencies.

Testing of K Reactor orifices identified empirical constants for predicting approximate Panellit pressures associated with incipient cavitation. In general, these appear to be above low trip pressures but some tubes may require special treatment. As corrosion proceeds in the process tubes the situation will be aggravated and some special treatment will probably become necessary.

Testing of C Reactor orifices has been completed. Test data indicate the problem to be less severe than at the K Reactors. Normally loaded fringe tubes probably do not lie in the region of suspicion. Rectification of data is in process.

Shielding Studies

Miniature Fission Chamber. A miniature fission chamber to detect fast neutrons has been developed using U^{238} plated on steel. The components for the chamber have all been fabricated and will be assembled for use in conjunction with attenuation studies.

Neutron Dosimeter. The neutron dosimeter was used on the top of DR Reactor to determine the neutron dose coming from the vertical safety rods. At the shield test well, at least 82 per cent of a total dose of 19 m rem was found to be coming from the vertical safety rods.

Increased Counting Efficiency. Increased counting efficiency with sulfur foils can possibly be obtained by burning off the excess sulfur leaving a residue rich in P^{32}. This increases counting efficiency by reducing self absorption and self scatter. By using an anthracene scintillation counter it is possible to eliminate mica window thickness and thereby make still further gains in counting efficiency. By this technique it is hoped to extend the range of sulfur foils so that they can be used out to the edge of the shield for attenuation measurements.

Attenuation Studies. Counting of sulphur foils and gold foils from the first test of Magnetite concrete heated to 100 C is 80% complete. The second test is still being exposed in the DR test well. The contaminant in the CuS foils giving a 0.6 mev gamma with a 25 day half life is still unidentified.

Thermal Shield Studies. The C Reactor shield test facility was inspected to determine exactly what maintenance will be required to put the facility in an operable condition. The general outline of the thermal shield studies program utilizing the C Reactor shield test facility was discussed with the Special Irradiation Operation of IPD. It was agreed that Special Irradiations will (1) write a production test covering the proposed work (2) have the necessary maintenance work done and (3) provide the necessary personnel and supervision for the future tests.

Fuel Element Performance

Outlet temperature probes on full length loadings of I & E slugs at C Reactor have confirmed earlier data showing high water temperature at the top of the annulus. As an aid in understanding the high top-of-annulus water temperatures, test assemblies used for I & E slug studies in the low pressure electrically heated mockup were equipped with process tube wall thermocouples at several locations near the downstream end. These indicated the existence of high top-of-annulus temperatures, but

the data have not yet been analyzed in sufficient detail for conclusions to be drawn.

To appraise the relative resistances of I & E slugs and solid slugs to surface hot-spots caused by regional cooling deficiencies, an analysis was made on the resistance grid analyzer. Solid slugs were compared with I & E slugs of the same diameter. A peripheral arc of 330 degrees was cooled normally and complete insulation was used on the remainder. Temperature distribution within the slugs was determined. The maximum jacket surface temperature in the solid slug was about 350 C and about 275 C for the I & E slug.

Calculations were made to determine the relative maximum temperatures for solid and I & E slugs for several cases in which concentricity was assumed and local cooling anomalies ignored. From these it appears that the maximum jacket temperatures are sufficiently close that one must clearly define the case of interests for a comparison to be meaningful. Outlet temperatures exert the strongest influence on calculated maximum jacket temperatures for any geometry. The higher heat flux in the hole of I & E slugs results in correspondingly higher skin temperatures for the case of common flow and power, with equal outlet temperatures in hole and annulus. However, slight diversion of extra flow to the hole can lower the hole outlet temperature enough to compensate for this. Since local flow anomalies and eccentricity effects exert a powerful effect on jacket temperatures, comparison between cases ignoring these effects is not too meaningful.

Process Water Corrosion Tests

Operation of 1706-KE Water Studies Semiworks. Corrosion rates of standard aluminum jacketed fuel element are being determined in process water at pH values of 7.0, 6.5, and 6.0. Examination of five tube charges discharged in October is still in progress.

Recent autoclave tests have shown that phosphate ion is an excellent inhibitor of aluminum corrosion in process water. However, the radioactivity hazards associated with addition of phosphate ion to single pass coolant systems makes this application appear impractical. Addition of phosphate ion to a single-in-reactor 1706-KE tube has been initiated to obtain quantitative information on the increase in P^{32} concentration. Data from this test will permit a firm decision concerning the addition of phosphate ion to Hanford process water, as well as providing a better understanding of the sources of P^{32} currently present in the reactor effluent. Preliminary data indicate that at least a tenfold increase in P^{32} results from addition of 5 ppm sodium phosphate.

Raising Permissible Outlet Temperatures of Selected Tubes at C Reactor. The last tube under PT-105-519-E has been removed from C Reactor and is being held for examination. Tube 2071-C has been examined. Accelerated corrosion was observed at the top of the tube. No satisfactory explanation has been determined for this type of attack.

Evaluation of pH 7.0 Process Water. Termination of process water pH adjustment under PT-105-554-E is scheduled for the first F Reactor outage after December 15, 1956. Slug jacket corrosion data will continue to accumulate after this date.

Corrosometer probes installed in the effluent lines give an indication of the relative steel corrosion rates in pH 7.0 and 7.3 effluent water. Initial corrosion rates appear to be as much as 35 per cent higher at pH 7.0 than at pH 7.3. Equilibrium rates are about 20 per cent higher. The maximum initial corrosion rate in pH 7.0 water was 35 mils per year.

Evaluation of Reducing Process Water pH Below 7.0. A report is being prepared to program in detail the testing and development of suitable Hanford Reactor cooling water with controlled pH less than 7.0. The major advantage of the use of cooling water at these low pH's continues to be reduced aluminum corrosion. However, the disadvantages of (1) increased steel corrosion (2) increased localized (pitting) corrosio of aluminum (3) increased chemical costs and (4) increased tendency for film formation in reactor flow channels may outweigh the advantages of using low pH cooling water.

1706 KE Mockup Tubes. The test of 2S and M-388 aluminum under heat transfer conditions at 175 C has been completed. Preliminary examination showed accelerated corrosion in the vicinity of the welds. No intergranular corrosion was evident in either the 2S or M-388 aluminum. Further examination will be made after cleaning and dismantling.

One hundred M-388 aluminum jacketed slugs have been obtained for corrosion testing in the mockup facilities. Tests will be run at 115 C and 135 C in pH 6.0, 6.5, 7.0 process water.

Pressurized Water Coolant Technology

In-Reactor Tests. The Zircaloy-2 process tube from the H-Loop has been split and the visual examination completed. No untoward effects were noted. Samples over the length of the tube will be examined by metallographic techniques.

KER Program. A rough draft of the proposed KER program has been issued for comment. Startup of the in-reactor loops will probably be limited by operational considerations rather than by the tie-in date. Committed schedules for delivery of KER fuel elements are well ahead of the anticipated KER schedule.

KER Component Testing. The KER nozzle-process tube connection using a gold-plated stainless steel solid "O" ring for sealing has been cycled for 160 cycles on ELMO-7 without leaking. Based on this test, the gasket is scheduled for use as the on-reactor KER nozzle-tube connection. An unplated solid stainless steel "O" ring will be tested to act as a backup.

A KER nozzle cap using four set screws instead of the present single one to seat the solid stainless "O" ring is being cycled in ELMO-7. The seal has held for nearly sixty cycles while the single screw type cap has been leaking since the sixth cycle. Other sealing arrangements will be tested to find alternate acceptable seals.

Out-Of-Reactor Tests. The ELMO-2 loop is operating at 150 C with water adjusted to pH 3.5 with phosphoric acid. Visual examination of M-388 and M-400 aluminum alloys and several stainless steel specimens indicated no corrosive attack. Water analyses show some increase in iron and nickel concentration since startup.

In ELMO-6, similar results have been obtained at pH 4.5 and a temperature of 300 C. However, M-457 alloy (M-388 plus oxide) appears to be undergoing intergranular attack. A hydrogen detection device and a hot crud probe have been installed in the loop to provide additional information on test variables.

Decontamination Studies. The decontamination loop (ELMO-10) has been modified to provide improved safety features. Preparation of the program report for decontamination continued.

Organic Coolant Development

In-Reactor Tests. Five quartz ampoules and five stainless steel capsules containing biphenyl, MIPB, and mixtures of these two were charged into the 2A test hole of KW Reactor to determine the effect of irradiation on the stability and physical properties of these materials. Temperatures up to 200 C are being maintained on the samples Post irradiation physical properties will be determined on these samples after a one-month exposure period.

A new in-pile test section has been installed in the ORA-2 loop and operation was resumed. The preliminary activity measurements indicate that the chlorine concentration of the MIPB is 3 to 4 times that found in the charge used in the original test run.

The MIPB in the loop is expected to decompose at a rate of approximately 5% per day. A goal of 35-50% equilibrium decomposition has been established. The damaged material will be used for subsequent heat transfer and corrosion experiments.

The feasibility study of converting one KER in-reactor loop to organic coolant testing continued. Current plans propose (1) immediate conversion to organic coolant following the KER tie-in outage (2) recirculation out-of-reactor until the system functions smoothly and (3) in-reactor recirculation in the spring of 1957. Mechanical and operational problems appear to be solvable. However, the primary difficulty is the large (100 to 1 ratio) of out-of-reactor to in-reactor volume of the system. This large ratio results in very slow buildup of decomposition products. If the system leak rate is excessive, the equilibrium tar content may be less than the probable goal concentration of 35%. A method of increasing the in-reactor volume is under study. It would utilize two side-to-side test holes, reducing the out-to-in ratio to about 12 to 1. This change would result in a satisfactorily high rate of decomposition product buildup.

Out-of-Reactor Tests. Modifications and safety improvements are being made in ORA-1.

Some leakage occurred in a KER valve exposed in ORA-3 to MIPB at 325 C. This was probably due to decomposition of the teflon packing. Substitute packings are under investigation.

Electric connections to the 185-D heat transfer test facility were completed and shakedown testing using HB-40 and MIPB commenced. The mechanical seal on the existing pump showed undesirable leaking tendencies. Efforts were started to procure a small canned-rotor pump. Other apparatus operated satisfactorily at temperatures

of 265 F with a flow rate of 2.5 gpm. This corresponds to a bulk velocity of 14 fps in the test section.

Minor revisions were made to the design of the 314 Building organic facility test section to permit the testing of larger mechanical seals. Almost all orders for major pieces of equipment have been placed.

Reactor Effluent Studies

The results of a recent reactor production test were used to calculate an apparent hold-up time for P^{32}. NaH_2PO_4 equivalent to 5 ppm of H_3PO_4 was added to the cooling water of one reactor tube. The apparent hold-up time for P^{32} in the production test was calculated to be 8 seconds as compared to a 2.5 day hold-up time obtained by addition of 2×10^{-5} ppm of phosphorus to the cooling water. These data show the marked reduction in apparent hold-up time that occurs with a large increase in concentration of the parent isotope.

The first purge-while-operating with disposal to the river was performed at 100-C. Ineffective film removal resulted.

Filter media samples were obtained from the Pasco filter beds prior to and after back-flushing for measurement of total radioactive material removed from the river. The results and data from similar samples taken when reactor purge material has passed through the filter will permit an estimation of the effects of purges on the radioactive material held on the filter beds.

Correlations between concentrations of fission products in reactor effluent were reviewed. Correlation coefficients were calculated to determine whether a linear relation existed between logarithms of the fission isotopes taken in pairs. Significant correlations were found. Further analysis revealed that the best straight line through the plot of log isotope A versus log isotope B had a slope insignificantly different from 1. This substantiated the linear relationship between these pairs of isotopes. Absolute average values of the ratios of isotopes are being reviewed to assist in establishing average irradiation time. The conversion of analytical data and reactor operating variables to a form suitable for correlation determinations is almost complete.

Continuous extraction of As^{76} from a flowing stream of reactor effluent was tested using As^{76} tracer and 107 Basin effluent water. Using the monitor reagent and sample flow rates, it was determined that only 0.015% of the As^{76} was not retained on the CuS bed. In another test, 107 Basin water from an in-pile test which produced relatively high P^{32} concentrations was passed through the ion column and the CuS bed. The P^{32} passed through the ion column and CuS bed as anticipated, although some hold-up was indicated. Evaporation losses from the pre-treatment cell for As^{76} removal were shown to be significant and reflux or an empirically determined correction may be required.

Counting measurements and calculations were completed on the yttrium, lanthanum and rare-earth radioisotopes from two samples of reactor effluent water. The results corrected to 4 hours after leaving the reactors follow.

"RARE EARTH" GROUP RADIOSOTOPES IN REACTOR EFFLUENT WATER

Radioisotope	Sample Activity, $\mu c/ml \times 10^6$	
	107 D - 11 Oct. 1956	107 C - 24 Oct. 1956
Y^{90}	8.8	12.8
Y^{91}	0.42	0.31
Y^{92}	31.7	10.4
Y^{93}	9.66	13.7
La^{140}	3.07	6.49
La^{141}	2.79	15.5
Ce^{141}	1.10	0.57
Ce^{144}	2.52	0.94
Pr^{142}	Not Determined	0.87
Pr^{143}	Not Determined	0.07
Pr^{145}	4.3	1.03
Nd^{147}	0.80	0.42
Nd^{149}	7.4	1.96
Sm^{153}	10.3	7.77
Sm^{156}	2.12	1.68
Eu^{152}	8.7	7.8
Eu^{156}	0.23	0.30
Eu^{157}	2.7	3.96

Y^{91}, Y^{92}, Y^{93}, La^{141}, Pr^{145}, Sm^{156}, Eu^{156}, and Eu^{157} are produced by fission. Pr^{142} and Eu^{152} are produced by neutron activation, and considerable amounts of Y^{90}, La^{140}, Sm^{153} are produced by neutron activation as well as by fission.

CHEMISTRY AND SEPARATIONS PROCESSES

Waste Storage Studies

Anticipated volumes and costs of separation plants waste storage were reviewed to establish the areas which would benefit most from research on reduction of waste volumes or improved waste storage methods. The Purex extraction waste volumes ultimately expected will be low enough that further reduction is not justified. Coating waste and Redox extraction waste volumes present an attractive area for efforts at cost and volume reduction. Under Redox Phase III conditions and with Purex at twice the capacity factor, the annual waste storage costs are as follows:

Redox extraction wastes at 28¢/gallon	$454,000
Purex extraction wastes at 38¢/gallon	91,000
Coating wastes at 20¢/gallon (old tanks)	420,000
Total annual waste storage cost	$965,000

These data show that Purex extraction wastes comprise only about 10% of the total stored waste costs, Redox extraction wastes amount to about 47% of the total, and coating wastes (from both Redox and Purex operations) account for about 43% of the total. Evaporation of coating wastes and aged Redox supernates was suggested as a means of reducing storage costs. Scavenging and the possible volume reduction obtainable via the calcium nitrate alternate salting agent, as well as the use of cheaper above-ground storage tanks for coating wastes, were also suggested as means of stored volume and costs reduction.

Work on the review of potential fission product radiation sources emphasized the need for evaluation of elements recoverable from wastes. The evaluation should be based on age, irradiation level and the activity which can be obtained from the mixed isotopes recovered during isolation of a specific element. Although certain isotopes may have high activity, the dilution by inactive isotopes of the same element required an evaluation, as mentioned above, in order to logically select or promote interest in specific fission product elements.

Chemical Preparation of Uranium (IV) Fluorides

A preliminary study was made of the direct precipitation of UF_4 from 50% TBP-CCl_4 solution by gaseous HF. Hexavalent uranium in HCl solution was reduced to U(IV) [redacted]

Purex

Hot Semiworks Waste Self-Concentrator. The HSW waste self-concentrator has been operated since July 31 with the 1/2-inch vapor header valve open and the 1-inch valve closed. Batch reflux conditions have been maintained. Except as noted below, no "bumps" have occurred since October 9.

The dip tube air purge rate was increased from 0.1 CFH (ft^3/hr) to 0.5 CFH on November 5. The tank pressure increased to 3.5 inches water over the half-hour period immediately following the purge air flow change. The tank remained pressurized for 95 minutes. Thirteen liters of condensate were collected during the "bump" period. Although the air rate has been maintained at 0.5 CFH, no other "bumps" have occurred. The total quantity of condensate collected in November was thirty liters.

Two gas samples were withdrawn directly from the vapor space of the waste self-concentrator tank. These samples were checked for H_2, O_2, N_2, CO_2 as well as oxide of nitrogen by spectrometric analysis. The dry gas H_2 analysis ranged from 0.2 to 0.3 (±0.05) volume per cent. On the basis of the dry gas H_2 analysis only, the tank gas contents are a factor of 10 below the explosive limit. This factor is increased considerably by steam dilution as long as the contents continue boiling.

Mini Runs. A control run was made employing a 60 C flowsheet and the extended scrub section, but omitting the addition of Versene to the scrub stream.

Comparison of the zirconium-niobium organic profiles indicates an improvement by a factor of about two in extraction section zirconium-niobium decontamination through addition of Versene. An additional improvement is noted in the scrub section. Over-all zirconium-niobium decontamination is increased by a factor of about five by addition of Versene to a concentration of 0.0067 M in the aqueous scrub solution.

Over-all ruthenium decontamination decreased by a factor of approximately 1.5 in the control run as compared with the run employing Versene-containing scrubs. However, the peak concentration of ruthenium observed in the organic phase was higher by a factor of five in the control run as compared with the run employing a Versene-containing scrub.

Profile Pulse Column Studies. Runs made with the profile pulse column indicate that scrub section zirconium decontamination is improved slightly and extraction section decontamination is decreased slightly by increasing the volume velocity. These data are tabulated below.

EFFECT OF VOLUME VELOCITY AND URANIUM SATURATION ON ZIRCONIUM DECONTAMINATION

U Saturation g/l	Volume Velocity (gal/hr, ft^2)		Zr Decontamination Factors		
	Extn.	Scrub	Extn.	Scrub	Over-all
72	425	160	1.2	18	21
88	450	160	1.8	8.5	16
74	850	320	0.9	32	29
91	900	320	1.5	11	16

The flowsheet used in these runs was chosen deliberately to yield poor extraction section decontamination and was as follows:

Feed: 1.26 M U, 3 M HNO_3 Flow: 23 to 30
Scrub: 1.6 M HNO_3 Flow: 20
Extractant: 30% TBP in Shell E-2342 Flow: 100

A run was also made in which the organic to aqueous ratio in the scrub section was altered by feeding fresh organic to a point just above the feed point, thus diluting the organic in the scrub section. This resulted in a slight increase in the scrub section decontamination which corresponds to a substantial decrease in zirconium HTU in the scrub section. The data are shown in the following table.

EFFECT ON ZIRCONIUM DECONTAMINATION OF DILUTING SCRUB SECTION ORGANIC WITH FRESH ORGANIC

Product U Saturation, g/l	Zr Decontamination Factors			Scrub Section HTU for Zr, ft.
	Extn.	Scrub	Over-all	
92	3.9	4.3	17	3.4
70	5.0	10.5	52	2.0

The increased extraction section decontamination observed in the low saturation run may have been due to equipment failures that forced postponement of the run. Storage of the feed for a week allowed some niobium to grow in.

Extraction Behavior of Zirconium. Versene is capable of forming a stable complex with zirconium even at the high acidities encountered in Purex systems. An evaluation of the complexing capability of other amino acids has been made to determine if a reagent having a higher solubility in acid solutions than Versene might be found. Several commercially available amino acids have been tested to determine their ability to suppress the extraction of zirconium tracer out of 2 M HNO_3 and into 30 per cent TBP. The distribution coefficient of zirconium was reduced by a factor of 80 by 0.05 M nitrilo triacetic acid. The extraction coefficient of zirconium was not altered when the aqueous phase was made 0.1 M in picolinic acid, alanine, asparagine, aspartic acid, nicotinic acid, glutamic acid, or acetylglycine.

The two hydrolysis products of ethyl dibutyl phosphoryl acetate have been prepared. Both of these acids have a very high water solubility and low solubility in nonpolar solvents and will be tested to determine their ability to form chelates with zirconium and plutonium (IV).

Recent results suggest that products formed by reaction of uranium carbides with aqueous HNO_3 solutions are responsible for suppressing the extraction of tracer zirconium from solutions prepared from certain types of uranium metal.

A comparison was made of the extraction behavior of zirconium tracer from solutions prepared from (1) Mallinckrodt production metal (2) Mallinckrodt dingot metal and (3) Fernald production metal.

The results obtained in these extraction studies confirmed those obtained in earlier experiments. Distribution coefficients for tracer zirconium were the same for solutions prepared by dissolving dingot metal in nitric acid as for solutions prepared from crystalline uranyl nitrate hexahydrate. Distribution coefficients were significantly lower than those obtained from solutions prepared from Fernald & Mallinckrodt production metal.

Aqueous solutions initially 0.5 M in uranium and containing varying amounts of HNO_3 were contacted with a sufficient volume of 30 per cent TBP in Shell E-2342 to yield a uranium concentration of 0.1 M in the organic phase. Zirconium distribution coefficients ranged from 0.014 at 1 M HNO_3 to 0.35 at 4 M HNO_3 for solutions prepared from Mallinckrodt dingot metal. Over the same acidity range, zirconium distribution coefficients ranged from 0.0084 to 0.11 for solutions prepared from Mallinckrodt production metal and from 0.005 to 0.066 for solutions prepared from Fernald production metal.

Spectrographic analysis of these metal samples did not reveal the origin of the "complexing agent" responsible for suppressing the extraction of zirconium out of the solutions of the two production metals. Significant differences in carbon content were noted for these three metals. This suggested the possibility that products formed from uranium carbides could be responsible for the observed effects.

The possibility of complexing agents for zirconium being formed by reaction of nitric acid with uranium carbides was investigated. A feed solution was prepared by simultaneous addition of 0.78 grams of a mixture of uranium monocarbide and uranium mononitride along with a 90-gram sample of Mallinckrodt dingot metal to boiling 11 M HNO_3. Samples of the solution were withdrawn after intervals of 10, 35, and 67 minutes, diluted four-fold with water, adjusted to 1.7 M HNO_3, spiked with zirconium tracer, and the distribution coefficient of the zirconium into 30 per cent TBP in Shell E-2342 measured. The distribution coefficients measured for zirconium in this experiment were lower by factors of two to three than those which had been obtained in experiments employing solutions prepared from dingot metal or from crystalline uranyl nitrate hexahydrate.

Additional extraction-scrub studies were carried out with plant dissolver solution to typify the ruthenium and zirconium species which limit decontamination in the Purex process. These experiments were performed with specially purified tributyl phosphate and employed distilled carbon tetrachloride as diluent. The behavior of zirconium did not indicate that it exists in these solutions as a single species or as a series of labile species, nor did it lend support to the theory that the irreversible extraction is due to an impurity in the organic phase. It appeared that the zirconium existed as a series of inert species.

Washing crude tributyl phosphate with a sodium carbonate-potassium permanganate solution for 24 hours at room temperature results in a very white product that has an ultra-violet spectra very similar to distilled TBP. This may be a better solvent clean-up than the present carbonate washes.

Adsorption of Zirconium by Autunite. Autunite exhibits properties of cation exchange and reversible hydration which suggest that it might have application as an inorganic ion exchanger for removal of fission products from uranium solutions.

Shaking 100 mg of Autinite with 10 ml of 0.1 M nitric acid containing tracer zirconium resulted in decontamination factors of 21 and 60 after 16 and 40 hours, respectively. The solubility of this precipitate appears too high for process use.

Low-Acid HA Column Flowsheet. The investigation of Purex HA Column performance with a low acid flowsheet was completed. Eight additional "cold" runs were made in a glass pulse column having a 3-inch-diameter by 9-ft-high extraction section and a 4-inch-diameter by 9-ft-high scrub section. The scrub section contained a "standard cartridge" and a "graded cartridge" similar to the one in the Purex plant extraction section. No louver plates were included. The results are summarized in the following table:

SUMMARY OF PUREX LOW-ACID HA COLUMN STUDIES

Flowsheet: Purex Plant Flowsheet No. IIA. HAS = 2 M HNO_3; HAF = 1.0 M HNO_3, 1.75 M U.

Interface Position: Top
Pulse Amplitude: 1.1 inch in the extraction section.

Flooding Studies

Capacity Factor[a]	Volume Velocity, Gal/(Hr)(Sq Ft)	Flooding Frequency Cycles/Min	
		Instability	Complete
1.0	409	90 ± 5	105 ± 5
1.5	600	75 ± 5	95 ± 5
1.5[b]	592	---	95 ± 5
2.5	991	70 ± 5	80 ± 5
2.7 ± 0.25	1100 ± 100	---	60

Efficiency Studies

Capacity Factor	Volume Velocity Gal/(Hr)(Sq Ft)	Frequency Cycles/Min	Per Cent of Flooding Frequency	U Loss Per Cent
1.5[b]	590	70	75	.0026
1.5[c]	590	85	90	.0017
1.5[d]	590	60	75	.0006

Notes:

(a) Nominal values.
(b) The HAF composition for this run was 15 per cent above flowsheet for both U and HNO_3.
(c) The HAF HNO_3 concentration for this run was 12 per cent above flowsheet.
(d) The HAF HNO_3 concentration for this run was 0.5 M.

The flooding frequencies for HW No. 3 Flowsheet conditions were lower than those normally obtained due to a restriction in the middle of the extraction cartridge caused by two sieve plates which had been installed as one. The low-acid and HW No. 3 Flowsheet results are believed to be comparable, however, and it can be concluded that Purex Plant Flowsheet No. IIA conditions give only a slight decrease in flooding frequency and have no effect on extraction efficiency.

Cesium Recovery from Purex Waste. Development of methods for the conversion of cesium zinc ferrocyanide to cesium chloride continued. A fresh sample of plant IWW was obtained for further full level testing of the ferrocyanide precipitation flowsheet.

A process converting cesium zinc ferrocyanide to cesium chloride was tested with feed containing cesium-137 tracer. Cesium recoveries were greater than 99 per cent and the weight of cesium chloride product agrees with theory. Other experiments have been aimed at substituting phosphoric acid for sulfuric. Dissolution of the ferrocyanide required about fifty per cent more phosporic acid than sulfuric acid Furthermore colloidal sulfide precipitates were obtained in the presence of the phosphate.

More observations have also been made on the calcination-chlorination process. Direct observation has shown that no liquid is produced during the chlorination cycle but that it is formed while air is being passed through the furnace (at 350 C) to convert the ferric chloride to oxide.

Simple calcination followed by water leaching has also been investigated as an alternative to chlorination or to sulfuric acid dissolution. Smooth conversion was obtained at 340 C in eight hours. Although no cesium activity was volatilized, subsequent leaching removed only about one-half of the cesium.

A cask of 700 MWD/T Purex IWW solution was found to contain a precipitate consisting largely of aluminum, phosphorous, and silicon, with a smaller amount of zirconium The precipitate contained a large amount of zirconium-niobium activity. Neutralization of the supernate resulted in very bulky precipitates. Caustic addition to a high pH, followed by adjustment to pH 5, gave precipitates which were about equal in volume and totalled 70 per cent of the original IWW volume. These observations suggest that gross amounts of aluminum were present in this sample of waste.

Strontium and Cerium Recovery from Purex Waste. The behavior of uranium and zirconium in the low pH (iron removal) precipitation step was studied. Apparent uranium losses to the precipitate ranged from 9 per cent to 17 per cent with gaseous ammonia precipitation. The final pH's were 1.5 to 3.0, respectively. However, washing the precipitates reduced the losses at pH 2.5 to one per cent, with ammonia precipitation, and to three per cent when urea hydrolysis was employed for iron precipitation. In other experiments with purified (niobium free) zirconium tracer, the bulk of the zirconium activity was carried down with the ferric hydroxide precipitat and was not appreciably removed by washing. Zirconium decontamination factors ranged from 9 to 51 and did not appear to be a function of pH in the range 2.0 to 4.0.

Several experiments were performed to determine the possibility of recovering uranium and plutonium from Purex IWW by a "one shot" batch extraction technique. Extraction of 10 volumes of IWW with 3 volumes of 30 per cent TBP was followed by batch scrubbing with 7.5 M HNO_3. The plutonium and uranium were then stripped with caustic. Ninety-two per cent of the plutonium and nearly all the uranium were recovered. Decontamination factors of 12,000 and 2,000 were observed for gross beta and gamma, respectively.

Neptunium in the Purex Process. The various Purex process streams are being analyzed to determine the path of neptunium and the feasibility of recovery. A neptunium count equivalent to 80 per cent of that present in the feed has been found in a sample of plant IBP. None has been found in the uranium product. Additional plant samples are being analyzed to verify these findings.

Redox

Calcium Nitrate as Salting Agent for Redox. Data reported previously indicate that, from the standpoint of decontamination and product recovery, calcium nitrate can be used to replace most of the aluminum nitrate as salting agent in the Redox process. Further studies have confirmed these conclusions. Some buffering, most readily obtained by low concentrations (ca. 0.2 M) of aluminum in scrub solutions, is necessary to maintain pH control.

The presence of ferric iron and sulfate in feed and recycled waste streams (resulting from ferrous ammonium sulfate and sulfamic acid) poses some problems with calcium nitrate as salting agent. Calcium sulfate will precipitate in the 2AF stream (about six volume per cent of solid) and to a lesser extent in the precycle and partition columns. Also, under extraction column conditions, precipitation of large amounts of a compound, thought to be basic ferric dichromate, has been observed after most of the uranium is extracted. The precipitation appears to be more severe with mixed calcium nitrate-aluminum nitrate salting agent than with aluminum nitrate alone.

Cesium Recovery from Redox Waste. Testing of the zinc ferricyanide scavenging process was extended to full level plant HAW solution.

The first full level experiments were carried out on a year old sample of first cycle waste. Cesium recoveries of about 50 per cent were obtained. Addition of sulfamic acid did not improve zinc ferricyanide scavenging. A series of tracer experiments was run to determine the effects of chromic ion and dissolved hexone and to better define the effects of nitrate and nitrite suppressors. Neither chromium (III) nor dissolved hexone had any effect and recoveries were consistently better than 99 per cent. Nitrite concentrations greater than 0.1 M were required to give a pronounced effect. The addition of either sulfamic acid or hydroxylamine restored the recoveries to 97 per cent or more. The inferior scavenging in the full level experiments was probably due to high aluminum concentrations because the ANN concentration in the aged IAW was 1.85 M rather than the flowsheet 1.3 M

Metal Recovery Process

UO_3 Studies. Reduction temperatures of 535 C and 870 C were used to determine the correlation of (1) particle size (2) surface area and (3) oxygen to uranium ratio of UO_2 derived from Hanford, Mallinckrodt, and Savannah River "pot" powders and Hanford continuous powders. When the reduction was carried out at 870 C, the particle size, surface area, and stoichiometry of the UO_2 samples were quite similar for material derived from both the "pot" and "continuous" powders. The oxygen to uranium ratio in these instances was about 2.05. At 535 C reductions of the "pot-type" UO_3 resulted in UO_2 with a smaller particle size and a smaller surface area. The oxygen to uranium ratio of the pot-type product was 2.17 and that of the continuous powder was 2.22.

UO_2 formed at the higher temperature was more inert and contained less oxygen. The quantity of oxygen in excess of the stoichiometric amount for UO_2 is probably a measure of the reactivity of the UO_2 sample. Larger particle sizes and the smaller surface area of the product are formed at 870 C because the UO_3 thermally decomposes to U_3O_8 prior to the reduction.

An examination of unmilled continuous-type UO_3 pellets disclosed smaller surface areas and porosities than in milled powder. Two exceptions were (1) a "continuous" powder containing about 3 per cent residual water and nitrate was heated for 2 hours at 450 C to yield a product of porosity and surface area like milled material and (2) a "continuous" powder produced at the high calcination temperature of 450 C had a very high porosity and the largest surface area of 8 kinds of pellets tested. In contrast, some pellets produced at 350 C had extremely low porosity and surface area.

Continuous Calcination. The 16-inch-diameter by 8-ft-long continuous calciner was operated to demonstrate the feasibility of calcining UNH and hydrating the UO_3 in a single unit. Such a procedure would reduce markedly the cost and installation time if a hydration step is found necessary to produce reactive UO_3.

The operation was achieved by utilizing the UNH feed points in the normal manner with feed point temperatures between 250 and 260 C and introducing sufficient hydrating agent to appreciably reduce the UO_3 temperature at the discharge. It was also desired to demonstrate whether water, 3 M HNO_3, or 6 M HNO_3 is the most desirable hydrating agent and whether further reactivity gain is obtained on dehydration.

Approximately 16,000 pounds of hydrated UO_3 were produced. Highlights of the study include (1) discharge temperatures of 150 C could be maintained without causing operating difficulties (2) neither water nor 6 M HNO_3 was effective as a hydrating agent in increasing the reactivity of the UO_3 (3) three M HNO_3 as the hydrating agent increased the reactivity of the UO_3 to 0.85 (4) subsequent dehydration of the hydrate formed by 3 M HNO_3 addition did not further increase the reactivity of the UO_3 (5) dusting averaged 2 per cent of the UO_3 discharge rate and (6) there appeared to be no caking problem during the water and 3 M HNO_3 addition portions of the study, but definite evidence of cake formation was found during the 6 M HNO_3 addition.

The study indicated that a calcination - hydration operation can be carried out in a single unit. However, because of the large increase of off-gas volume, dusting, and the fact that best UO_3 reactivity obtained is lower than that of pot-produced UO_3, the process does not appear too attractive for plant application without further testing.

High Pressure Feed System - Continuous Calciner. To determine the feasibility of using heated uranyl nitrate trihydrate (UNT) as a feed to the continuous calciner, experiments were performed to establish the equilibrium vapor pressure in the range of 150 to 170 C. For the system, $UO_2(NO_3)_2 \cdot 2.9\ H_2O$, the vapor pressure in the range of 155 to 180 C is represented by $\log P_{mm} = \frac{1940}{T} + 7.51$. Below 155 C the vapor pressure is higher than predicted because of the freezing out of the dihydrate leaving a liquid phase richer in water than the original $UO_2(NO_3)_2 \cdot 2.9\ H_2O$. The heat of dehydration derived from the slope of the vapor pressure - temperature curve was 9.2 kcal/mole H_2O, nearly equal to the heat of vaporization of water.

Fluidized Bed. The 2-stage fluidized bed unit was inaugurated in a reduction run using Hanford continuous UO_3 as a feed material. Despite the usual minor startup difficulties, the operation was quite successful.

Ion Exchange Processes

Anion Exchange. Studies in the adaptation of anion exchange to plutonium processing were continued with (1) evaluation of various anion exchange resins and (2) the decontamination of Purex IBP solution by adsorption on and desorption from an anion exchange column. The characteristics of various anion exchange resins are being studied by comparison of product breakthrough obtained under loading conditions of constant flow rate and feed composition. A 40 cm column of Dowex-1, X-8 (50 to 100 mesh) was loaded to breakthrough with a solution 1 g/l in plutonium and 7.2 M in HNO_3 at a flow rate of 10 ml/min/cm^2. The capacity at 1 per cent breakthrough was 45 grams plutonium per liter of resin. Elution with 0.25 M HNO_3 at a flow rate of 0.5 ml/min/cm^2 removed 75 per cent of the plutonium at a concentration of 7 g/l.

Since the 50 to 100 mesh resin requires pressurization to achieve practical flow rates, investigation of coarser resins has been undertaken. To obtain more favorable kinetics, resins of lower cross linkage are also being examined.

A feed solution of Purex IBP adjusted to 7.2 M HNO_3 containing 0.01 M sodium nitrate was passed through a 35 cm long column of Dowex-1, X-4 (50 to 100 mesh) at a rate of 17 ml/min/cm^2. The plutonium content of the feed solution was about 0.5 g/l. The column was loaded to 0.15 per cent breakthrough and was then washed with 20 column volumes of 7.2 M HNO_3 at a flow rate of 17 ml/min/cm^2. Elution was effected with 0.25 M HNO_3 at a flow rate of 0.5 ml/min/cm^2. The decontamination factor for gross gamma, zirconium-niobium, and ruthenium were 390, 510, and 180 respectively. The decontamination factor for iron was 5000 leaving a residual iron content of 400 ppm Pu. The average loading of the resin was 33 g of plutonium per liter of resin and the plutonium concentration of the product was 10 g/l. Plutonium lost in the effluent feed and in the wash was 0.025 and 0.050 per cent, respectively.

A similar experiment was performed differing only in the rate of flow of the wash solution. Reduction of this flow rate from 17 ml/min/cm^2 to 2.4 ml/min/cm^2 resulted in gross gamma, zirconium-niobium, and ruthenium decontamination factors of 3100, 3700, and 1650, respectively. This represents an approximate eight-fold improvement in decontamination. The 8/AT ratio of the final product was 6×10^{-12}.

Continuous Ion Exchange. The 321 Building moving bed ion exchange unit operated for 120 hours during the month without mechanical difficulty. Data obtained on a nitric acid-ferric nitrate system with 50 - 100 mesh Dowex 50W(8X) indicated at least 10 stages in both XA and XC columns (31 and 61 inches high, respectively) at a throughput of 140 gal/(hr)(sq ft). Microscopic examination of resin which had been cycled ten times through the unit showed no evidence of broken particles.

Absorption of Ions from Fused Salt Systems. Uranium(IV) was successfully removed from a solution of UCl_4-LiCl-KCl eutectic by passage of the molten salt through a column of "3M Inorganic Exchanger".

Twenty cubic centimeters of the exchanger were converted to the sodium form by treatment with aqueous 0.1 M NaOH. It was dried at 110 C and heated to 400 C under vacuum. The exchanger was then added to the molten eutectic to form a slurry. This was poured into a heated Pyrex column (1 cm diameter by 10 inches long) where the solid settled out forming a bed. A dilute solution of uranium chloride in the eutectic was passed through the column and a pale green band about 1 inch in length was observed to form at the top of the column. Washing with pure eutectic did not remove nor displace the band. The column was cooled, sectioned and part of the uranium recovered by treatment with hot 6 M HCl. It is not known whether the uranium is held in the manner of zeolitic absorption or whether it has actually occupied sites in the lattice.

Thermal Diffusion Studies

Preliminary experiments were conducted to test the possibility of separating molten metals by thermal diffusion. A mercury alloy containing approximately 1 per cent gallium was processed in the 0.20 mm thermal diffusion column with a temperature differential of 73 degrees. A 2.5 cc sample was withdrawn daily. The gallium content of the samples at 0, 6.5, and 47 hours were 0.69, 0.78, and 0.83 per cent, respectively. The reservoir analysis at the termination of the run was 0.69 per cent gallium. The separation was rapid but the product rate was high and the temperature differential low. The separation factor was 1.2.

An experiment was undertaken to determine the behavior of fission product zirconium-niobium in an aqueous thermal diffusion system. A sample of Purex 2DF, low in activity, was processed in the 0.20 mm thermal diffusion column using a temperature differential of about 81 degrees and a product take-off rate of about 7.5 cc per day. The results were not completely reproducible. However, a 3-fold decrease in the zirconium-niobium to uranium ratio was observed at the top of the column.

Flurex

Flurex process studies were continued to determine the effects of stannuous ion in the catholyte on the current efficiency-current density relationship. In the presence of 0.05 M stannous ion, current efficiencies of 94 per cent or greater were observed when the current density at the cathode was varied from 0 to 10 amperes per square inch. In the absence of tin, the current efficiency remained at greater than 95 per cent up to a current density of 3 amperes per square inch and then dropped steadily to about 40 per cent at 10 amperes per square inch. These experiments demonstrated the beneficial effect of tin in the system.

Analytical Development

Determination of Total Fissions for Irradiated Uranium. Methods for measuring the number of fissions in uranium irradiation were reviewed because of some weaknesses in the conventional Cs-137 method at high burnups (fission of more than 0.5 per cent of total uranium). Cs-134 is formed from fission product Cs-133 because of a 29 barn thermal cross section and a 2600 barn absorption resonance at 5.9 ev and other resonance peaks. In some cases of high burnup, the yield of Cs-134 is so great that a serious interference results during counting of the 0.66 gamma of Cs-137. This interference is caused by the 0.60 and 0.79 Mev gammas of Cs-134. A pure Cs-134 source was used to measure the contribution of this isotope to the Cs-137 count. Counting Cs-137 at 0.638 to 0.688 Mev in a large well crystal seems to be preferable. The efficiency for Cs-137 is 5.7 times greater than Cs-134 in a 3-inch crystal and 7.9 times greater in a 4-inch crystal.

Electron Microscopy. In applying the electron microscope to a study of the physical appearance of microscopic UO_3 particles, a replication method for sample preparation is used. This includes (1) dusting a small amount of the sample on a microscope slide (2) coating the sample with a palladium plus carbon film by vacuum sputtering (3) floating the film off the slide and (4) dissolving the sample particles off the film. This film replica of the particles of sample is then photographed with the electron microscope. The method can be applied to substances opaque to electrons only if a suitable solvent for the sample can be found which does not also attack the replica. A study of replication methods for uranium tetrafluoride samples containing small amounts of uranium dioxide is underway. Five likely solvents for the tetrafluoride-oxide mixture have been studied, and a 0.5 M solution of ceric sulfate in 2 M sulfuric acid seems to be the most effective.

In-Line Analysis

Sulfamate Determination. The sulfamate concentration in the IBP stream is usually determined by a titration with nitrite. An automatic and continuous determination of sulfamate can be made by nitrite titration with flowing streams. Such an automatic titrator has been assembled and is operating successfully in the laboratory. Tests for interferences showed that the ferrous and ferric iron concentrations are not influential but nitric acid concentration variations have a definite effect on results.

Chemical Instrumentation

Contact Alpha Counter The contact alpha counter has operated several weeks on aqueous plutonium solutions. Precision estimates, background levels, and detection limits established are (1) plutonium at a concentration of 0.60 grams per liter can be determined to plus or minus 0.03 grams per liter (2) alpha activity buildup can be controlled at reasonable levels by periodic rinsing and (3) the lowest background obtainable to date is equivalent to 0.01 grams per liter of plutonium and represents about the minimum concentration of plutonium that can reliably be detected with the present apparatus.

Plutonium Resin Column Controls. The laboratory prototype of the scanning device to locate the resin level in the reservoir section of the column has been demonstrated. The device locates the resin level to plus or minus 0.5 inch in the control section and gives both visible and audible alarms when the resin level is either too high or too low.

Equipment and Materials

Deepwell Turbine Liquid Seals. Accelerated wear testing with a carborundum-in-water slurry has been discontinued. Life testing of a CSGBF pile graphite liquid throttle bushing is in progress The bushing is protected with a conical stainless steel deflector mounted on the shaft just below the throttle bushing. At the end of 24 days operation, no liquid leakage past the liquid throttle could be detected. In comparison, new bushings which are not protected by the conical deflector will leak a small amount and the leakage increases as the throttle bushing wears.

Continuous Solvent Washer. A Purex-type organic recovery flowsheet was used to measure the efficiency of the recently installed continuous solvent washer. The washer unit consists of a 24-inch-diameter agitator rotating at 100 rpm within a 25-inch-diameter diffuser. The diffuser is supported three feet from the bottom of a 10-ft-high, 4800 gallon tank by a 30-inch-diameter baffled tube. Orifices with 6- and 11-inch-diameters are located above and below the diffuser, respectively. The organic enters continuously below the diffuser and overflows near the top of the tank.

Three efficiency runs were made utilizing 1750 gallons of a 2.5 to 2.6 weight per cent sodium carbonate solution as a scrub to remove "cold" uranium (0.035 g/l) from 30 volume per cent TBP in Shell E-2342 diluent. The operating temperature was 11 to 13 C. Run results are tabulated below:

Feed Rate, GPM	Organic Holdup Time, Hr.	Uranium D. F. (1)
20.7	2.45	180
39.3	1.3	133
53	1.0	103

(1) D. F. defined by: $\frac{\text{Conc. U in feed}}{\text{Conc. U in organic effluent}}$

In all three runs, the organic effluent contained less than 0.04 per cent aqueous entrainment.

Chemical Compatibility. Qualitative studies showed that Stellite-6, as used for facing on male ends of connectors in the Redox plant, is rapidly corroded by boiling nitric acid in the concentration range 45 to 60 per cent, but not visibly attacked by boiling nitric acid 15 per cent or less in concentration.

Chemical Effluents

Special Studies and Advance Planning. The basalt ridge cropping out along the north side of Cold Creek Valley and about two miles south of 200 West Area is the probable extension of the Yakima Range structure. The structure probably underlies much of the zone traversed at depths above the regional water table, thus restricting movement of the ground water from the Rattlesnake Hills into the confines of the project. Similarly the movement of water from Dry Creek Valley into the project may be appreciably decreased. The discharge of effluent to swamp sites thus becomes even more significant quantitatively in determining the probable path of movement of the ground waters beneath the Hanford Project.

Curves of decreasing activity vs time in individual wells and decreasing activity vs distance from the source were prepared. The curves were based on data obtained in past years from wells in sites of ground water contamination. The curves are comparable to similar curves of dispersion or dilution obtained theoretically and experimentally by P. S. Day. The work by Day indicates a means by which the amount of dispersion of salt water in fresh ground water can be predicted, and shows the importance of that process. Testing and application of Day's equations at Hanford may provide a measure of the dilution of radioactive materials in the ground water as they move toward the Columbia River, and thus provide a measure of the safety factors in the present methods of radioactive waste disposal.

Geochemical and Geophysical Research. Questionable data were revealed by a detailed comparison of the dilution test results with the well casing perforation schedule reported in drilling records. It became apparent that a number of tests had been inadvertently conducted within unperforated zones of some of the wells. Although some of these tests properly indicated negligible velocities, others resulted in dilution rates that were interpreted as significant ground water velocities. These latter results are apparently caused by vertical movement within the well; the quantitative significance of this effect is being investigated.

Geologic studies in the 200 West Area region defined a site in which the future disposal of wastes will present the least radiological hazard. The site lies between plant coprdinates N 30,000 and N 35,000, and between W 67,000 and W 71,000. The general suitability of the area is also proven by the past performance of the 216-S-1 and -2 cribs; no detectable Cs or Sr has yet been found in the ground waters there despite the very large volumes and total amount of radioisotopes discharged to that site.

An attempt is being made to characterize the potential "storage" capacity of proposed waste disposal sites for radio-cations by a single number. This would permit quantitative comparison of the potential life of waste cribs to be built in different

locations. The numerical characterization to be applied may be termed the "capacity factor" (CF) of the crib site and is defined as the average cation exchange capacity per cubic foot of soil times the depth of the soil profile. The expression is calculated for the column of soil between a depth of 20 feet and the water table. The expression defines the cation exchange capacity of a column of soil with the same cross-sectional area as the crib. As information becomes available it will be modified to include the expected degree of lateral spreading beneath a given crib site. The capacity factors calculated from data obtained from a number of wells adjacent to the proposed Purex waste disposal area are shown below.

CAPACITY FACTORS OF LOCATIONS ADJACENT TO THE PROPOSED PUREX WASTE DISPOSAL AREA

Plant Coordinates	Length of Soil Column, Feet	Capacity Factor Meq/Sq Ft
N 25,000 - W 55,000	285	7.0×10^5
N 36,000 - W 61,000	330	7.8×10^5
N 35,000 - W 70,000	260	11.9×10^5
N 40,000 - W 62,000	350	7.0×10^5
N 25,000 - W 70,000	195	7.4×10^5

These capacity factors may be compared to that calculated for the waste disposal area in the vicinity of the Purex plant of 5.8×10^5 meq/sq ft.

The removal of yttrium ion by Hanford soils was studied to determine the efficiency of ion exchange of this material from various solutions. It was found that the presence of other cations in solution reduces the equilibrium uptake of yttrium by soils but that the reaction is much more sensitive to small pH changes than to changes in the total salt content. The effect of various concentrations of Li^+, Na^+, K^+, NH_4^+, Cs^+, Mg^{++}, Sr^{++}, Ba^{++}, Ca^{++}, Ce^{+++} and La^{+++} upon the exchange of yttrium on soils was investigated. In general the removal of yttrium ion by soils is more complete than the removal of strontium ion so that a fresh sample of ground water containing measurable quantities of Sr^{90} could be expected to be somewhat depleted in the Y^{90} daughter.

Process Development. Experimental work to determine the effect of decontaminating solutions on the adsorption of radioisotopes by soil was continued. The results indicated that the addition of 0.2% by weight of the detergent Turco 4306 B to a 4 M solution of sodium nitrate containing Cs^{137} increased the adsorption of Cs^{137} approximately 5 times above the value obtained without Turco. A similar experiment with Uranium Recovery Plant synthetic waste resulted in a 2 to 3-fold increase in the adsorption of Cs^{137} due to the addition of the detergent. The use of this chemical additive to increase the removal of Cs^{137} from solution by soil has possible application to waste disposal problems where Cs^{137} is the limiting radioisotope provided the removal of other hazardous radioisotopes by soil is not inhibited. The addition may also have application to the separation of Cs^{137} from salt solutions by ion exchangers.

Equilibrium experiments were continued to define the mechanism of Sr^{90} adsorption by soil from solutions containing phosphate and sodium ions. The adsorption of phosphate from sodium phosphate solutions by bentonite or kaolinite was increased

by pre-treating the clays with strontium. A further increase in phosphate adsorption was obtained by adding one mole per liter of sodium chloride to the phosphate solution; this was particularly effective for the bentonite clay and was interpreted to be a result of the expansion of the layer lattice. The accumulated equilibrium data were interpreted to mean that strontium was removed from solution by an ion exchange reaction and that phosphate was preferentially adsorbed at the sites occupied by strontium ions. The attached phosphate apparently prevents strontium from undergoing an exchange reaction with the sodium ions in the solution with the result that the equilibrium established favored the adsorption of additional strontium and phosphat Experimental curves prepared from the data suggested that the reaction is probably a monolayer adsorption.

Equilibrium experiments were conducted to investigate the feasibility of ground disposal for Purex G-8 and R-8 wastes, consisting of sodium carbonate washes of used organic solvent. Preliminary data indiated that both Cs^{137} and Sr^{90} were present in low concentration ($<5 \times 10^{-4}$ μc/ml) and were removed by soil adequately for ground disposal. Pu^{239} however, at a concentration of 5×10^{-3} μc/ml had a distribution coefficient of <2, indicating inadequate adsorption by soil for ground disposal.

Soil column experiments were started to evaluate the adsorption characteristics of the radioisotopes in Purex A-8 waste (condensate from A-farm). The initial data indicated that the concentration of total beta emitters was approximately 2.3×10^{-3} μc/ml, which is about 45 times higher than the present limits for disposal to surface swamps.

Experiments were continued to evaluate the feasibility of disposing of radioactive wastes containing aluminum to the ground by immobilization of the total liquid mass. One-molar sodium aluminate solutions, used to represent aluminum coating removal waste, were completely immobilized by reaction with equal volumes of sodium silicate solution of 1.185 specific gravity. The addition of solid calcium carbonate to a solution of aluminum nitrate also produced a gel resulting in total immobilization of the liquid. These reactions are considered as possible disposal methods for Hanford process wastes. The reactions might also be useful for solidifying the contents of a leaking tanl as an emergency measure.

Process Demonstration and Analysis. The continued presence of Co^{60} in TBP scavenged supernate above the recommended cribbable limit of 4×10^{-5} μc/cc necessitated recommending tank storage or ground disposal on a specific retention basis of all TBP waste scavenged during November. To date, 6.3 million gallons of high Co^{60} scavenged waste have been disposed to the ground on a specific retention basis at the BC trenchsite. Approximately 1.5 million gallons of TBP scavenged waste will be produced before the metal recovery operation is discontinued. In addition, 16 million gallons of TBP wastes, produced prior to installation of the waste scavenging program, are being held in underground storage for future "in farm" scavenging.

A theoretical evaluation was made to determine the maximum leak from an underground storage tank which could be tolerated without radioisotopes entering the ground water table beneath the tank. The leak was considered as a point source and the analogy between seepage flow pattern and stress distributions in a loaded

homogeneous soil mass were utilized in arriving at the shape of the wetted volume. Results of this evaluation indicate that a total leak of approximately 150,000 gallons could be tolerated. Because of the limited experimental data applicable to this problem, it will be recommended that a 50,000 gallon loss be readily detectable from a given tank. Positive indication of any amount less than this should also require immediate corrective action, namely that of transferring the contents to a sound tank. Research in progress should provide data to substantiate or qualify the indicated limit.

Equipment and Instrumentation. Progress was made on the assembly of the mechanica features of the isotope analyzing monitor. Testing of cup magazines, cup actuators, and drive assemblies indicated satisfactory performance of these components; evaporation hoods were made available and mounted; the programmer for cycling the cup actuators was tested; the decay track required modification to allow for dimensional variation in the purchased sample cups.

The Ballistic Well Perforator was tested with 0.006" steel bursting discs behind the projectile to contain the gas until high pressures were reached, thus giving the projectile an initial high velocity. Tapered nose 0.509" diameter roller bearings were used as projectiles. The propellent was 70 grains of a standard fast burning double base smokeless pistol powder. The electrode insulation failed in all cases, prohibiting use of an electrode for more than one charge. The electrode cones were swaged well into the entrance holes by the great pressures. The projectiles only partially penetrated the casing wall, indicating the need for a heavier bursting disc.

The requirement for end shielding of the center anode wire for internal GM counting of tritium was demonstrated. One-inch brass tubes which previously gave poor voltage plateaus and non-reproducible counting were modified to give shielding to the center wire near the tube end A significant improvement in plateau length and decreased slope resulted. Coincident with the improved plateaus was a reduced background, from 8 to 3.7 c/m, with the anti-coincident shield. This development helps explain the superior performance of standard thin-window glass GM tubes for internal counting of tritium over brass tubes with an unshielded center wire.

The counter to identify very small movements of Sr^{90} in a soil column was further tested. Using a column with one cm diameter and 10 gm/cm^2 walls, the counter gave a counting rate of 800 c/m for a source of 2 μc Sr^{90}-Y^{90}. A vertical displacement of 1 mm resulted in the counting rate changing from a maximum to a minimum. Sketches were made for shop fabrication of counter and column supports for application in Process Development research.

Environmental and Radiation Chemistry

Paper chromatographic separation studies were made on Zr^{95} - Nb^{95}, Ba^{140} - La^{140}, Sr^{90} - Y^{90}, Ru^{106}, Ag^{110}, Cs^{137}, Sc^{46}, and Np^{239}, with 50 per cent concentrated hydrochloric acid - 50 per cent acetone as developing solution. Nb^{95} is separated cleanly from Zr^{95}, Sr^{90}, Y^{90}, Ba^{140} - La^{140}, Cs^{137} and Sc^{46} in four to six hours. Ru^{106}, Ag^{110} and Np^{239} are not separated from the Nb^{95}, so preliminary volatiliation of Ru^{106}, precipitation of Np^{239}, and absorption counting to distinguish Ag^{110} would be necessary to obtain Nb^{95}. Studies are being continued to develop suitable simple analytical procedures for mixtures containing Nb^{95}.

The search for sensitive dosimeter systems was extended to colloidal materials No effect of gamma radiation on the stability of aqueous gold sols was observed up to 14,000 rads absorbed. A slight increase in absorbance was noted after 18,000 rads had been absorbed. Initiation of sol formation in 0.01 per cent chloroauric acid was not observed even after 30,000 rads were absorbed.

Three oxidation-reduction indicators, Patent Blue A, Xylene Cyanole FF and Phenosafranine, were irradiated to measure their suitability as sensitive chemical dosimeters. None of them were more sensitive than Erioglaucine A. Patent Blue A, Xylene Cyanole FF and Erioglaucine A are all triphenylmethane derivatives and have approximately the same radiation susceptibility.

The contemporary tritium in two water samples was determined in some preliminary tests of the low level tritium determination apparatus and found to be 5 to 30 times higher than the amount expected in water from this area.

BIOLOGY

Biological Monitoring

Contamination in terrestrial and aquatic organisms as a result of radioactive materials introduced into the atmosphere and Columbia River follows:

Atmospheric Contamination. Concentrations of I^{131} in rabbits are arranged in decreasing order in the categories of monthly or bimonthly collection, in the following table:

Collection Site	$\mu c\ I^{131}$/g thyroid Average	$\mu c\ I^{131}$/g thyroid Maximum	Trend Factor
Prosser Barricade	6×10^{-3}	6×10^{-2}	+[illegible]
1 mile SE of Redox	5×10^{-3}	1×10^{-2}	+[illegible]
East of 200 East Area	5×10^{-3}	7×10^{-3}	+[illegible]
100-B Area	2×10^{-3}	3×10^{-3}	-
Meteorology Tower	2×10^{-3}	2×10^{-3}	+2
West of 200 West Area	1×10^{-4}	2×10^{-3}	-3
4 miles SW of Redox	8×10^{-3}	1×10^{-3}	-
3 miles S of White Bluffs	3×10^{-3}	4×10^{-3}	+4*
Wahluke Slope, E	1×10^{-3}	1×10^{-3}	-*
Route 4S, mile 14	1×10^{-3}	1×10^{-3}	-*
Wahluke Slope, NE	7×10^{-4}	9×10^{-4}	-*

* The last four trend factors compare values with September instead of October since corresponding collections are taken bimonthly.

The contamination levels are about the same as those observed one year ago.

In addition to I^{131} contamination of Plant origin, fallout debris of apparent off-Plant origin occurred in the amounts and distribution shown in the following table:

Sample Type	μc FP's/g sample (Average)	Trend Factor
Bone	6×10^{-5}	-
Feces	5×10^{-5}	-
Liver	6×10^{-6}	-

Swamp Contamination. Waterfowl obtained from the 221-U swamp contained the following concentrations of fission products:

Sample Type	μc FP's/g tissue Average	Maximum	Trend Factor
Coots			
Bone	3×10^{-3}	3×10^{-3}	+3
Soft tissue	5×10^{-4}	7×10^{-4}	+2
Diving ducks			
Bone	5×10^{-3}	5×10^{-3}	-
Soft tissue	2×10^{-3}	2×10^{-3}	+3
Grebes			
Bone	6×10^{-3}	1×10^{-3}	+9
Soft tissue	5×10^{-4}	1×10^{-3}	-
Puddle ducks			
Bone	1×10^{-3}	2×10^{-3}	-5
Soft tissue	3×10^{-4}	6×10^{-4}	-3

Puddle ducks obtained from the 200 East waste swamp contained concentrations approximately one-third those of similar samples from the 221-U swamp. The levels are comparable with those of one year ago.

Columbia River Contamination. The contamination levels in representative aquatic forms and in waterfowl for November are shown in the following table. With the exception of plankton, virtually all of the activity is from P^{32}.

Sample Type	Collection Site	μc radioisotopes/g tissue Average	Maximum	Trend Factor
Plankton	Hanford	3×10^{-2}	5×10^{-2}	-
Caddis larvae	"	1×10^{-2}	2×10^{-2}	-
Minnows	"	5×10^{-3}	8×10^{-3}	-2
Whitefish*	"	7×10^{-4}	2×10^{-3}	-
	Priest Rapids	2×10^{-4}	9×10^{-4}	+10
	Ringold	6×10^{-4}	1×10^{-3}	-
Shorebirds*	Hanford	4×10^{-3}	3×10^{-2}	-2
Grebes*	"	1×10^{-3}	5×10^{-3}	-
Diving ducks*	"	1×10^{-3}	8×10^{-3}	-
Gulls*	"	1×10^{-3}	8×10^{-3}	-2
Mergansers*	"	5×10^{-4}	1×10^{-3}	-4
Herons*	"	2×10^{-4}	5×10^{-4}	-
Puddle ducks*	"	1×10^{-4}	7×10^{-4}	-30

* Values are for flesh. Concentrations in bone are ten times higher for fish and three times higher for waterfowl.

The concentration of radioactive substances in most Columbia River organisms continued to decline, which is typical of this time of year. The tenfold increase for whitefish at Priest Rapids is due to the upstream migration of contaminated fish from the Project area. During the month the concentration of radioisotopes in fish collected near Hanford was three times higher than that observed one year ago.

Current values in puddle ducks and mergansers are one-half to one-sixth those of one year ago, and those of grebes and diving ducks are two to ten times greater than those of 1955. This contrast of trend factors results from varying migration patterns during the two periods being compared. The decrease in average P^{32} concentrations from last month reflects an influx of migrant birds into our region, resulting in our sampling of specimens of short exposure to river contaminants.

Samples of waterfowl obtained from hunters beyond Project boundaries contained activity densities of 9×10^{-6} $\mu c/g$ of muscle and 2×10^{-5} $\mu c/g$ of bone.

A total of 94 salmon nests was seen in the vicinity of the reactor areas this fall. Although this exceeds the 64 counted last year, it is a marked decrease from the 545 seen in the parent year of 1952. We have no evidence that the Hanford operation is responsible for this decline.

Although young whitefish exposed to different concentrations of reactor effluent under laboratory conditions suffered some mortality during the late summer, their survival has been good since the temperature of the river has dropped below 15 C.

"Separan" had no effect on trout at the strength used in the treatment of process water. The test was stopped after four months.

The difference in radioactive contamination of most small aquatic organisms held in "retained" versus "non-retained" reactor effluent is less than the difference for the two types of effluent. With the possible exception of diatoms, there appears to be little change in the ratios between summer and winter conditions.

The river water supply to the Aquatic Biology facilities at 100-KE has intermittently become contaminated with "service" water. A new supply arrangement will have to be installed before reliable data can be obtained at this laboratory.

Metabolism and Toxicity of Radioactive Materials

Reactor Effluent. Rats fed a single dose of reactor effluent, concentrated 35 times and spiked with P^{32}, deposited 6.9 per cent of the administered P^{32} in their skeletons. This is approximately half the percentage deposition observed when P^{32} was administered in ordinary aqueous solutions. Therefore, it appears quite probable that the lower than expected buildup of P^{32} in rats chronically exposed to concentrated reactor effluent was due, at least in part, to the effect of the high salt concentration

Plutonium. Rats maintained for 24 days on plutonium-containing drinking water were sacrificed and samples are in process of analysis. Additional rats were exposed to small volumes of relatively concentrated plutonium solutions deposited directly on the tongue. This method of administration proved successful. No evidence was found of contamination around the mouth of the animal and there is apparently no appreciable hold-up of plutonium in the mouth seven days after administration. These animals will be sacrificed and absorption of plutonium determined. These experiments were designed to evaluate practical plant exposure problems.

Complete data from the experiments studying absorption of plutonium as a function of age indicate that absorption after three weeks of age is comparable to that in the adult. Absorption at three weeks shows considerable variation due perhaps to the abrupt change in eating habits brought on by weaning.

Cesium. Complete analysis of the data on cesium retention following a single acute dose indicates that muscle is the critical organ, acquiring 57 per cent of the orally administered dose and retaining it with a half-life of 13 days. The MPC for drinking water calculated from these data is 1.8×10^{-3} μc/ml, in essential agreement with the ICRP value of 2×10^{-3} μc/ml. While muscle is the critical organ, kidney, liver, spleen, and heart receive at least one-half as much radiation as does muscle. Results from chronic feeding studies are available through 100 days and confirm the fact that muscle is the critical organ with kidney running a very close second.

Iodine. The reduction in thyroid avidity for I^{131} in the ewes fed 0.5 μc/day since 1953 appears to be significant. A similar reduction is evident in their 1955 offspring maintained on a similar I^{131} regimen.

The long-term study designed to define (1) the threshold damaging levels of I^{131} daily administered to pigs and (2) the possible differences in thyroid metabolism between pigs maintained on a high and those on a low plane of nutrition was initiated. The results following daily administration of I^{131} to pigs indicate that at equilibrium the I^{131} concentration is lower in the thyroid of pigs on a low plane of nutrition, confirming previous findings in acute studies. The thyroid I^{131} concentration at equilibrium in pigs at this season of the year is only about two-thirds that in sheep with undamaged thyroid glands. This difference, if maintained, coupled with the fact that pigs have larger thryoid glands, larger litters, and a shorter nursing period would indicate that the pig may require substantially larger daily doses of I^{131} than sheep to produce damage.

Radioactive Particles. The first in a series of acute exposures of mice to $Ru^{106}O_2$ aerosols were completed and the animals sacrificed.

Attempts to deflesh bones using N, N-dimethylformamide, tetrahydrofuran plus NaOH, and benzil trimethyl ammonium hydroxide in alcohol were unsuccessful. The latter defleshed rapidly but also dissolved portions of bone.

Gastrointestinal Radiation Injury. Reduction of serum albumin concentration appears to be a characteristic symptom of gastrointestinal radiation injury. Per cent albumin in serum proteins, as compared with unirradiated controls, is reduced as much as 50 per cent following irradiation of either the exteriorized or in situ

intestine at levels of 1500 r (a barely sub-lethal dose). Six hundred roentgens total body irradiation, which is of comparable lethality to the 1500 roentgens intestinal irradiation, produces a much smaller effect. Intragastrically administered Y^{91} produces an effect similar to X-irradiation of the intestine but somewhat delayed. The maximum serum albumin depression is observed about six days after irradiation and a return to normal levels occurs at about 12 to 14 days following irradiation.

Genetic Effects of Metabolized Radioisotopes. To compare the biological effects of S^{35} metabolized within the cell with the effects of irradiation applied externally, attempts were made to adsorb S^{35} on the yeast cell wall. Cells were exposed to relatively high concentrations of S^{35} for short periods of time, but the amount of adsorption was not sufficient to give radiation doses comparable with those estimated in experiments with metabolized S^{35}. Colloidal forms of S^{35} are being produced in the hope that they will adsorb more readily and also avoid the possibility of direct uptake into the cell.

Uptake of Radioactive Substances by Growing Plants. Data obtained on effluent plots watered with reactor effluent in the fourth year of testing are generally consistent with those from previous years. The concentration of radioactive substances in the grain and vegetation is still less than twice that in vegetation control plots. A reduction in nitrogen content in plots watered with 100 per cent effluent was noted, but this may be the result of the application on one occasion of thermally hot water. The data show that even after four years of irrigation with 100 per cent reactor effluent, no marked effects of the effluent water could be observed.

Effluent water, concentrated as much as 25-fold while in contact with the soil to simulate natural evaporative conditions, did not reduce the yield of barley when grown under controlled conditions. Radioactive materials in the barley were about five times more than those in barley grown in soil to which no effluent had been added. This extends the data obtained on outdoor plots and again gives assurance that even with conditions which would concentrate the effluent, no toxicity would result.

SUPPORTING ACTIVITIES

OPERATIONS RESEARCH AND SYNTHESIS

Reactor Data

An estimate of the amount of machine work that would be required at the initiation of the IPD data center has been made in terms of equivalent IBM 702 hours. The areas of professional activity have been determined and an estimate of the size of the professional staff has been made. The estimated machine load will be converted into equivalent hours for a middle class machine, i. e., IBM 650. Alternative approaches will also be outlined.

Economic Liaison

An extensive report on "Economics of Uranium Enrichment" has been issued for comment.

A preliminary investigation is being made to establish a balance between increased plutonium production and decreased unit cost. This balance can be expressed by the function: $(ICL)\,(Q_A - Q_B) + (E_B - E_A)$

where:

- ICL = incremental cost limit per gram of plutonium production
- Q_B = base production in grams
- Q_A = actual production
- E_B = base expenditures including uranium depletion costs
- E_A = actual expenditures

Factors Affecting Manpower Utilization

Investigation and discussion were continued on the work-mix concept for radiation and non-radiation work. An initial report on the model and its possible uses is being prepared.

Personnel Data Processing

Investigations have been conducted to determine what decisions and actions are necessary for effective utilization of manpower and which require information or data processing. A list of decisions and actions has been compiled. It pertains to the utilization of individuals and groups of individuals. Investigations are proceeding to determine what information is both desired and needed for each of the decisions and actions.

Reactor Model

Further mathematical developments have made possible the simplification of the form of the "variable discharge curves" for reactor operations. A final report is being prepared.

Project Bluenose

A special study was made of several diffusion problems associated with Project Bluenose.

Statistics Studies for CPD

A study is being conducted to determine a method for calculating the exact amount of additions which will reduce process variance in Task IV at both Savannah River and HAPO 234-5 Buildings. An initial examination of the data indicates that an empirical relationship can probably be found which will improve the process variance. However, the measurement error at both SRL and HAPO will impose a limit on the amount of reduction in variance that can be accomplished.

CPD collected laboratory data on the ceric sulfate method for determining the plutonium content of the Z-Plant final product. This data was analyzed to obtain an estimate of the process variance. Other components of variance estimates were also obtained. Various changes in analytical procedure are being studied to determine the effect on the required statistical tolerance limits on plutonium content.

Investigations are being made to determine (1) the sample size required to detect biases of different magnitudes between the standard MC-P and MC-2 sample positions and (2) the proposed sample position in the casting neck for the permanent mold casting procedure.

Statistics Studies for IPD

Assistance is being given to the Facilities Engineering Operation for investigating the distribution of the type of work performed by 100 K Area maintenance crews. Sampling techniques and methods for unbiased estimation of various distributional parameters are being studied.

Statistics Studies for FPD

Further calculations were made on the rupture rates of cored slugs versus solid slugs to determine the probabilities of superiorities in rupture rates by factors of k or more.

Studies have been undertaken to (1) compare performance of the different canning crews on various types of canning rejects (2) determine the effect that AlSi concentrations from the different vendors have on various reject rates (3) establish control limits for all types of rejects (4) determine the precision of the testing technique used to evaluate AlSi and related materials (5) determine the feasibility of sampling incoming lots of caps and cans by drawing random samples from randomly selected containers rather than sampling every container (6) review and set up the best procedure for checking and controlling the efficiency of visual inspection and misclassification of "good" or "bad" canned slugs and (7) assist on calculation and application of one-sided tolerance limits.

Statistics Studies for HLO

Corrosion data for three different aluminum alloys were analyzed to determine the significance of (1) differences between alloys (2) times of exposures (3) two water treatments involved and (4) interaction effects. Water treatments used were (1) circulated hot and cool deionized water and (2) alternating hot and cool deionized and process water.

Data from PT-313-105-37M were analyzed and the results were reported. The purpose of this test was to determine the effects of water autoclaving, in different media, on slug corrosion.

Assistance was given in designing a corrosion experiment involving three factors (pH, $PO_4^{\equiv}$, and $CrO_4^{\equiv}$) each at four levels, in blocks of eight.

A corrosion rate study was initiated using a fractionally replicated design involving 5 factors each at 4 levels. Several difficulties have been encountered in this analysis. They include (1) missing values (2) lack of homogeneous variance within experimental units and (3) an excessive number of zero values. For this reason the data are being analyzed to determine (1) which effects have a high probability of significance and (2) the best possible descriptive measures of the response surface.

Penetration data from several more couples in the experiment to determine the penetrations of AlSi into uranium and vice-versa were analyzed. All the couples in this experiment have now been analyzed.

Discussions were held with personnel interested in the problem of the correlations between fission products as measured in the retention basins. The use of the logarithmic transformation on the raw data was discussed. If the slope parameter in the transformed data were one, the relationship would also be linear on the original scale. A check was made of the correlations found to determine if the individual slope parameters differed significantly from one.

Aloft experiments have been conducted at the meteorological tower in 200 Area to investigate the dispersal characteristics of smoke plumes emitted from a stack. The problem of interpretation of the experimental data is greatly simplified if the various aloft runs are classified into homogeneous subgroups. Statistical procedures for accomplishing this classification were reported.

Twelve sets of equations concerned with replacement reactions were solved. These cases embraced the replacement of a divalent cation with another onto a trivalent anion, and the simple combination of a divalent cation onto a trivalent anion. Each set of equations was solved for one of the variables in terms of the others. Inferences were made as to the shape of the curve as a function of one of the specific variables with the others fixed.

Uptake of Sr^{89} and Ca^{45} into bone and soft tissue of rats was investigated as a function of the concentration and chemical composition of the inert carrier and of the mode of administration of tracer. The data are currently being analyzed to determine which, if any, of these factors increases uptake rates.

A statistical evaluation was made of the proposed safety award program for HLO.

Limits of expected variation of the monthly minor injury frequency rates for HLO were derived.

RADIATION PROTECTION

The average daily emission of I-131 from chemical processing plants was 1.4 curies with a maximum daily emission of 4.4 curies from the Purex Plant. The maximum total emission during any seven-day period was 20.5 curies. Essentially all of this emission was released from the Purex Plant.

No new cases of plutonium deposition were confirmed during the month. The total number of employees who have a measurable deposition of plutonium remained at 196. No new cases of fission product deposition were detected with the routine bioassay program.

A complete review was made of one plutonium deposition case. The employee had received his exposure to plutonium at another AEC site.

The conversion of film badge recording analysis to data processing techniques progressed on schedule. The total beta and gamma exposure from 1944 through the third quarter of 1956 has been summarized. Input cards have been key punched for all operational personnel who currently have an active area clearance.

A preliminary review was completed for the AEC on radiation protection aspects for river navigation through the Hanford project.

A major review of the regional monitoring program was initiated to permit (1) revising sampling programs and (2) evaluation of doses in relation to maximum permissible limits.

A review was initiated of the values to be expected between individuals and the reliability of the Langham equations used for estimating the quantity of plutonium in humans. This review utilizes the excretion data tabulated by Langham, et al. A review of the blood concentration following intravenous administration was made using data from the same source. The present curve extends to only 2.5 hours after the incident and is derived from only a particular type of administration. Within these limits it appears that blood samples may well serve as an early criterion of plutonium absorbed to the body.

CONDENSED EXPOSURE RECORDS

Type	Number of Readings	Potential High Results	Confirmed High Results
Pocket Chambers - Gamma	268,894	41	0
Pocket Chambers - Slow Neutron	2,116	0	0
Film Badges - Beta - Gamma	55,944	58	0
Film Badges - Neutron	918	0	0
Pu Bioassay	1,310	50	0
F.P. Bioassay	1,553	3	0
U. Bioassay	400	1	1
Alpha Hand Counts	63,079	0	0
Beta Hand Counts	55,682	0	0
Thyroid Counts	55	0	0

REGIONAL MONITORING

The general findings are summarized in the following:

Sample Type and Location	Activity Type	Average Activity Density μc/ml	Trend* Factor
Drinking Water and Related Materials			
Benton City Water Co. Well	alpha	1.1×10^{-8}	---
Richland, N. Richland, Benton City Wells	alpha	$(<0.5 \text{ to } 1.1) \times 10^{-8}$	---
100 Areas	beta	$(0.04 \text{ to } 1.4) \times 10^{-5}$	+3
200 Areas	beta	$(1.1 \text{ to } 1.5) \times 10^{-6}$	---
Pasco, Kennewick, McNary Dam	beta	$(<0.05 \text{ to } 2.4) \times 10^{-6}$	---
Backwash Solids - Pasco Filter Plant	beta	3.3×10^{-2} μc/gm	---
Backwash Liquids - Pasco Filter Plant	beta	2.4×10^{-6}	---
Anthracite, Sand Filter - Pasco Filter Plant	beta	1.1×10^{-4} μc/gm	---
Other Waters and Related Materials			
300 Area Wells Nos. 1, 3 and 4	U	$(1.7 \text{ to } 1.9) \times 10^{-7}$	---
200 East Wells	beta	$<1 \times 10^{-7}$ to 1.5×10^{-1}	+2
200 West Wells	beta	$<1 \times 10^{-7}$ to 1.4×10^{-2}	---
Wells Near 200 Areas	beta	$(<1 \text{ to } 2.3) \times 10^{-7}$	---
107 and 108 Wells	beta	$<1 \times 10^{-7}$ to 9.6×10^{-5}	+6
Outlying Wells	beta	$<1 \times 10^{-7}$	-2
Columbia River - Hanford Ferry	beta	3.2×10^{-5}	---
Columbia River - Below Reactors	beta	2.4×10^{-5}	---
Columbia River - Paterson to McNary	beta	5.6×10^{-7}	---
Columbia River - Shore Mud	beta	$(0.2 \text{ to } 1.9) \times 10^{-4}$	---
Raw Water-Operating Areas	beta	$(0.005 \text{ to } 1.7) \times 10^{-5}$	---
Reactor Effluent Retention Basins to River	beta	7,800 to 34,000 μc/sec/reactor $(3.4 \text{ to } 7.5) \times 10^{-3}$	--- ---
Reactor Effluent Retention Basins to River	alpha	<0.04 μc/sec/reactor $<5 \times 10^{-9}$	--- ---

* The trend factor shows the n-fold increase (+) or decrease (-) from last month, where values of n less than 2 will not be noted.

Sample Type and Location	Activity Type	Average Activity Density μc/ml	Trend* Factor
Other Waters and Related Materials (contd)			
I-131 in Farm Wastes to River		38 μc/day	+2
	I-131	6.7×10^{-7}	+2
I-131 in Columbia River-Hanford	I-131	6.4×10^{-4}	---
300 Area Pond Inlet	alpha	8.4×10^{-7}	---
Atmospheric Pollution			
Gross Alpha Emitters	alpha	$(<4 \text{ to } 8) \times 10^{-15}$	---
Gross Dose Rate - Separations Areas	beta-gamma	2.5 to 6.0 mrad/day	---
Gross Dose Rate - Residential Areas	beta-gamma	0.7 to 2.8 mrad/day	-4
Active Particles - Separations Areas	beta	$(1.4 \text{ to } 4.3) \times 10^{-13}$	-2
I-131 Separations Areas	I-131	$(0.4 \text{ to } 3.2) \times 10^{-13}$	---
I-131 Separations Stacks	I-131	1.4 curies/day	+7
Ruthenium - Separations Stacks	Ru-103-106	<0.02 curie/day	---
Active Particles -Wash. Idaho, Ore., Mont.	---	0.02 to 0.1 ptle/m^3	-6
Active Particles - Project	---	0.006 to 0.1 ptle/m^3	-3
Vegetation			
Environs of Separations Areas	I-131	$(<3 \text{ to } 5.5) \times 10^{-6}$ μc/gm	---
Residential Areas	I-131	$(<3 \text{ to } 3.0) \times 10^{-6}$ μc/gm	---
Eastern Washington and Oregon	I-131	$(<3 \text{ to } 3.0) \times 10^{-6}$ μc/gm	---
Non-Volatile Beta Emitters Wash. and Ore.	beta	$(0.08 \text{ to } 3.2) \times^{-4}$ μc/gm	---
Alpha Emitters - Separations Areas	alpha	$(<0.7 \text{ to } 6.6) \times 10^{-7}$ μc/gm	+2

* The trend factor shows the n-fold increase (+) or decrease (-) from last month, where the values of n less than 2 will not be noted.

LABORATORIES AUXILIARIES

The administration of landlord responsibilities was transferred to the Facilities Engineering Operation. This assignment will relieve research managers of building operation details and improve costs control of building maintenance.

Radiographic testing of welds in construction materials for Project CG-558 continued. Radiography of the piping at D Reactor was completed.

Two stainless steel Class I process vessels, an H-4 tank for Redox and a K-2 extraction column for Purex, were examined.

Exposures were made on electrical splices in the Hanford high tension electrical distribution system. A specially designed film holder was placed in the 50-foot high lines and a gamma - radiograph was taken at distances up to 35 feet.

Eddy currents were used to test titanium tubes suspected of being cracked. A probe coil generating and analyzing eddy currents was passed through the test pieces. Close correlation with previous radiographic results was achieved.

Project Number	Title	Using Component	Est. Total Project Cost	Authorization Information Amount-Date	Project Progress in Per Cent Design	Const.	Scheduled Project Comp. Date	Remarks and Progress
PLANT AND EQUIPMENT PROJECTS-FY 1956 (AEC 2-23X-6009F)								
CG-682	High Level Examination and Cut Off Cell-327 Bldg.	Reactor and Fuels	$355,000	$16,500 6-6-56	Sch. 47 Act. 58	0 0	15 months after author.	Scope changed. Revised proposal requesting $30,500 total design funds being routed for signatures.
GENERAL PLANT PROJECTS (AEC 2-23X-56-L-2)								
CG-635	Redox Stack Particulate Sampler	Radiation Protection	$ 42,000	$40,000	Sch.100 Act.100	85 98	3-1-57	Construction completed except for installation of steam jet. Actual completion date 11-30-56.
CG-660	Modifications and Additions to the Metallographic Cell-327 Bldg.	Reactor and Fuels	$135,000	$135,000	N. S. Act. 55	0 0	9-30-57	Additional information requested of vendors. Further design dependent on metallograph dimensions. Design sch. to be revised after bid for Metallograph has been accepted.

Project Number	Title	Using Component	Est. Total Project Cost	Authorization Information Amount-Date	Project Progress In Per Cent Design	Const.	Scheduled Project Comp. Date	Remarks and Progress
CG-664	350 C, Flow Loop - 314 Bldg.	Reactor and Fuels	$121,500	$121,500 5-11-56	Sch. 100 Act. 100	0 0	8-1-57	Requests for bids were issued to two vendors on 11-20 for comp. Test Loop fab. and assembly will be opened on 12-20.
CG-672	Monochromatic Neutrom Beam Facility -105- KE Bldg.	Physics and Instr.	$178,500	$112,000 5-1-56	Sch. 100 Act. 100	0 0	6-30-57	Revised project proposal being prepared by CEO requesting an add. $68,000 author. and ext. of comp. date.
EQUIPMENT NOT INCLUDED IN CONSTRUCTION PROJECTS (AEC-9950)								
CG-620	Melt Plant Modifications- 306 Bldg.	Reactor and Fuels	$120,000	$143,000 3-15-56	Sch. 40 Act. 38	0 0	5-31-57	Rev. Proj. Prop. in process of appr. and distr. to Contract Admin. Vac. furnace purchase ord. has been placed.

Project Number	Title	Using Component	Est. Total Project Cost	Authorization Information Amount-Date	Project Progress in Per Cent Design	Const.	Scheduled Project Comp. Date	Remarks and Progress
CG-661	Additional Heat Gener. Facility 189-D Bldg.	Reactor and Fuels	Not Determ.	$ 22,400	Sch. 100 Act. 100	0 0	To be established	Preliminary design appr. by AEC. Detailed design contd.
CG-681	Hanford Equip. in The ETR	Reactor and Fuels	$1,200,000	$ 80,000	Sch. 23 Act. 42	0 0	To be established	Rev. Proj. prop. will be submitted for GE appr. on 11-30. Proj. to be AEC managed.
PLANT AND EQUIPMENT PROJECTS FY 1957 (AEC-2-28-57-N-2 GEN. PLANT PROJ.)								
CG-658	Shielded Personnel Monitoring Station	Physics and Instr.	$ 150,000	Dir. AEC-97 11-28-56	Sch. 0 Act. 0	0 0	Awaiting work author.	Project proposal awaiting AEC approval.
CG-680	Corrosion Testing Facilities-314 Bldg.	Reactor and Fuels	$ 140,000	$ 29,500 9-24-56	Sch. 28 Act. 32	0 0	1-24-57	Work release revised for design only CEO for $17,400 due to limited Gen. Plant Funds.
CA-685	Alterations to Bldgs. 325 and 326.	HLO	$ 23,000	$ 23,000 9-5-56	Sch. 100 Act. 100	0 0	8-15-57	Acceptable bid on elev. controls returned by vendor.
CA-700	Geological and Hydro. Wells.	Chemical Research	$ 137,000	$137,000 10-29-56	Not Sch. Act. 5	0 5	12-1-57	First well by USGS near compl.

Project Number	Title	Using Component	Est. Total Project Cost	Authorization Information Amount-Date	Project Progress in Per Cent Design	Const.	Scheduled Project Comp. Date	Remarks and Progress
B-5772	High Level Exposure Facility. Addition 141-H Bldg.	Biology Research	$ 26,000	Pending Pending	Sch. 0 Act. 0	0 0	To be established	Proposal forwarded to Contr. Adm. accomp. by letter requs. author. be withheld until funds become available.
B-5776	Effluent Eng. Test Facility	Chemical Research	$125,000	Pending Pending	Sch. 0 Act. 0	0 0	To be established	Prop. prep. halted because of lack of Gen. Plant Proj. Funds. Prop. reacticat ed.
A-00617	Ventilation Improvements. 222-U Bldg.	Chemical Research	$ 73,000	Pending Pending	Sch. 0 Act. 0	0 0	To be established	Proposal forwarded Contract Admin.
	Biology Controlled Activity Water System.	Biology Research	$ 15,000	Pending Pending	Sch. 0 Act. 0	0 0	To be established	Project Proposal to be initiated 12-56.
	Modifications - 321 Bldg. - Fluoride Studies	Chemical Research	$ 75,000	Pending Pending	Sch. 0 Act. 0	0 0	To be established	Project Proposal data being accumulated.
EQUIPMENT NOT INCLUDED IN CONSTRUCTION PROJECTS (AEC-2900)								
CA-695	Radio Telemetering Network	Physics and Instrs.	$ 95,000	Pending Pending	Sch. 0 Act. 0	0 0	To be established	Awaiting AEC Approval.

Project Number	Title	Using Component	Est. Total Project Cost	Authorization Information Amount-Date	Project Progress in Per Cent Design	Const.	Scheduled Project Comp. Date	Remarks and Progress
NEW CONSTRUCTION - FY 1957, FUNDS YET TO BE AUTHORIZED								
None	Plutonium Metallurgy Facility Research	Research and Fuels	$347,000	none to date none to date	Sch. 0 Act. 0	0 0	To be established	Proj. Prop. revised and requires approvals. To date adm. of this proj. by Reactor and Fuels person.
None	1706 KE Expansion	Chemical Research	$ 40,000 HLO portion	none to date none to date	Sch. 0 Act. 0	0 0	To be established	This work to be combined with IPD sponsored proj. on 1706 KE and to be financed and managed by IPD.
NEW CONSTRUCTION-FY 1958								
B-57129	Critical Mass Laboratory	Physics and Instr.	$2,000,000	none to date none to date	Sch. 0 Act. 0	0 0	To be established	Proj. prop. being routed for approvals.

EMPLOYEE RELATIONS

On November 30, the staff of the Hanford Laboratories Operation totalled 1160 employees, including 473 exempt and 687 non-exempt personnel. Of the total

Benefit Plans. HLO participation in the benefit plans follows:

	November	October
Pension Plan	97.5%	97.0%
Insurance Plan	99.3	99.1
Savings and Stock Bonus Plan	58.6	57.5
Savings Plan	8.0	8.5

Selective Service. Hanford Laboratories Operation currently has 212 employees subject to Military service of which 95 are Reservists or National Guard members. 40 are technically trained or engineering personnel for whom deferments have been granted or are being processed and 77 are non-technical non-veterans of whom 17 are classified 1-A.

Placement and Records. Five attendance recognition pins were awarded during the month.

Non-exempt employment and transfer activities are summarized below. There are currently 45 openings for non-exempt personnel of which 18 are for males. Little difficulty is anticipated in filling these vacancies. There are 3 secretarial or stenographic openings. The remainder of the 27 female openings are for General Clerks and Personnel Meters Clerks.

EMPLOYMENT - NON-EXEMPT

Requisitions open at end of month	45
Requisitions filled	20
Requisitions cancelled	2
Requisitions received	18
Applications considered	2
Active transfer cases at end of month	51
New requests*	6
Transfers effected	15

*All new requests for transfer have been discussed with the applicant.

Of the 51 requests for non-exempt transfers, approximately 50% represent cases which can be processed under the transfer procedure. The remainder fall into categories where the desired benefits of the transfer are not gained on the basis of a request by the employee.

Technical Personnel Placement

Technical recruiting and transfer activities are summarized below. Early returns from the PhD recruiting effort indicate that the supply is somewhat more limited than during the past year. Competition continues to increase in this area. PhD metallurgists, physicists, mechanical engineers, and physical chemists possessing the qualifications of interest to HLO represent the most critical areas. Experienced BS/MS recruiting efforts appear slightly more favorable. An increased number of recent graduates possess minor experience and are being referred for direct placement as contrasted with training program placement.

PhD Recruiting

	Cases Considered	Visits to Richland: Extended Visit Inv.	Visited	Rejected	Offers: Extended	Accepted	Rejected	On The Roll
Engineering:								
Electrical	1	1	1		1			
Mechanical	1	1	1		1	1		
Metallurgy	3	2	2*		1		1	
Science:								
Chemistry	3	3	3		3	1	1	1
Physics	4	3	3		3	2		2
Plant Physics	3	2	2		1			
DVM	2	1	1		1			1

* Includes one case - no offer made

BS/MS Experienced Recruiting

	Cases Considered	Visits to Richland: Extended Visit Inv.	Visited	Rejected	Offers: Extended	Accepted	Rejected	On The Roll
Engineering:								
Architectural	2	1						
Ceramic	2	1	1		1	1		1(10/1/56)
Chemical	4	1	1					
Electrical	5	1	1		1	1		1(10/8/56)
Industrial	1		1		1	1		1
Mechanical	8	4	1	2	1	1		1
Metallurgical	3	3		1				
Science:								
Bacteriology	2*							
Biology	1*							
Botany	1*							
Bus. Adm.	2	1	1					
Chemistry	3	2	1	1	1		1	
Math-Stat.	2	2	2		2	2		2(9/20/56)
Physics	4	3	2		1		1	

* No HLO Interest

TRANSFER CASES-EXEMPT

	November	October
Total Cases Handled since 9-1-56	45	41
Initiated by employee	31	30
Initiated by management	14	11
Active Cases at end of month	24	31
New Cases during month	4	10
Initiated by employee	1	5
Initiated by management*	3	5
Cases closed during month	11	7
Transfers effected within HLO	0	0
Transfers effected within HAPO	0	0
Transfers effected to other GE	1	2
Requests withdrawn	9	3
Terminations	1	2

* Includes ROF's, transfers proposed by employee's management and requests from other GE departments.

Union Relations

The certification election for Field Inspector's "A", "B", and "C", Regional Monitoring was held on November 26, 1956. The vote was 19-1 in favor of Union affiliation.

One new grievance was received during the month bringing the total since September 1, 1956 to 10. This grievance was settled satisfactorily at Step I. A summary of active grievances follows.

Grievances Processed September 1, 1956 to date.

Total Processed — 10 (includes 3 carryovers from preceding year).

Step I
- Pending Step I answer 0
- Answered Satisfactorily 3

Step II
- Pending Step II discussion* 0
- Pending Step II answer 2
- Answered Satisfactorily** 3
- Answered Unsatisfactorily
  - Pending Time Limit*** 1
  - Applied for Arbitration 1

* Step I grievances which Council indicated a desire to discuss at Step II but not scheduled for discussion within three months are considered settled at Step I.

** Step II grievances in which the Council formally applied for arbitration but for which no further action is taken within three months are considered settled at Step II.

*** Step II answers which HAMTC has informally indicated are unsatisfactory but for which they do not submit formal application for arbitration within 15 days are considered to be satisfactory answers.

Health, Safety and Security

Laboratories personnel worked a total of 179,516 employee hours during the month with no disabling injuries. There were 40 medical treatment cases with a frequency of 2.23 for the month as compared with 1.43 for the previous month. Medical treatment frequency since September 1, 1956 is 2.04.

There were three fires during the month, two in laboratory hoods and one in a vehicle. An extensive water leak occurred in one of the laboratory buildings. No significant property damage resulted.

The HLO Health and Safety Council has initiated a program for accumulating and centralizing information on technological hazards.

Three HLO Safety Suggestions were evaluated and one evaluation was continued pending further investigation. Two of the suggestions were adopted.

There were five security violations reported during the month, bringing the total since reorganization to 18. This may be compared with a total of 9 violations during the month of October.

FINANCIAL

An estimate of the number of HLO employees at the close of each quarter was submitted to Contract Administration. The Mid-Year Budget Review estimates of operating costs of HLO components were completed and reviewed with the Level 3 Managers.

The Chemical Processing Department transferred to HLO responsibility for research and development budgeted at $317,000 for FY 1957.

The issuance of 3" x 5" reports to HLO managers was initiated in November. These reports contain information on financial and other activities and will be issued monthly hereafter.

Property Record Unit Catalogs were distributed to all HLO property control custodians. Each copy is registered to provide control and insure maintenance on a current basis.

In accordance with an agreement between the Company and HAMTC, isolation pay will be included as part of employees' normal straight time earnings for administering the Insurance Plan. Cost to the employee will increase slightly with a substantial increase in benefits. This became effective December 1, 1956.

The gross payroll paid during the month was $598,752, of which $331,351 was paid to exempt employees and $267,401 to non-exempt employees. Payment to non-exempt employees represents four weeks. The gross payroll paid during October was $587,016.

VISITS TO OTHER INSTALLATIONS

Name	Dates of Visit	Company Visited and Address	Reason for Visit	Personnel Contacted	Access to Restricted Data
RE Heineman	11/1/56	Mass. Inst. of Tech. Cambridge, Mass.	Recruit Ph. D. 's	---	No
DE Jenne	11/1/56	US Weather Bureau Records Center Asheville, N. C.	Confer with USWB personnel on meteorological observations and data processing.	L Smith	No
DC Fleckenstein	11/1-2/56	Reed College Portland, Oregon	Conference regarding AFSWP training course.	AF Scott KF Oerlein	No
HW Lefevre and BR Leonard, Jr.	11/1-2/56	Gatlinburg, Tenn. (sponsored by Oak Ridge Nat'l Lab)	Attend Conference on Neutron Physics by Time-of-Flight	---	No
JM Batch, DJ Foley, LH McEwen	11/1-2/56	AEC Heat Transfer Conference, New York, N. Y.	To attend AEC sponsored reactor heat transfer conference and present a paper (Batch).	JE Viscardi	Yes
WJ Bair	11/1-2/56	Reed College, Portland, Oregon	Consult with officals on AFSWP Training Program.	Dr. Scott	No
JW Healy	11/1-2/56	Reed College, Portland, Oregon	Discuss training of AFSWP Officers.	Dr. Scott Dr. Oerlein	No
	11/13/56	Washington, D.C.	Meeting on decontamination and countermeasures.		Yes
SH Bush	11/2/56	GE Metallurgical Detroit, Mich.	Discuss refractory and ceramic materials	WE Reich	No
	11/3/56	Trent Tube, East Troy, Wisc.	Discuss zirconium tube procurement	H Bowman	No
	11/4-9/56	Univ. of Mich.	PhD recruiting		No
TK Bierlein	11/4-6/56	GERL, Schenectady	Discuss technical problems	DW Lillie	No
	11/7-8/56	Bridgeport Brass Co., Bridgeport, Conn.	Present paper at metallography mtg.	DW White, KAPL	Yes

Name	Dates of Visit	Company Visited and Address	Reason for Visit	Personnel Contacted	Access to Restricted Data
TK Bierlein	11/26-30/56	North American Philips, Mt. Vernon, N. Y.	Attend course in electron microscopy	Mr. Woods	No
RE Nightingale	11/5-9/56	Wash. University, St. Louis, Mo. Univ. of Arkansas Fayetteville, Ark.	Technical recruiting	---	No
DR deHalas	11/5/56	Calif. Research, Richmond, Calif.	To discuss organic coolant technology.	JG Carrol	Yes
	11/7-9/56	Atomics Internat'l. Canoga Pk., Calif.	Same as above and attend meeting on ORE and SRE.	DW Bareis	Yes
JH Rector	11/6-7/56	Buckner-Weatherby Co. Seattle, Wash.	Discuss and observe operation of drill press.	J McCullough	No
DP Granquist	11/6-8/56	Oak Ridge Nat'l Labs. Oak Ridge, Tenn.	Consultation in connection with PRP	FR Bruce CH Secoy	Yes
JW Lingafelter	11/7/56	Allegheny Ludlum Albany, N. Y.	Observe fabrication of zirconium.	H Rohrabaugh	No
LA Hartcorn	11/7-8/56	Bridgeport Brass, Bridgeport, Conn.	AEC Metallographic Group Meeting.	DW White	Yes
	11/9/56	KAPL, Schenectady, N. Y.	Metallography discussions.	TF Fisher	Yes
	11/12-13-14/56	Savannah River Lab., Augusta, Ga.	Meeting of Cooperative Metallographic Group of the Metal Quality Committee.	W West	Yes
RE Brown	11/8-9/56	Pac. NW Sect. Amer. Geophysical Union annual mtg., Seattle, Wash.	Present paper "The Geological Conditions at the Hanford Works Regulating the Disposal to Ground of Radioactive Liquid Wastes"	---	No
	11/7/56	State of Wash. Div. of Mines and Geology, Olympia, Wash.	Discuss geological problems of Pasco Basin.	S Glover	No

Name	Dates of Visit	Company Visited and Address	Reason for Visit	Personnel Contacted	Access to Restricted Data
RE Brown	11/10/56	Dept. of Geology, Univ. of Washington Seattle, Wash.	Discuss geological problems of Pasco Basin	J H Mackin F Neumann	No
AR Keene	11/7-9/56	General Electric Schenectady, N.Y.	Attend conference on radiation protection.	WA McAdams	No
MW McConiga	11/7-9/56	Wash. Div. of Mines and Geology Seattle, Wash.	Discuss findings on the Ringold formation.	S Glover	No
		NW Regional Amer. Geophysical Union, Seattle, Wash.	Attend 3rd Annual Meeting	W. Smith G Klein	No
MK Millhollen	11/7-9/56	Phillips Petroleum Co., Arco, Idaho	Observe experiment in MTR	D Alvord	Yes
JJ Fuquay	11/8-9/56	Univ. of Wash., Dept. of Meteorology and Climatology, Seattle, Wash.	Confer with Staff on micrometeorological problems.	DE Church FI Badgley R Fleagle	No
PFX Dunigan	11/8-9/56	E. I. DuPont Savannah River Lab. Aiken, S. Carolina	Hot Laboratory Committee Meeting	WJ Laird, Jr.	Yes
JR McHenry	11/8-10/56	NW Regional Meeting of the American Geophysical Union, Seattle, Wash.	To present paper "Chemical and Radioactive Tracers in Groundwater Studies"	Personnel of Amer. Geophysical Union	No
PP Rowe	11/8-10/56	Same as above	To attend Amer. Geophysical Union Meeting.	Same as above	No
ED McClanahan	11/11-16/56	BNL, L.I., N.Y.	Discuss construction and operation of titanium vacuum pump.	CL Gould	Yes
		KAPL, Schen., N.Y.	Same as above	WD Davis	Yes
HA Paulsen	11/12/56	General Eng. Lab., Schen., N. Y.	Salary Reconciliation	H Maxwell	No
	11/13/56	KAPL, Schen., N.Y.	Same as above	RZ Bouton	No
	11/13/56	Research Lab., Schen., N. Y.	Same as above	LW Steele	No

Name	Dates of Visit	Company Visited and Address	Reason for Visit	Personnel Contacted	Access to Restricted Data
HA Paulsen	11/14/56	Flight Propulsion Lab. AGT., Cincinnati, Ohio	Salary Reconciliation	W Mendell	No
	11/15-16/56	GE Co., AGT., Cincinnati, Ohio	Salary and Wage Administration Workshop	W Mendell Chairman	No
RM Fryar	11/13-14/56	Purdue University Lafayette, Indiana	Recruiting trip		No
CA Bennett	11/13-14/56	AEC Div. of Research Wash., D. C.	Discussion of Project Bluenose	SG English	Yes
	11/30/56	Same as above	Same as above	SG English	Yes
M Lewis	11/13/56	Battelle Memorial Institute, Columbus, Ohio	Technical discussions on graphite.	LD Loch	Yes
	11/16/56	Brookhaven Nat'l. Lab., Upton Long Island, N. Y.	Same as above	RA Meyer	Yes
TG Marshall	11/14-15/56	Argonne Nat'l. Lab. Chicago, Illinois	Professors Institute Courses	R Taecker	Yes
	11/16/56	Div. Reactor Dev. Wash., D. C.	Professors Institute	JG Kaufmann	Yes
	11/19-21/56	GE Company, N. Y. New York.	Conference Employee Relations Managers	WV Merrihue	No
LP Bupp	11/14-15/56	Univ. of Utah, Salt Lake City, Utah	Technical recruiting (PhD)	---	No
WJ Ozeroff	11/15-16/56	Atomic Power Equip. Dept., San Jose, Calif.	Discuss new job assignment.	RD Bennett	No
BG Lindberg	11/16/56	Society of Non-destructive Testing Seattle, Wash.	Give Lecture	---	No
RB Socky	11/16/56	Engineer's Club, Artic Bldg., Seattle, Wash.	Program Chairman Present program on "The Application of Radioactive Isotopes"	Society for Non-destructive Testing Members	No

Name	Dates of Visit	Company Visited and Address	Reason for Visit	Personnel Contacted	Access to Restricted Data
JM Tobin	11/16/56	ANL, Lemont, Ill.	Attend conference on metallurgical prob.	ES Fisher	Yes
	11/17-21/56	Mich. State, Lansing	PhD recruiting	J Wolfe, GE	No
RL Dillon	11/16/56	KAPL - Schen. N.Y.	Discuss corrosion of uranium, aluminum and other materials.	RM Haag E L Brady	Yes
	11/19/56	WAPD - Bettis Field Pittsburgh, Pa.	Same as above	B Lustmann	Yes
ID Thomas OJ Wick	11/17-21/56	UCRL, Livermore, Calif.	Discuss fabrication problems	W Ramsey	Yes
WL Nicholson	11/29-30/56	Gen. Elec. Co. New York, N.Y.	Present paper at GE symposium on Statistical Methods	JH Davidson et. al.	No
PM Thompson	11/19/56	Univ. of Calif. Los Angeles, Calif.	Invited speaker	NH Jacoby	No
EJ Seppi	11/21/56	ANL- Lemont, Ill.	Discuss neutron measurements.	LM Bollinger	Yes
	11/23-24/56	Amer. Physical Soc. Chicago, Ill.	Present a paper		No
	11/26-28/56	Atomic Energy Proj. Chalk River, Canada	Discuss inelastic neutron scattering measurements.	BN Brockhouse	Yes
	11/29-30/56	KAPL - Schen., N.Y.	Discuss cross-section program.	ER Gaerttner	Yes
HA Kornberg	11/24-29/56	Oak Ridge, Tenn.	Member of a committee to review and evaluate UT-AEC agricultural program.	Drs. Roth, McCloud, Winters Ewing, Chance, Sapirie, Shoup, Morgan, Hollaender.	Yes
WR DeHollander	11/25-30/56	Univ. of Chicago Chicago, Ill.	Technical Recruiting	J Wolfe	No
HT Hahn	11/26-27/56	Univ. of Minnesota Minneapolis, Minn.	Technical Recruiting		No
	11/28-29/56	Univ. of Wisconsin Madison, Wisconsin	Technical Recruiting		No

Name	Dates of Visit	Company Visited and Address	Reason for Visit	Personnel Contacted	Access to Restricted Data
HT Hahn	11/30/56	Minnesota Mining and Manufacturing Co., St. Paul, Minn.	Consultation on inorganic exchangers	Dr. Johnson	No
JD McCormack	11/26-28/56	Third Internat'l. Automation Exhibition New York, N. Y.	To view equipment on display		No
	11/29-30/56	E.I. duPont Co., Aiken, S. Carolina	To discuss instrumentation problems and observe in-line stream instrumentation.	JM Wilson	Yes
WP Wallace	11/26-30/56	Carnegie Tech. and Univ. of Pittsburgh, Pittsburgh, Pa.	PhD Recruiting		No
OF Hill	11/26-27/56	UCLA, Los Angeles Calif.	PhD Recruiting		No
	11/28-30/56	Cal. Tech., Pasadena, Calif.	Same as above		No
N Ketzlach	11/27-28/56	National Lead of Ohio Cincinnati, Ohio	Discuss nuclear safety problems in the handling, storage and shipping of enriched uranium fuel.	JC Mead	Yes
JE Faulkner	11/29-30/56	UCRL, Berkeley, Calif.	Attend symposium on "Time Dependent Behavior of Sub-Critical Assemblies" Sponsored by J. Carothers.		No
CG Stevenson	11/29/56	Univ. of Washington Seattle, Washington	Deliver speech to Library School	I Lieberman	No

VISITS TO HANFORD WORKS

Name	Dates of Visit	Company or Organization Represented and Address	Reason for Visit	HW Personnel Contacted	Access to Restricted Data	Areas and Bldgs. Visited
F Morris and J Ball	10/30/56	Turco Products, Inc. Los Angeles, Calif.	Discuss decontamination agent-Turco 4306-B	CM Unruh	No	None
JA Spaziani	11/2/56	Manufacturing Engrg. APED, San Jose, Calif.	General discussion of fuel element manufacturing process and equipment.	RW Benoliel	Yes	300: 3760, 313, 3703, 328, 306, 305 Bldgs.
LK Keay	11/8/56	Lukens Steel Co. Coatesville, Pa.	Present paper before ASM	RB Socky JH Rector RB Switters	No	300: 327
RD Rowe and EJ Simons, Jr.	11/8/56	General Machinery Co., Spokane, Wn.	Consultations on pump development	AE Smith J Dunn	Yes	300: 321
GW Watt	11/8-9/56	Univ. of Texas Austin, Texas	Consultations on Chemical Development Program Consultant	OF Hill RE Burns	Yes	300: 328
	11/12-16/56			MT Walling LL Burger RL Moore EE Voiland WH Reas	Yes	300: 325
TS Weissmann	11/9/56	KAPL, Schen., N.Y.	Discuss fuel element development	JE Minor	No	300: 326 306, 303
KD Mann JT O'Brien	11/16/56	Firth-Sterling Pittsburgh, Pa.	To discuss zirconium fabrication.	FW Woodfield	Yes	300: 303 100: 105
RE Hulse WC Greenleaf	11/16/56	U.S. Industrial Chemicals Co., Div. of National Distillers, Pittsburgh, Pa.	To discuss zirconium fabrication.	FW Woodfield	Yes	300: 303 all 100 Areas all 105 Bldgs.
HF Beeghly	11/16/56	Jones and Laughlin Steel Company, Pittsburgh, Pa.	Discussions on carbon steel applications in recirculation coolant systems.	JM Atwood	Yes	100K: 105-KE.

Name	Dates of Visit	Company or Organization Represented and Address	Reason for Visit	HW Personnel Contacted	Access to Restricted Data	Areas and Bldgs. Visited
RS Russell	11/21/56	Atomic Energy Authority, United Kingdom, Harwell, England.	Discuss lab. activities	HA Kornberg and Staff	No	100-F: 108-F, 141-M 146-FR, 1705-F, 300 : 3760
P Berner	11/26-28/56	Star Machinery, Seattle, Wn.	Supervise the installation and instruct in operation of special grinding machine.	EA Evans	No	300 : 325
JD Kreager	11/26-28/56	Engelberg-Huller Co., Los Angeles, Calif.	Same as above	EA Evans	No	300: 325
MR Kennedy	11/28-30/56	KAPL, Schen., N.Y.	Discuss radiation protection methods	AR Keene and Staff	Yes	300 : 100-F, 108-F
	11/29/56		Discuss mutual problems assoc. with monitoring of river orgainisms.	RF Foster DE Warner	No	100-F: 146-F, 108-F
EA Yunker	11/29/56	Oregon State College Corvallis, Oregon	Discuss radiation protection method.	AR Keene	No	None
B Kennedy	11/30/56	Knolls Atomic Products Laboratory, GE, Schenectady, N. Y.	Discuss radiochemical separation analysis technique and instrumentation.	LC Schwendiman	No	300 : 325.

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
J. L. Weeks and C. R. Anderson	"Filtering Process for Fission Product Recovery. Variegated Filter Media for Fission Product Recovery."
K. J. Hahn	"Electrode Clamping Device for pH Flow Cell."
V. P. Kelly	"A Process for Removing Metal Jackets in Molten Baths."
W. L. Lyon and E. E. Voiland	"The Use of Ion Exchange in Fused Salt Systems for the Separation and Possible Decontamination of Uranium Salts."
R. L. Moore	"A Dissolver with Sacrificial Liner for Dissolution of Zirconium Clad Fuel Elements."
J. M. Tobin and J. H. Sako	"Electrolytic Recovery of Uranium from Uranium Oxide (UO_2) Fuel Material."
V. H. Troutner and C. Groot	"Preparation of Uranium Carbide."
C. Groot and H. C. Bowen	"Separation of Dense Compounds from Tars."

H. M. Parker
Manager, Hanford Laboratories