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HANFORD LABORATORIES OPERATION MONTHLY ACTIVITIES REPORT

JUNE, 1962

JULY 16, 1962

[redacted]

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

GENERAL ELECTRIC

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HANFORD 82566

HANFORD LABORATORIES OPERATION

MONTHLY ACTIVITIES REPORT

JUNE, 1962

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Compiled by
Operation Managers

July 16, 1962

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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PRELIMINARY REPORT

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

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TABLE 1 HLO FORCE REPORT

DATE: July 5, 1962

	At Beginning of Month		At Close of Month		Total
	Exempt	Salaried	Exempt	Salaried	
Chemical R & D	133	127	139	134	273
Reactor & Fuels R & D	171	156	175	170	345
Physics & Instrument R & D	95	60	97	60	157
Biology	34	58	37	57	94
Operations Res. & Syn.	18	4	19	4	23
Radiation Protection	39	93	41	101	142
Finance and Administration	95	97	120	96	216
Programming	16	2	16	3	19
General	3	4	3	4	7
Test Reactor & Auxiliaries	51	183	52	184	236
TOTAL	655	784	699	813	1,512

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BUDGETS AND COSTS

June operating costs totaled $3, 054, 000; total costs for FY 1962 were $27, 104, 000 or 97% of the $27, 863, 000 control budget. Hanford Laboratories' research and development costs for June, compared with last month and the control budget are shown below:

(Dollars in Thousands)	COST Current Month	Previous Month	FY To-Date	Budget	% Spent
HLO Programs					
02 Program	$ 47	$ 46	$ 535	$ 605	88%
03 Program	39	52	124	175	71
04 Program	1 273	716	10 923	11 024	99
05 Program	137	76	1 008	1 065	95
06 Program	297	223	2 528	2 637	96
	1 793	1 113	15 118	15 506	97
FPD Sponsored	168	82	1 351	1 375	98
IPD Sponsored	189	106	1 395	1 348	103
CPD Sponsored	133	105	1 578	1 636	96
Total	$2 283	$1 406	$19 442	$19 865	98%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

Five NPR inner fuel tubes were irradiated to 3000 MWD/T under N-Reactor operating conditions and examined in the Radiometallurgy Laboratory. The general appearance of the elements was good, but some distortion of the inner bore due to inner clad buckling was evident.

A prototype N-Reactor tube-in-tube fuel element has been successfully irradiated to goal exposure in the ETR M3 Loop. The average exposure of the fuel element at discharge is estimated to be 1200 MWD/T, and the maximum exposure is approximately 2000 MWD/T.

Two experimental braze closures of NPR fuel elements were tested, one using copper and the other a copper-tin braze alloy. Steam

[redacted]

autoclave testing showed some discoloration in the electron beam welds made over the braze region. The copper brazed closure appeared to have somewhat better corrosion resistance than the alloy brazed closure.

Scratching of an NPR process tube by NPR fuel element feet occurred on hand charging a column of fuel elements and on pulling a single fuel element through a nozzle-less tube. The effect of tube cleanliness (the tube in these tests had been flushed with filtered, deionized water and dried with cotton rags) on scratching will be investigated in future tests.

During the month seventeen UO_2-PuO_2 fuel element clusters were fabricated through final assembly.

Seven PRTR-length UO_2-PuO_2 fuel rods were vibrationally compacted in 304-L thin-wall (0.008-inch thick) stainless steel cladding to fuel densities of 87 to 89 percent of theoretical.

Loosened wire wraps were noted after irradiation on UO_2-PuO_2 rods from a 7-rod cluster. This is the first time loosening has been found on Zircaloy clad oxide elements.

Fission gas release data on irradiated cosine-enriched PuO_2-UO_2 rods showed the greatest release from rods operating at the lower temperatures.

Essentially pure PuN was made by reacting PuH_3 with nitrogen for two hours at 600 C. A sample of this material heated on a tungsten ribbon under one atmosphere of nitrogen appeared to sinter at 2270 C, volatilization without melting occurred at 2675 C increasing in rate up to 2900 C. PuN decomposes or volatilizes at 2675 C in one atmosphere of argon.

A UO_2 fuel element clad in AISI-406 stainless steel, which reportedly failed during irradiation in the MTR, was examined in the ETR hot cells. Leak testing and visual examination failed to confirm the existence of a cladding penetration.

The thermal conductivity of a UO_2 single crystal was found to increase with increasing temperature above 700 C. At 1200 C the crystal had the same value of conductivity as at room temperature.

Haynes Alloy 25 valve seat material was found not to be attacked by alkaline permanganate decontaminating agent, whereas pits 2 to 4 mils deep were found in Stellite-6. Dendritic pitting of valve seat alloys by alkaline permanganate is one of the more serious corrosion problems associated with decontamination.

The effect of tin in the range 0. 1 to 1. 5 w/o and oxygen in the range of 900 to 1300 ppm on the corrosion rate of Zircaloy-2 in 360 C water was found to be negligible. In 400 C steam lower corrosion rates were obtained for samples with high tin and low oxygen compositions.

A 1584-hour ex-reactor loop test of Zircaloy-2 corrosion under heat transfer conditions was performed, employing 580 F, deionized water adjusted to pH 10 with LiOH. In areas of the specimen with heat flux of 280, 000 Btu/hr-ft^2 and a calculated temperature of 637 F, a film layer less than 0. 1 mil thick was formed. At a reduced diameter portion of the rod specimen where the heat flux was estimated to be 360, 000 Btu/hr-ft^2, the surface temperature 660 F and surface boiling probable, a heavy film was formed.

A complete inspection of the 85 pressure tubes in PRTR was made during May and June to determine the extent of the wear-corrosion that is occurring at the contact points between the fuel elements and the pressure tubes. One tube had 78 marks, with a maximum penetration of 22 mils. The maximum penetration found on any tube was 26 mils, and 11 of the 85 tubes had one or more wear marks greater than 10 mils in depth. Tube wall thickness is about 150 mils. However, room temperature burst tests on sections of three tubes removed from the reactor do not indicate an impairment of reactor safety. The data are being analyzed, and in-reactor and ex-reactor tests are being planned to determine the cause(s) of and correct the fretting corrosion.

Creep rupture tests that are being run on Zircaloy-2 pressure tube sections from three types of N-Reactor tubing with 15, 18, and 35 percent cold work, and KER tubing with 30 percent cold work. After 300 hours at 57, 000 psi hoop stress and 300 C, the total strain of the 30 percent cold worked KER tubing is approximately four times that of the N-Reactor tubes. The fabrication history of the KER tube is similar to that of the production order, N-Reactor tubes which have not been tested as yet.

[redacted]

In in-reactor creep tests, accelerated creep during reactor outages has been observed for cold worked Zr-2 at 250 C and 30,000 psi. This effect was also observed in an earlier in-reactor test at the same stress at 310 C. The increased rate during shutdowns accounts for most of the observed in-reactor deformation.

Cold worked specimens of Zircaloy-2, representing the transverse direction and irradiated to about 10^{21} nvt at 280 C, have exhibited both delayed yielding and a dual fracture upon tensile testing at 300 C. The fracture consists of combined ductile tear and shear, and is associated with extensive, localized necking.

The first instance of growth in two directions for graphite irradiated at 600 C to high exposures was observed in this laboratory for samples of raw coke graphite. Growths of 0.048 and 0.024 percent were measured in directions parallel to and perpendicular to the molding directions.

Samples of a hot worked graphite with an apparent density of about 1.95 g/cm^3 were discharged after an exposure of 3120 MWD/AT. No net contraction occurred in the transverse direction compared to 0.05 percent contraction of CSF graphite.

Flow tests were conducted in the hydraulics laboratory to determine the hydraulic characteristics of the fittings, fuel, and process tube assembly which are proposed for use in retubing K Reactors with Zircaloy tubes.

Flow tests to determine the effect of over-tightening the front hydraulic connectors on C Reactor process tubes showed that typical necking down of the fittings would cause only a small increase in the pressure drop and thus a small decrease in the flow.

Additional boiling burnout points were determined with electrically heated models of 19-rod bundle fuel elements to determine the effects of rod spacing. The burnout heat fluxes for 0.050 inch spacing between rods were lower than those for the 0.075 inch spaced test section by factors up to two but were nearly twice those for a 0.015 inch spaced test section.

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Thermal cycle and pressure drop tests on the rupture loop in-reactor test section showed that seal leak rates were low and pressure drops acceptable.

Leak tests of a new ferrule type gas seal for the PRTR inlet bellows to pressure tube gas seal showed a very low helium leakage rate compared to the existing PRTR gas seals.

As the basis for calculation of performance of boiling metal cooled fast reactors for spacecraft missions, a conventional fuel-pin-type reactor has been selected for study. The core is a right cylinder, 40 cm in diameter and height, containing 3175 3/16-inch fuel pins on a 0.233-inch triangular spacing with material proportions (by volume) in the core of 50: 35: 15 coolant (potassium), fuel (plutonium ceramic), and structural material..

A study of the feasibility of increasing PRTR power level indicates that the power may be increased to about 100 MW without alteration of major equipment, though some changes will be required in lesser items of equipment, and more detailed analyses of certain transients and accidents will be required to determine the necessity for, or type of, change in several cases.

2. Physics and Instruments

Critical Mass experiments were done with a light water reflector around the 14" diameter criticality vessel. The range of concentration of plutonium nitrate in the vessel was chosen to overlap that used in the Hanford P-11 experiments which were done some ten years ago. Preliminary results indicate that there will be agreement between the new and old data, thus increasing confidence in the correctness of the information.

Studies of safety rod action in production reactors advanced with the successful simulation on the analog computer of a reactor model consisting of eleven coupled regions. This advance was made possible by a substantial improvement in the computer reliability resulting from a preventive maintenance program recently started. Most of this maintenance work is carried out on second shift when the computer is not normally in operation.

[redacted]

Data on reactivity changes which would result from various non-standard conditions in the NPR have been obtained with the completion of analysis of all exponential experiments completed to date. Loss of coolant water produces a substantial decrease in reactivity. Various conditions of flooding of the graphite stack produce changes ranging from a moderate reactivity decrease to a slight increase well within the capacity of the control system.

Satisfactory operation of the gamma spectrometers on the NPR Fuel Failure Monitor is indicated by preliminary tests on a prototype. Meanwhile assistance was given in the design of Monitor system and also on neutron sensitive ion chambers for use at NPR.

Analysis of PRTR Pu-Al fuel element 5075, which was exposed to 49.5 MWD, has been completed by Analytical Laboratories. These burnup data are now being evaluated. At this point, the Cs-137 data seem to be the most consistent. The fourth of the six low exposure Pu-Al elements has received its scheduled exposure, and has been removed from the PRTR. An analytical model of the PRTR continues to be developed, and when completed will be used in evaluating various experiments in the reactor including gross operating characteristics.

A three-group, one-dimensional diffusion theory analysis has been completed on the heterogeneous 1.82 w/o Pu-Al rod experiments in H_2O. The calculated critical numbers of fuel rods agree within 3% of the experimental numbers and the effective multiplication constant within one percent. These are considered to be in excellent agreement. Similarly satisfactory results were previously obtained when the method was applied to the 5% Pu-Al rod experiments.

Analysis work continued on k_∞ measurements on graphite lattices fueled with Pu-Al alloy. The calculational techniques used have not been able to reproduce the experimental values of k_∞ for a 10-1/2-inch lattice.

Analysis also continued on experiments done to measure the effective resonance integral of Pu^{240} relative to the dilute resonance integral. Tentative results indicate that the effective resonance integrals for several Pu-240 concentrations will be determined to an accuracy of 5-10%.

[redacted]

Support has been provided on hazards analyses pertinent to the operation of the Plutonium Recycle Criticality Facility. Experimental work is going forward aimed at preparing for an absolute determination of the power level of the PRCF.

PCTR measurements were made to try to determine the boron and plutonium contents in fuel pieces which are to serve as standards for Phoenix fuel irradiation experiments. Analysis of the results is in progress.

A series of critical mass calculations for fast spectrum, rubidium cooled cores have been performed in connection with studies conducted on space application by Reactor Engineering Development Operation. Of the three fuels studied, PuO_2, UO_2, and PuN, the PuN requires the least critical fuel volume fraction. UO_2 was the highest, being approximately twice PuN.

Neutron cross section measurements continued with both thermal and fast neutrons. Quasi-elastic scattering of 0.15 ev neutrons from light water was studied using the three-axis crystal diffraction neutron spectrometer. Total neutron cross section data in the range 3-15 Mev were obtained on bismuth, indium, copper, sodium and lithium using time-of-flight techniques.

The single frequency laboratory eddy current testing equipment was modified to permit faster sensing of null conditions with a single control. The technique is adaptable for use in the multiparameter tests.

Investigation continued on the use of drilled holes instead of notches for standardizing and calibrating the ultrasonic test of thin-walled fuel element sheath tubing. An unexplained difference in response with the drilled hole was encountered with a line-focused transducer, but not with the flat rectangular transducer.

Measurements of the radioactivity in Alaskan eskimos were begun. Initial results confirm the findings of other investigators in their studies of Laplanders that certain Arctic ecological conditions lead to body burdens of fallout material an order of magnitude higher than for most Europeans or Americans.

[redacted]

A prototype direct-reading sampler for use with Atmospheric Physics' fluorescent tracer techniques was field tested with encouraging results. Concentrations of less than 2×10^{-6} gms of zinc sulfide tracer per cubic meter of air were detected. In many experimental studies this device could replace the present more time-consuming method of catching the tracer on a filter with subsequent laboratory analyses and also opens the possibility of measuring the variation of concentrations with time.

A study of the mechanism of failure of the recharging type pocket dose meters indicated that the .0005 inch diameter fiber was making electrical contact with the center rod at about 1000 r accumulated dose.

3. Chemistry

A reactor tube study of the effect of addition of silicate on radioisotope production in the effluent water was interrupted after five days and normal process water was fed to the experimental tube for five days. When addition of 10 ppm silicate was resumed, effluent concentrations of the pertinent radioisotopes were only 20 to 30 percent lower than in the control tube. When pH adjustment was abandoned (laboratory data having indicated that the conditions necessary for adjusting pH from 7 to 6.6 lowered the effective concentration of silicate) effluent concentrations of As-76 and P-32 dropped to levels about half that of the control tube and remained at this level for two weeks.

Use of 20 ppm silicate and application of a separate nitric acid addition to maintain a pH of 7 is now under test. Initial results show a three-fold reduction in As-76 concentration.

Analysis shows higher carbon-14 release rates for KE than for KW reactor, as expected from differences in pile gas composition. Release rates were 0.068 curie per day from KE and 0.024 curie per day from KW.

A program of stress measurements in moving beds of anion exchange resins was completed except for comparative measurements with incompressible solids. Nitrate forms of 10-20 mesh and 20-40 mesh Permutit SK and 50-100 mesh Dowex-1 were studied. Results for the 20-40 mesh Permutit SK were well correlated by theory on the assumption of constant coefficient of friction between resin and wall and

[redacted]

constant ratio of radial to vertical (axial) stress. Results for the other resins were less well ordered and in particular the coefficient of friction varied with the resin history. In no case was the coefficient of friction independent of the resin velocity.

Hot cell runs disclosed no difficulty in achieving 98 to 99% removal of cesium from Purex waste tank supernatant by precipitation as cesium nickel ferrocyanide. However, attempted metathesis with a slurry of silver carbonate was unsatisfactory, presumably because of difficulties in solids transfer. Metathesis with soluble silver compounds is now under study.

Isotopic analysis of fission product cesium and rubidium recovered from Purex tank 103 A supernatant showed no dilution of either radioisotope by their natural counterparts. Isotopic analysis of recently recovered strontium product showed 53 atomic percent strontium-90 vice 56 percent in earlier products, suggesting increased dilution by natural strontium.

Miniature pulse column runs testing dipicrylamine extraction of cesium from simulated Purex tank 103 A supernatant achieved satisfactory hydraulic performance and 98% recovery of cesium at equal flows of aqueous and organic when the feed was diluted with an equal volume of water and 0.01 M DPA in 50% nitrobenzene - 50% tetralin was used as the solvent. Recent studies show weakly dissociated organic acids can be substituted for ammonium salts as scrub agents to achieve desired separation of cesium from sodium in countercurrent dipicrylamine extraction. Separation efficiencies are somewhat reduced with organic acid scrubs but the added step of volatilizing ammonium salts from the cesium product should be obviated.

Studies continued of zeolite-type inorganic exchangers for removal of cesium from Purex formaldehyde-treated waste. Using simulated FTW (adjusted to pH 3.5 with sodium hydroxide and sodium citrate and thus simulating the raffinate from strontium and rare earth solvent extraction) loadings to 50% cesium breakthrough required passage of from 131 to 280 column volumes of feed for five different exchangers tested. This would allow removal of the cesium from one ton of uranium on 0.5 to 1 gallon of these exchangers.

Installation of the Radiant Heat Spray Calciner in A cell was delayed by the recent strike of northwest construction workers but is now about 50% complete.

Performance in cold tests of the Hanford-developed electrostatic bubble scrubber (to be used in the hot cell calciner off-gas train) has been gratifying. Using magnesium oxide "smoke" generated by burning magnesium ribbon, dust loadings were reduced by the scrubber to the "ultra pure" range of 15 to 100 micrograms per cubic foot, depending on gas flow rate.

Recent studies show significantly different behavior for the two rare earths, europium and promethium, in molten chloride salt systems. Precipitation of PuO_2 from molten 2.6 LiCl-KCl yielded a decontamination factor of 600 for promethium vice 60 to 110 for europium. Electrodeposition of UO_2 produced decontamination factors of 1200 for promethium and 135 for europium.

Previous work having shown electrolytic heating to be capable of sustaining a chloride salt bath in a molten condition, a method for initiating melting by AC electrolysis was tested. Using one movable electrode in conjunction with a fixed electrode with a movable shroud, expenditure of 1200 watts allowed 10 pounds of salt to be melted in 15 minutes.

Voltametric measurements of the cathodic reduction of plutonium (III) to metal in molten KCl-LiCl eutectic show the process occurs in two steps: $Pu(III) + e^- \rightarrow Pu(II)$ and $Pu(II) + 2e^- \rightarrow Pu(0)$.

X-ray diffraction measurements of two segments of a UO_2 crystal, one unirradiated and the other irradiated to 5000 MWD/T show a change in lattice spacing from 5.4710 Å to 5.4723 Å on irradiation. An extrapolation method using the X-ray spectrometer was employed for this measurement. Quicker than the photographic technique, this method appears to be accurate to ± 0.0001 Å.

Scouting studies indicate X-ray fluorescence techniques may be useful in monitoring nickel diffusion bonds on uranium. Calibration curves with nickel foils indicate nickel film thicknesses 2 mils or less could be measured with a sensitivity of about 0.5 microns by attenuation of fluorescent uranium X-rays. Measurement of nickel fluorescence should allow measurement of the thickness of nickel films 0.2 mils or less with very high sensitivity.

4. Biology

By utilizing neptunium radionuclides of different specific activities, a marked effect of concentration of neptunium on its GI uptake was observed. At very low mass concentrations, the absorption was less than at higher.

An interesting observation was made on the development of the "secondary disease syndrome", which occurs when "foreign" bone marrow is employed for radiation protection. Mice, which were later to become doners of bone marrow, were injected when less than one day old with spleen fragments from a second, genetically dissimilar, strain of mouse. Bone marrow from these pretreated donors afforded protection when injected into irradiated animals of the second strain, without subsequent development of the secondary disease.

In the strontium toxicity experiment using miniature swine, new animals are being added to the 125 µc Sr-90/day group, because of the scarcity of readily observed effects at this level.

Two groups of pigs were compared in their ability to take up I-131 in the thyroid. One group had been fed 5 µc I-131 daily for 4-1/2 years, while the other group had received no I-131. Thyroids in both groups took up about 18% of the administered dose. The time at which the I-131 tracer peaked was slightly earlier, and the effective half-life for I-131 retention was somewhat longer in the animals which had previously received I-131. This work is still in progress.

Virulence of columnaris grown at 25 C was greater than when grown at either higher or lower temperatures.

Chloramphenicol was found to inhibit ion (rubidium) accumulation by barley plants, evidently by blocking active transport systems.

Five and seven percent concentrations of effluent water from the KE Reactor were toxic to fish but not three percent.

5. Programming

Preliminary computations have supported HLO's contention that Phoenix fuel action need not be limited to Pu-240 as the primary fertile source. The over-all performance of a Phoenix fuel has been essentially duplicated starting with equal amounts of U-235 and U-238 physically arranged so that the U-238 absorption resonances are unshielded while the U-235 absorptions are heavily shielded. The amount of moderator is adjusted so that a classical U-238 resonance escape probability of approximately 0. 5 is achieved. This provides nearly as many neutron absorptions in U-238 as in U-235, which provides uniform reactivity for an extended period as in "Phoenix" fuels.

In the same vein, it is currently believed that the use of plutonium with some Pu-240 -- but an insufficient amount for Phoenix in its own right -- can use a small amount of U-238 sufficient to hold the initial reactivity down until enough Pu-240 is formed to supply the classical "Phoenix" fuel. For this application with Pu-239, the U-238 and moderator geometry will not be as sensitive as for the application with U-235.

Computations with more elaborate models are now being made under the direction of the HLO Theoretical Physics group to firm up the application of U-238 as described, and to investigate the possibility of using thorium in the same fashion.

TECHNICAL AND OTHER SERVICES

One serious plutonium contaminated puncture wound occurred to a CPD employee. The wound initially contained about 1 μc Pu before flushing and excision (the maximum permissible body burden is . 04 μc). Two separate excisions removed the contaminant at the wound site down to about . 006 μc. Treatment with DTPA was initiated promptly by the industrial physicians. Analysis of bioassay samples for the four-week period following the injury indicated that the employee excreted about . 01 μc Pu. It is not yet possible to provide a reliable estimate of the magnitude of the plutonium deposition because of the effect of the DTPA treatment on excretion rate.

Nine plutonium contamination incidents involving potential inhalation of plutonium for 13 CPD employees at the 234-5 Building were reported during the month. Five plutonium contamination incidents involving nine HLO employees at the 231-Z Building were also reported. All except two of these incidents resulted from ruptured hood gloves or the failure of plastic bags containing contaminated material. Special bioassay sampling was initiated for all employees involved.

Three new cases of plutonium deposition were confirmed by bioassay analyses during June. The total number of plutonium deposition cases that have occurred at Hanford is 291, of which 209 are currently employed.

On two separate occasions difficulties in removing thermocouple stringers at the K reactors resulted in IPD maintenance personnel receiving whole body radiation exposure in excess of control limits. In one case the exposure received was 1.9 r, and on another occasion the exposure received by another maintenance employee was 1.2 r. Contamination of the X-1 level up to 500 r/hour was caused by fractious pieces of removed materials.

Fresh fallout (estimated by the I-133/I-131 ratio to be five days old) was detected in the Pacific Northwest in the middle of June. This fallout was evidenced in samples of produce which were found to have fission product concentrations up to 20 times that noted for early June. The I-131 content of beef thyroids collected in late June were also some 20 times higher. Local milk samples were up to ten times higher in I-131 content.

A formal document was issued in cooperation with FPD personnel which describes the effects of reactor environment on the dimensional distortion of Hanford production fuel elements.

Estimates were made of the precision and relative bias in the determination of the Pu-240 content of fabricated parts by neutron count and by mass spectograph.

In connection with the forthcoming attitude survey, methods of analysis were developed and submitted for programming after approval by other members of the survey task force.

Time effects have been removed from several series of data on mid-column aqueous uranium concentration by the use of the variate difference method. The next step will be the estimation of the spectral density from the trend corrected differences. It is hoped that the stability of the column can be defined in terms of this spectral density.

A special magnetic tape has been prepared to test the lathe control components which have arrived on plant and are being assembled on the experimental Gorton lathe.

[redacted]

The final version of a computer program to index cubic crystals has been turned over to the customer and is functioning routinely. An associated program was written to extrapolate the value of the lattice constant for a cubic crystal.

Authorized funds for 13 active projects amount to $2, 704, 600. The total estimated cost of these projects is $8, 767, 000 of which $1, 308, 000 has been spent through May 31, 1962. Two new projects, CAH-958, Pu Fuels Testing and Evaluation Laboratories, 308 Building and CAH-962, Low Level Radiochemistry Building were authorized this month by the Commission.

During FY 1962, the Laboratory Equipment Pool has successfully assigned equipment valued at $107, 290 in lieu of purchasing new equipment, while operating costs of the Pool aggregated $11, 330, thus providing a net savings to Hanford Laboratories for the year of $95, 960.

SUPPORTING FUNCTIONS

PRTR output was limited to 185 MWD for a plant efficiency of 8. 8%. Because of the low operating efficiency, exposure data normally included in this report did not change significantly during the month and will not be presented.

The reactor was shut down on June 3 for refueling and process tube examination. Seventy-six process tubes, four UO_2 elements, three LX Pu-Al elements and one mixed oxide element were inspected during the outage. One Pu-Al element was found to have lost a band, and was removed to the storage basin. Another Pu-Al was removed when the results of stagnant water tests indicated that it probably had defective cladding. A UO_2 element showed severe wear on the spiral bundle wrapping wire and was removed to storage.

A piece of 304 stainless steel found in the inlet jumper of a process tube led to radiographic examination of the flow straightening vane upstream of the bulk flow venturi, which confirmed damage to the straightening vane. At month end the reactor was discharged of all fuel elements in preparation to draining the primary system to further investigate and repair the vanes and evaluate possible corrosion problems throughout the system.

D_2O and helium losses were 3694 lbs and 66, 600 scf, respectively. High D_2O losses are attributed to extensive charge-discharge and tube inspection

 [redacted]

work which required the system to be open. In addition, a D_2O spill from which 7440 lbs of D_2O were recovered resulted in an additional loss estimated at 150 lbs from stack gas condensate sampling. The spill came about during preparations to remove a process tube while another process tube was partially disconnected, a combination pressure and syphoning action caused D_2O to drain from the primary system into the upper and lower access spaces. The water loss was stopped after about 5 or 6 minutes.

A weld failure was found on the 26-inch expansion spool bellows providing containment between the vessel and the export steam line. About a 120° of seam weld on one of the folds had failed. Repairs were made and the rest of the bellows including the one on the relief line was thoroughly inspected and leak tested. No leaks or other defects were found.

Total productive time for the Technical Shops was 22,499 hours. This includes 14,677 manhours performed in the Technical Shops, 7,332 hours assigned to Minor Construction, 375 hours assigned to off-site vendors, and 115 hours to other project shops. Total shop backlog is 19,474 hours, of which 70% is required in the current month with the remainder distributed over a three-month period. Overtime hours worked during the month was 8.9% (1702.1) of the total available hours.

PRTR operating costs in June included heavy water charges of $52,080 to cover losses of $51,320 and scrap of $760. Total FY 1962 heavy water charges to operating costs were $472,240 representing (1) losses, $444,410 and (2) scrap, $27,830.

Six Ph.D. applicants visited HAPO for employment interviews. Twelve offers were extended; two acceptances and three rejections were received. Current open offers total 14. Four program offers and 13 direct placement offers were extended to BS/MS applicants. Program results included one acceptance and eight rejections with three open offers remaining. Direct placement offer response this month included five acceptances, 13 rejections and five remaining under consideration.

One Technical Graduate was placed on permanent assignment, 34 new members were added to the rolls, nine transferred to other Company sites in accordance with the planned program, and one terminated. Current program members total 56.

[redacted]

A Symposium on the Biology of the Transuranic Elements was held on May 28 - 30. About 70 visiting scientists, including ten from foreign countries, met with an equal number of Laboratories scientists.

Paul F. Gast
for Manager
Hanford Laboratories

HM Parker:PFG:mlk

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REACTOR AND FUELS RESEARCH AND DEVELOPMENT OPERATION

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 2000 PROGRAM

1. METALLURGY PROGRAM

Corrosion Studies

Corrosion of Beryllium Brazed and Pressure Bonded End Closures. Additional corrosion results on beryllium brazed end closures and copper bonded end closures have been obtained. The corrosion rate for both the welded and unwelded beryllium brazed closures exposed to 400 C steam was about 2.5 mils/year. Except for copper contaminated weld areas, the copper bonded and welded end closures continue to show good corrosion resistance following 68 days in 400 C steam. Small portions of the copper contaminated welded areas showed corrosion penetrations of approximately one mil. Small sections of pressure bonded titanium end closures are currently being tested in both 400 C steam and 360 C water. A penetration of the bond layer of about 2.5 mils occurred during a 90-hour exposure to 400 C steam.

Effect of Tin and Oxygen Content on Zircaloy-2 Corrosion. The effect of tin in the range of 0.1 w/o to 1.5 w/o and oxygen in the range of 900 ppm to 1300 ppm on Zircaloy-2 corrosion in both 400 C steam and 360 C water is being determined. Following 30 days of exposure in 360 C water, no effect of tin or oxygen content was noted. All samples had an approximate corrosion rate of 18 mg/dm^2. The samples exposed to 400 C steam for 42 days show considerable scatter but an average of all of the results tends to demonstrate a lower corrosion rate with increasing tin content and low oxygen concentration. The average corrosion weight gains for the high oxygen samples ranged from approximately 55 mg/dm^2 for 1.5 w/o Sn to 120 mg/dm^2 for 0.1 w/o Sn as compared to 40 mg/dm^2 for 1.5 w/o Sn to 60 mg/dm^2 for 0.1 w/o Sn for the low oxygen samples.

Metallic Fuel Development

Fuel Irradiations. The prototype NPR tube-in-tube fuel element successfully completed its irradiation to goal exposure in the ETR M3 Loop. The average exposure of the fuel element at the time of discharge is estimated to be 1300 MWD/T and the maximum exposure is approximately 2000 MWD/T. The fuel element is to be disassembled, visually examined, and measured at the ETR. It will then be returned to Hanford for radiometallurgical examination.

Five N-inner fuel tubes (NIE's), which were successfully irradiated to 3000 MWD/T under N-Reactor operating conditions, are being examined in the radiometallurgy facility. The general superficial appearance of the elements is good with no indication of unusual corrosion behavior or crud deposition. The inner bore of these elements is starting to distort, presumably due to inner clad buckling resulting from fuel swelling. The distortion is visibly perceptible but has not yet been measured. Outside diameters have increased an average of 0.007-inch as a consequence of fuel swelling. This diameter increase amounts to approximately 0.6 percent cladding strain and accounts for about three-fourths of the measured 1.7 percent fuel swelling.

The variable braze thickness irradiation test, GEH-4-68, 69, and 70, has been received from the MTR after six cycles of irradiation. Radiometallurgical examination has begun. All elements appear in good condition and there is no external evidence of any cracks, stressed areas or other incipient failures.

Cladding Deformation Studies. Four cladding studies capsules, recently discharged from DR Reactor at an exposure of 1500 MWD/T, were opened in the radiometallurgy facility. The visual examination made on the 12 fuel rod specimens from these capsules showed no evidence of localized strain either in the cladding in the uniform cladding thickness regions or associated with the intentional striations in the cladding. The irradiation conditions for the capsules were chosen so that the cladding should have experienced a 1.5 to 2.0 percent cladding strain. Measurements of diameter and length of all the rods and metallography of selected rods is planned. Fabrication of components for a second irradiation test of fuel rods with non-uniform thickness of Zircaloy-2 cladding continued. Machining of the fuel samples is completed and welding of end caps to the cladding has started.

N-Fuel Support Development. The response of N-Reactor fuel element assemblies to vibrations was investigated to explore the condition under which the fuel supports would experience excessive cyclic fatigue loading or in which rubbing of components could produce a fretting corrosion situation. The results of shaker table tests show that the measured natural frequency of fuel assemblies vibrating in the simpler modes agrees with calculated frequencies based on measured support spring constants. The effect of a water environment is to lower the natural frequencies to about 93 percent of their air environment values.

N-Fuel Support Testing. A program of scratch testing the N outer fuel element support was renewed after tests in the 314 Building charging machine mockup resulted in scratching the process tube. Testing in 306 Building was done under the following conditions:

Test I

a.	Process tube length	5 feet
b.	Lubricant	Process water (306 Bldg.)
c.	Accumulated distance	1002 feet

Test II

a.	Process tube length	5 feet
b.	Lubricant	10 v/o A-60 soluble oil in water
c.	Accumulated distance	1002 feet.

The fuel element was cycled back and forth over a distance of three feet for 334 cycles to accumulate the total distance. Periodic inspections of the supports and process tube were made, and only normal wearing of the support surface was observed. Borescope examination of the process tube revealed only light deposits of steel on the tube surface. The process tube was submerged during both tests, and a charging speed of 18 feet per minute was used during cycling. Previous wear tests on Weber's apparatus indicates a wear distance of 1540 feet before failure. As a result of a second charge failure (scratching) in 314 Building, a third test, using the elements from the 314 test, was run in the five-foot tube in 306 Building. The elements were rotated so that the four supports in the upper position during 314 testing were in the lower position in the 306 test. The entire charge (20 elements) was pushed through the tube, one element at a time, without damage to the process tube.

Fluted Fuel Development. Arrangements for irradiating an N-inner size fluted Zircaloy-2 clad fuel tube in an ETR pressurized loop are nearly complete. Two 12-inch long fuel elements are on hand, the fuel baskets and associated hardware are nearly complete, and the test proposal document is in final draft form. The maximum core temperature of the fuel is calculated to be 527 C and goal exposure is to be 3000 MWD/T.

Fluted Fuel Elements. Cylindrical fuel elements with fluted exterior surfaces have been proposed as a fuel element design in which the fuel is capable of undergoing large volume expansions without failure of the Zircaloy-2 cladding. Mechanical analyses have been made assuming that the fuel behaves as a viscous fluid and that the cladding behaves in an elastic manner. The results of these analyses have been used to estimate the effects of plastic yielding on the circumferential strains within the cladding. Estimates of the ratio of bending strain to mean extension of the cladding indicate that the mean tensile strain would be less than one-tenth the bending strain.

Projection Welded Brazed Closure. The projection welded brazed closure is being developed as an alternate closure. Recent work shows that it is possible to make a good ID and OD weld, and then to burn off the ID cap projection above the weld, making it appear that the ID weld had had too much current. This condition is caused when the central portion of the cap butts against the large relatively cold uranium and stops forging the weld and projection. A re-design of the cap is expected to solve this problem. Problems have been encountered with the center plug arcing across the weld. Krylon coated stainless steel shows the most promise to date of preventing arcing. Consideration is being given to a laminated plug. A number of closures have been made during the month with acceptable ID and OD welds. Several closures have been made which showed acceptable bonds in conjunction with good welds. Some of these had arc burns on the ID or surface mars on the OD which will have to be cleaned before the process is ready for production testing.

Resistance Brazed Closure. After experimenting with the principle of forming a uranium-Zircaloy bond at the closure by short-time, relatively low-temperature pressing of the closure end of the fuel element in various types of die, it appears that the earlier concept of using an intermediate alloy to facilitate bonding leads to best results.

By heating Zircaloy rings in vacuo, while in contact with metal plating and/or chips of various alloys, the melting and freezing characteristics of the alloys were studied. Electro-deposited copper plating tends to blister, while displacement-plated copper does not. Displacement-plated copper, overlaid with a thin electro-deposited nickel plate forms with the zirconium a ternary alloy whose melting point is in the range of 800-850 C. Attempts to make a bulk batch of alloy of this eutectic composition have not been entirely successful, although one (80 Zry-10 Ni-10 Cu) having an apparent melting point of about 930 C, approaches the desired composition. This temperature is about 30 C lower than the melting point of the Be-Zr alloy used in the current brazed closure process. However, it is not known whether alloys of the Ni-Cu-Zr composition will have adequate corrosion resistance and/or toughness.

Copper Brazed Closure. A method has been developed using pure copper as the only braze material necessary to obtain a good brazed closure. During the heating cycle of the brazing process, the copper alloys with the zirconium end cap to form a Zr-Cu eutectic, which then acts as the low melting brazing alloy. The process has been developed to the point that brazed closures can be made at as low a temperature as 1000 C. The addition of a small amount of tin to the copper has made possible brazes at temperatures of 980 to 990 C.

Two fuel elements were braze closed, one with the copper braze, the other with the copper-tin braze. After brazing, the elements were electron beam welded over the braze and autoclaved for 72 hours in 400 C steam to test the corrosion resistance of the copper and the copper-tin contaminated welds. After autoclaving the welds had a slightly different color than the uncontaminated Zircaloy but showed no severe corrosion. The copper closure appeared to have slightly better corrosion resistance than the copper-tin closure. The welds are being examined for possible uranium contamination and the elements will be examined metallographically. Longer term auto-claving tests are scheduled.

Hot Headed Closure Studies. The objective of Phase I is to determine the conditions for projection welding a cap to a "hot-headed" fuel element and obtain a minimum void area between the cap and the fuel element.

Projection welds made during the previous month using a modified design were evaluated. The results indicated that while the projection design seems to be satisfactory the recesses adjacent to the projection were too large. When the recesses adjacent to the projection are too large, small cavities are formed at the peripheries of the recesses when the weld is made. "Set down" of the caps was sufficient to provide good interface contact.

The objective of Phase II is to investigate and develop methods to obtain a continuous bond between the cap and the fuel element in an area circumscribed by projection welds.

During this past month effort was directed to vacuum plating techniques as a method of preplacing bonding materials. Plating fixtures were designed and constructed to prevent the deposition of bonding material on the projection and recesses of the caps during the vacuum plating operation. Although vacuum plating of beryllium onto the Zircaloy caps has been accomplished in the electron beam welding chamber, attempts to vacuum plate copper to the Zircaloy caps were not successful. This result was attributed to the short focal length of the present type electron guns and the high power requirements. A commercial electro-magnetically focused electron beam gun with a longer work to gun distance was purchased. It will be adapted for use on the existing equipment.

Seven Zircaloy caps were vacuum plated with a thin layer (approximately nine micro-inches) of copper in a vacuum plater designed to produce thin metal coats on optical components. These caps were projection welded to flat Zircaloy disks of the same geometry as the caps. The post heat treatment of these assemblies after projection welding was varied while they were still in the welding machine. Metallographic examination of cross sections of these joints at 100x and 400x revealed that two of these assemblies were bonded across the entire interface between the projections with several small discontinuities. The other assemblies showed varying amounts of bonding.

Nitrogen Shielded Copper Welding. Increased penetration is an advantage of welding copper using nitrogen as a shielding gas rather than argon or helium. However, the welds continue to contain some porosity. The welds appear to show some sensitivity to thermal shock. This may be caused by very fine porosity. These disadvantages nullify the benefits of increased penetration if the weld were to be used as a billet closure. No further effort is planned on this project.

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

Primary extrusion of a 19-inch long NON high Fe, Al ingot whose beta treatment was reported last month was completed by Coextruded Product Engineering. Sections of this primary extrusion were beta heat treated and were annealed at 600 C for periods of 24 hours, 6 hours, and $1\frac{1}{2}$ hours prior to coextrusion. These treatments are being employed for possible reduction in stiffness of this alloy. The second NON Fe-Al ingot is being held, pending the results of these coextrusions.

Dimensional Measurement of N-Fuel Elements. A computer program using paper tape input data is used to interpret measurement data from tubular fuel elements. The program provides mean diameters, warp, and ovality based on 72 circumferential measurements of radial displacements and thickness. The program was temporarily altered to take every second and fourth data point to see the effect of making fewer measurements. The results of these analyses showed that no significant changes in the results occurred due to the reduction of the number of data points.

Fuel Element Swelling Model. Test data from fuel element irradiations should provide a means of evaluating parameters in a model for fuel swelling. To account for the temperature and burnup distributions within different fuel element geometries, the model should consider the volume expansions of the fuel element within localized temperature regions. An empirical model suitable for statistical evaluation of fuel element data has been formed for averaging the effects of temperature and irradiation conditions. The initial investigation using a model with a power function dependence on temperature and burnup did not account for the observed data. A new model including a factor with a cut-off temperature below which swelling does not occur is being programmed.

Cerium-Iron Braze Alloy. A cerium - 7.4 w/o iron alloy was prepared by arc button melting. This alloy is a eutectic composition, reported to melt at 595 C. The material was vacuum brazed at 625-640 C to determine whether it would wet Zr-2 or uranium and the extent of reaction. The molten alloy does not flow easily but does wet both uranium and Zr-2. No diffusion zone was observed between Zr-2 and the braze with 500x metallographic examination. Diffusion and formation of a compound layer occurred on the uranium side. Fracture of the braze-uranium interface leaves a reaction layer on the uranium resistant to air oxidation. Attempts are being made to form a thin foil of the material for additional brazing tests.

Cerium - Modified Zr-2. Small additions of cerium have been added to high oxygen zirconium to determine if there might be a scavenging effect (May 1962 Progress Report). Five buttons have been melted containing 0.1, 0.5, 0.75, 1.0 and 2.0 w/o cerium.

The cerium additions have little or no effect on the room temperature hardness of the as-cast buttons.

Hot rolling the buttons from a 730 C salt bath to 0.100-inch showed a tendency toward edge cracking in the higher cerium addition. Chemical analyses for cerium, oxygen and nitrogen will be obtained. High pressure steam corrosion tests will be conducted on these alloys.

Facilities and Equipment. A viewing autoclave facility has been activated in the 306 Building. This facility permits the exposure of one inch diameter samples to water temperatures up to 315 C. Direct visual observation or photography of the specimens during

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their exposure is possible. Included also in the facility are two standard 3500 psig-500 C autoclave, 300 ml and 1400 ml vessels, both of which are capable of rapid quenching. The facility is to be used in studies of the failure behavior of coextruded metallic fuels.

2. REACTOR PROGRAM

Gas Atmosphere Studies

Electrical Resistance of ZrO_2 Films. The temperature dependence of the electrical resistance of ZrO_2 films on crystal bar zirconium and Zr-2 was measured using a graphite contact. With the samples in an oxidizing atmosphere (H_2O) a plot of log R versus 1/T gave a straight line for both metals yielding a ΔE of about 11 K cal for Zr-2 and 16 K cal for crystal bar zirconium. The absolute values of the resistance of the oxide film on the crystal bar were higher (10-20 megohms at 450 C) than the Zircaloy-2 (0.5-0.8 megohms at 450 C). After prolonged heating in a vacuum the electrical resistance of the films on both metals dropped to 100-500 ohms and showed almost no variation over a temperature range of 150 to 550 C.

These results correspond to the expected electrical properties of a semiconductor and are explicable by the previously proposed vacancy model. In the oxidizing media, the electrical conductivity is imparted by thermal formation of free electrons and electron holes. The "intrinsic" conduction, and the measured ΔE values relate to the energy required to form pairs of electrons and holes. The absolute value of the resistance of oxide formed on crystal bar zirconium is higher because the impurities in Zr-2 (Sn, Fe, Cr, and Ni) create defects in the crystal structure which make electrons available to the conduction band.

After heating in a vacuum the creation of oxygen anion vacancies also makes electrons available to the conduction band accounting for the observed drop in resistance. The energy required to release these electrons from the vacancies to the conduction band is of the order of 0.5 K cal which means the vacancy associated electrons are in the conduction band even at room temperature. This effect overrides the intrinsic conduction effect, so little variation of R with temperature is seen when a high concentration of oxygen vacancies is present.

Distribution of Hydride in Zircaloy-2. A Zr-2 sample was examined for hydride distribution by metallography after 300 days exposure at 425 C to a He 2% H_2, 1% CO 0.07% H_2O gas mixture. The hydrogen level of the sample was about 2000 ppm by hot extraction analysis. The hydride was found to be uniformly distributed with no hydride surface layer. Since the solid solubility limit at 425 C is about 350 ppm, this result indicates an unexplained mechanism for diffusing hydrogen through saturated alpha zirconium. From the published Zr-H phase diagram a hydride case might have been expected.

In another experiment samples of NPR process tube were charged uniformly with hydrogen to 400 ppm and then corroded in 400 C, 1500 psi steam for 230 days. The corrosion hydrogen pickup fraction was a normal 30 percent and the additional hydrogen was found to be uniformly distributed throughout the sample rather than as a case.

Graphite Burnout Monitoring. Burnout rates of small monitoring samples (0.43-inch diameter x 4-inch long) from 2577 DR were recently measured. Although the gas composition of the reactor atmosphere has not yet been examined, the burnout rates indicate a significant reduction in the amounts of oxidants. The highest rate for the period from March 31, 1962, to June 8, 1962, was 1.9 percent per 1000 operating days at a position 150 inches from the front face of the graphite stack, whereas the highest rate from January 7, 1962, to March 31, 1962, was 107 percent per 1000 operating days at a distance of 100 inches.

In the latter test there appears a slight peak between 75 and 100 inches; the peak is approximately 0.7 percent per 1000 operating days and is believed to result from the graphite-oxygen reaction.

Burnout measurements in Channels 1960 C and 2780 C from March 16, 1962, to June 3, 1962, showed substantially the same rates as the previous period from August 31, 1961, to March 16, 1962; burnout rates from both test periods showed similar results with regard to peak height and location. A tabulation of the rates is shown in the following table.

BURNOUT RATES FROM CHANNELS 1960 AND 2780 AT C REACTOR DURING THE PERIOD FROM MARCH 16, 1962 TO JUNE 3, 1962

Distance from front of graphite stack to burnout monitor, in.	Channel 1960 Burnout rate of monitor, % per 1000 operating days	Channel 2780 Burnout rate of monitor, % per 1000 operating days
3	0.14	0.31
7	0.16	0.28
11	0.17	0.26
15	0.32	0.32
19	0.25	0.38
63	0.87	0.83
67	dropped	1.38
71	1.67	1.77
75	6.48	3.61
79	17.8	5.01
99	24.07	2.56
103	23.85	1.49
107	22.26	1.19
111	25.11	1.10
115	17.78	1.21
137	0.98	0.81
141	1.32	0.92
145	0.45	1.34
149	0.32	0.70
153	0.40	0.71
171	0.45	0.68
175	0.44	1.28
179	0.38	0.52
183	0.16	0.56
187	0.79	0.58

A decision was made to retube 1889 C, thus removing it from monitoring service. The retubing was postponed, however, when borescope examination revealed apparently greater oxidation in the lower portions of the stack. This evidence is supported by the burnout rates shown in the above table. The rates in the area between 99 and 115 inches are 18 to 25 percent per 1000 operating days for channel 1960, whereas the rates in the same region of channel 2780 ranged from 1 to 3 percent per 1000 operating days. Burnout monitors are currently in all three channels.

Corrosion and Coolant Systems Development

Fuel Element Rupture Tests. Several rupture tests of NPR inner tube elements were made in TF-9 during the month to determine the effects of the addition of iron and silicon to the uranium core of coextruded U-Zr-2 fuel elements. The first two elements tested were defected with a pinhole near the Zr-5 w/o Be braze. After three hours at 300 C, 1800 psi and 20 fps, the unalloyed element was badly ruptured with the end cap held on only by the internal tube and the Zr-2 cladding bulged out over the entire end. The alloyed element had only a 1/2 inch diameter hole in the cladding but a large portion of the U core at the end cap was gone. The weight losses for the elements were almost the same at 308 grams. The other two elements were defected with a pinhole in the side. After one hour at 300 C and a slow cooldown, 1800 psi and 20 fps, the alloyed element was ruptured much worse than the regular element. The alloyed piece had a large torn and bulged area measuring 3-inch x 1½ inch and a weight loss of 76 grams. The regular element had a bulge 7/8-inch x 1/2-inch and a weight loss of 10 grams.

Corrosion of Zr-2 Under Heat Transfer. The third heat transfer test in TF-3 was terminated. Zr-2 cladding was exposed to a heat flux of 280,000 Btu/hr-ft^2 in high-purity water at 580 F adjusted to a pH of 10.0 with LiOH. The surface temperature was calculated to be 637 F or just below the point at which nucleate boiling occurs. The sample was removed after 1584 hours. A very thin (<0.1 mil) crud film was observed on the Zr-2 in the region of heat transfer. In a region where the cladding had been reduced in diameter larger amounts of crud were present. Also present was a white deposit of unknown composition. This white deposit exhibited cracks which appear to extend into the metal. The region is subject to large stresses due to the greater expansion of the internal heater. Because of the reduced diameter the heat flux and surface temperatures were higher in this region (being estimated as 360,000 Btu/hr-ft^2 and 660 F), and surface boiling probably occurred. Complete examination of the specimen is scheduled to determine film thicknesses and composition and the presence of cracking or other serious corrosion.

Caustic Cracking. Stressed coupons of 304 S/S, A212 C/S, and Zr-2 were examined after 1584 hours exposure in 580 F, 0.5 w/o LiOH solution, in a capsule charged into TF-3. Fresh solution was placed in the capsule at each inspection period. No serious corrosion, pitting, or cracking was observed in the stressed or crevice areas. Testing will be resumed at higher temperatures.

Ammonium Hydroxide for pH Control. A test has been completed to evaluate the inhibition effectiveness of NH_4OH as an alternate to LiOH. Samples were exposed in TF-8 for 2100 hours in deionized water adjusted to pH 10 with NH_4OH. A rate of 0.044 mil/year was obtained on 304 stainless steel, which is comparable to rates obtained with LiOH. A corrosion rate of 0.4 mil/year was obtained on A212 carbon steel, which is higher than rates obtained with LiOH. However, the higher rate could be due to a cleaning pretreatment with HCl, which is normally not used. The carbon-steel portion of the test will be repeated.

Corrosion of Carbon Steel in Raw Water. The test in TF-13 to determine the corrosion rate of A212 carbon steel in 210 F raw Columbia River water was completed. The water velocity past the sample was about 5 ft/sec. The surfaces were covered with a 10-30-mil thick oxide and numerous tubercules, with broad, shallow pits beneath each tubercule. The corrosion rate decreased from 6.5 mils/year during the first 1500 hours of exposure to 0.5 mil/year during the last 3500 hours. The total exposure was 7500 hours.

Corrosion by Decontaminants. Photomicrographs were obtained of Stellite alloys exposed to alkaline permanganate for 12 hours. Haynes alloy 25 was not attacked, whereas pits from 2 to $4\frac{1}{2}$ mils deep were found in coupons of Stellite-6. Corrosion of Hastelloy-X in alkaline permanganate and ammonium citrate solutions was found to be negligible, i.e., 0.003 mil or less in 4 hours.

On-Line Analysis of Recirculating Water Coolants. On-line operation of the automatic, wet chemical analyzer was studied. The silica and chloride procedures developed during laboratory testing were each evaluated during 5-day, on-line tests. During these tests the analyzer ran unattended to demonstrate its usefulness for continuous operation. The results of these tests confirm the accuracy and sensitivity data reported for the laboratory tests and demonstrated a high degree of reproducibility for these analyses. Similar tests of the phosphate and hydrazine procedures will be performed in the near future.

Performance of the thallium-based dissolved-oxygen analyzer is still satisfactory. Results indicated by the analyzer are in good agreement with those calculated from the analyzer inlet and outlet conductivity readings. The time required for this instrument to respond to a change in the dissolved oxygen concentration is still unknown.

A second type of analyzer was obtained for dissolved-oxygen analysis. This instrument consists of a galvanic cell with silver and zinc electrodes. Oxidation of the zinc electrode releases zinc ions and reduction of the dissolved oxygen produces hydroxide ions in the water as it passes through the cell. The increase in the cell current is thus directly proportional to the oxygen concentration in the water. An electrolytic cell for oxygen generation is an integral part of this analyzer and is used for calibration of the measuring cell. The manufacturer claims that the response time of the analyzer for sensing changes in the oxygen concentration is about 2-5 seconds. To date we have not been able to make this analyzer work properly. The oxygen generated in the calibration cell is not being detected properly by the measuring cell.

K-Reactor Nozzle Snap Rings. Modified K-Reactor nozzle snap rings fabricated from four different alloys have been tested for corrosion evaluation. Those of 347 stainless steel, 302 stainless steel, and 15-7 Mo stainless steel were tested in the stressed and unstressed condition for 4½ hours in boiling 42% $MgCl_2$. The 347 and 302 stainless steels in the stressed condition were cracked, but the 15-7 Mo S/S sample was not cracked. None of the unstressed samples were cracked. The stressed 15-7 Mo S/S samples were exposed an additional ten days without cracking occurring. An Inconel-X snap ring which is currently being exposed to boiling 42% $MgCl_2$ has not shown any cracking after four days of exposure.

Structural Materials Development

Creep Rupture of Zircaloy-2 Pressure Tubes. The effect of the various metallurgical histories on the creep rupture properties of Zircaloy-2 pressure tubes for N-Reactor is being determined. In tests performed at 300 C, two anomalous results have been obtained. After 250 hours in test, the secondary creep rate and total creep strain are the same for tubes with 15, 18 and 35 percent cold work. Tests on sections of 30 percent cold worked KER tubes have yielded higher secondary rates and in one case the total creep strain was four times greater than that for a similar stress at the other cold worked levels. The source of these anomalies may be the high stress and strain rates used in these tests. Additional tests are under way.

Initial startup tests have been completed on the prototype unit for burst tests on irradiated pressure tubes. The unit was installed in the 327 Building basin and tests are continuing.

Graphite Damage Studies

Graphite Moderator Distortion. Least squares slopes have been calculated for the rate of change in elevation with cumulative exposure at the top center of the graphite stack for H, DR, C, KE and KW Reactors. The results are shown in the following table.

CHANGE IN STACK HEIGHT WITH EXPOSURE

Reactor	Moderator Stack Contraction ($in/10^6$ Mwd)*	Time Interval
H	0.95 ± 0.06	1957 to present
DR	0.83 ± 0.03	1952 to present
C	0.79 ± 0.04	1954 to present
KE & KW	0.64 ± 0.07	June 1956 to June 1960
KE & KW	0.82 ± 0.08	June 1960 to present

*99 percent confidence limits.

NPR Graphite Irradiations. Capsule H-5-1, one of a series of long term irradiations of NPR graphite in the GETR, was removed from the reactor May 20, after being irradiated for four reactor cycles or a total of 90.5 effective days at full power. All thermocouples operated satisfactorily during the entire irradiation. The capsule was found to be in excellent condition during disassembly in the hot cell. All 24 samples and 65 flux monitors were recovered and have been returned to HAPO. Sample and flux monitor measurements are currently in progress.

The initial second-generation capsule, H-4-2, was installed in the GETR May 21. The capsule is presently operating satisfactorily at design temperatures with all thermocouples in service.

The third first-generation capsule, H-6-1, is operating satisfactorily in the GETR for the third reactor cycle. Sample temperatures were approximately 50 C higher than the normal 425-800 C range for the first week of the present reactor cycle but returned to the normal range when the control rod positions were changed.

Thermal Hydraulic Studies

Heat Transfer Characteristics of NPR Fuel Elements. The studies to determine boiling burnout conditions for the NPR tube-in-tube fuel elements were continued. Twenty-two boiling burnout points were obtained in the laboratory with an electrically heated model of the center portion of the fuel element. The test section for these runs consisted of a 12-foot long tube, 0.44 inch ID, with flow through the inside. The tube was heated by electrical resistance heating and boiling burnout conditions were detected by noting temperature excursions as measured by thermocouples attached to the outside wall of the test section.

Six of the experiments were conducted with a system pressure of 1500 psig. These experiments were conducted to verify the occurrence of upstream burnout conditions as discussed in HW-73902. This condition of burnout originating upstream from the outlet end of the test section had been observed with both vertical and horizontal positioning of a 12-foot long test section at mass flow rates of 5, 6, and 7 x 10^6 lb/hr-sq ft with inlet coolant conditions near saturation. To make sure that this condition was not peculiar to the particular test section, the burnout experiments were repeated but with a different test section. The data checked very closely with previous results.

Sixteen of the boiling burnout experiments were conducted with a system pressure of 1000 psig and mass flow rates ranging from 2 to 7 x 10^6 lb/hr-ft^2. These data were obtained to allow comparison of burnout heat flux for different system pressures and also to determine if upstream burnout occurred with 1000 psig pressure.

The experimental data have not been examined in detail yet, but the following general conclusions can be drawn. Upstream burnout did occur at 1000 psig system pressure at mass velocities of 5, 6, and 7 x 10^6 lb/hr-ft^2. The comparison between the 1000 psig and the 1500 psig pressure data shows that in the high mass velocities (5 x 10^6 to 7 x 10^6) the burnout heat flux for any given outlet quality is quite comparable. At lower mass velocities the 1000 psig system pressure will allow higher outlet qualities for any given burnout heat flux.

Plans were made to investigate the effect of self-supports on NPR fuel element burnout heat flux. Previous studies with test sections simulating "old reactor" fuel elements with a crushed self-support have shown that steam bubbles issuing from beneath a metal strip results in a non-wetted area and a lowered burnout heat flux. Clearance between the metal strip and the heater rod surface in these tests was appreciably less than that between an NPR self-support and the fuel element surface, and therefore the NPR element may have no problems. However, it is believed that the matter should be investigated.

The first experiments will be conducted in a glass tube at low pressures so that flow patterns may be observed. If these experiments indicate that the self-supports might affect the burnout heat flux, further experiments at high pressures will be conducted.

Flow Tests for K-Reactor Zircaloy Tube Assemblies. Flow tests have been conducted in the Hydraulics Laboratory to determine the hydraulic characteristics from front header to rear header of the fittings, fuel, and process tube assembly which is proposed for use in retubing K-Reactor with Zircaloy tubes. The equipment assembly consisted of a standard K venturi (0.419 inch throat), a standard front flexible pigtail, a new front nozzle (SK-1-23101), a ribless Zircaloy-2 process tube (1.724-inch ID), a revised rear nozzle, and a standard K-Reactor outlet pigtail. The rear nozzle consists of a standard K rear nozzle with the barrel reamed to a larger ID and a perforated plate inserted in the nozzle water outlet region to prevent self-supported fuel shoes from lodging in the recessed area.

The new inlet nozzle design results in a pressure loss from the front header to the tube inlet which is essentially identical to that for the present standard K front nozzle assembly. The fuel charge results in a pressure drop from the tube inlet to end of active section of 238 psi at a flow rate of 60 gpm at 20 C. The pressure drop from the end of the active charge to the rear header will vary depending upon the type of downstream support charge; the minimum will result from using self-supported perfs in the nozzle and thin-wall perfs against the fuel charge.

Hydraulic Tests. Flow tests were conducted to determine the effect on pressure drop of reducing the ID of the flare fitting adapters which might occur through excessive tightening of the front hydraulic connectors to a C-Reactor process tube. The tests were made with necked down front header Parker fittings and with necked down pigtail-to-nozzle adapter fittings. By excessive tightening of a pigtail nut, the ID of the stainless steel front header Parker fitting was reduced from 0.470 to 0.460 inch. This resulted in an increase of 3 psi in the 200 psi pressure drop from front header to venturi throat. Excessive tightening of the pigtail nut caused an aluminum pigtail-to-nozzle adapter to be reduced from 0.468 to 0.390 inch ID. This also resulted in a 3 psi increase from front header to venturi throat at a ΔP of 200 psi. These results show that necking down of the magnitude found in these tests will affect the pressure drop only a small amount.

Flow tests were conducted with four different combinations of process tube outlet fittings which may be used in a modification program for BDF Reactors. From the results of the tests normal operating flow rates and relative pressure drops with and without vaporization could be calculated for process tubes equipped with the various combinations of fittings. Using typical reactor conditions of fuel loading and front and rear header pressures, the normal operating flow rates were determined (1) by changing the header fitting and the nozzle outlet fitting size to 0.550 while retaining the 5/8-inch OD pigtail, and (2) by using a new 3/4-inch OD pigtail with 0.610 header Parker fitting. In (1) a flow rate increased about 5 percent and in (2) about 8 percent. Since the new assemblies would probably not operate in critical flow conditions at normal rear header pressures, the rear header pressure would affect the operating flow rate, whereas for the present standard assembly, rear header pressure does not affect normal operating flow rate. Thus, the actual percent increase in operating flow rate must be adjusted according to the rear header pressure for the individual tube. The results of these tests are presented in HW-63756-4.

B. WEAPONS - 3000 PROGRAM

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

C. REACTOR DEVELOPMENT - 4000 PROGRAM

1. PLUTONIUM RECYCLE PROGRAM

Thermal Hydraulic Studies

Fuel Element Design. An equation was developed for determining the heat transfer flux limits for a 7-rod cluster $MgO-PuO_2$ fuel element. The calculations showed that allowable tube powers for these elements, based on a heat flux limit, would be about two-thirds of the powers allowed with the present 19-rod cluster elements. This lower power limit results primarily from the decrease in heat transfer surface area.

Component Testing and Equipment Development

Mechanical Shim Rod. Design of the second generation shim rod continued. Principal effort was directed to a water cooling system for the drive and position indicating components.

Specifications for a servo-type rod position indicating system were formulated, and a purchase requisition has been prepared.

EDEL-1 Modification. Preliminary design criteria for the long term modifications to EDEL-1 have been established. This work was interrupted to perform a code review of the loop and design work to ready the loop for studies in support of PRTR Zircaloy wear corrosion problems. Renovation work on the loop will include: (1) overhaul of pumps and valves, (2) modification of the loop pressurization system, (3) simplification of the loop controls and safety circuits, and (4) provision for a steam condensate supply to the loop demineralizers.

Shroud Tube Replacement Mockup. Work on the mockup pit was resumed for a week when the construction crafts returned to work. It is currently stopped again by a strike of construction iron-workers.

Inlet Bellows to Process Tube Gas Seal. Leak rate tests were performed on the two new seal designs. Leakage rates were measured during heatup, cooloff, and at steady state temperatures of the seal assembly during several temperature cycles. Conditions of the tests were as follows:

Helium pressure across seal	2" Hg
Maximum tube temperature	500 F
Maximum ambient temperature surrounding seal assembly	460 F.

Leakage rates for the various seal assemblies during the various cycles were as follows:

Thermal Cycle	Leak Rate Liters/Hour @ 2" Hg 1	2	3	4	5	Bolt Torque
Existing Copper "O" Ring	0.4	11.8	21.2	111	132	30 ft lb
Zirconium "O" Ring	200	260				30 ft lb
*B-F Ferrule (Zirconium)	3.2	1.5	1.4			15 ft lb

*Ballard-Foley

All test flanges were made up with the bolt torque indicated and not retightened between cycles. Additional seals will be tested.

Outlet Nozzle to Top Shield Gas Seal. Design work continued on an improved nozzle hold-down design to increase the gas seal gasket pressure and to improve alignment between the seal faces. A purchase requisition was issued to procure a highly compressible metallic gasket for this gas seal. An invention report (HWIR-1523) has been submitted for this gasket design.

Hazards Analysis

Reactor Safeguards Reviews. The following information was presented at the seventh meeting of the General Electric Company Technological Hazards Council:

(a) IPD audit of operation of the PRTR and the interim response by TRAO to the audit.

(b) The final safeguards analysis for the PRTR Fuel Element Rupture Testing Facility Analysis.

(c) Results of the PRTR pressure tube monitoring program.

Safeguards Analysis. A study of reactivity effects of coolant voids in the PRTR with a mixed oxide loading was completed. Three cases were examined corresponding to D_2O coolant containing 0.25, 6, and 10 percent H_2O. Results of these calculations are summarized as follows for the cold-clean reactor:

Complete loss of coolant containing 0.25% H_2O: $-5 \times 10^{-3} \frac{\Delta k}{k}$
Complete loss of coolant containing 6.0% H_2O: 0
Complete loss of coolant containing 10.0% H_2O: $+4 \times 10^{-3} \frac{\Delta k}{k}$

Calculations with hot primary and moderator systems gave essentially the same results within $1 \times 10^{-3} \frac{\Delta k}{k}$. It should be pointed out that the "four-factor" estimates depend quite strongly upon the streaming correction to the migration area. An independent calculation by the Applied Physics Operation using 3-group perturbation theory resulted in $-17 \times 10^{-3} \frac{\Delta k}{k}$ for the 0.25% H_2O case. The reason for this substantially more negative result is not immediately apparent.

Analog runs of PRTR accident studies for a mixed oxide fuel element loading with 56 percent of the fissions occurring in the plutonium were completed. Preliminary analysis of the data for studies which were terminated by reactor scram showed no severe fuel element temperature increase.

PRTR steam generator shield temperature data were checked; no likelihood of structural failure due to concrete deterioration was found.

Response to Electric Boat Divisions analysis of the PRTR primary coolant system for earthquake safety was investigated. Establishment of a definite schedule for accomplishment of items recommended by Electric Boat Division (installation of hydraulic snubbers, recompute piping stresses with snubbers installed, and analyses of steam generator foundation with earthquake loads included) was suggested to Test Reactor and Auxiliaries Operation.

Design Studies

PRTR Power Level Study. A study of the feasibility of increasing PRTR power level indicates that the power may be increased to about 100 MW without alteration of major equipment. Some changes will be required in lesser items of equipment, and more detailed analyses of certain transients and accidents will be required to determine the necessity for or type of change in several cases.

Operating data for certain systems were examined for comparison with design operating conditions to allow an extrapolation to performance at higher powers. Preliminary investigations indicate the following:

1. Process water system and secondary coolant system equipment (pumps, heat exchangers, and relief valves) should be adequate for operation up to 90-100 MW. For higher power levels, replacement of heat exchangers for the moderator, reflector, and shield cooling systems would probably be necessary. The steam generator capacity appears adequate for higher power levels, however.

2. Power levels up to 125 MW should not produce unacceptable temperatures in shim rods and shroud tube bellows.

3. Additional relief capacity for the primary coolant system would probably be required for any significant power raise.

4. Changes in pressurizer and steam generator liquid levels and in pressurizer pressure during a scram should be acceptable for powers up to 125 MW.

The effects of higher powers on the consequences of various equipment failures (e.g., process tube leaks, power failures, etc.) remain to be investigated. A report summarizing the study is now being prepared.

Plutonium Recycle Critical Facility

Hazards Analysis. Five PRCF process specifications were issued for comment. Written material for comment issues of PRCF process specifications is approximately 70 percent complete.

PRTR Rupture Loop

In-Reactor Test Section. Analysis of the data obtained during the eleven thermal cycle tests of the in-reactor test section assembly and during special pressure drop tests of the process tube inlet valve assembly have been completed. Test results are as follows:

The pressure drop for the inlet valve may be determined by the equation

$$\Delta P = \left(\frac{1}{V_f}\right)\left(\frac{F}{427}\right)^{1.995}$$

and the pressure drop from the inlet jumper to the outlet jumper by the equations

Empty tube: $\Delta P = \left(\frac{1}{V_f}\right)\left(\frac{F}{299}\right)^{1.85}$

Loaded tube (fuel basket and two fuel elements):

$$\Delta P = \left(\frac{1}{V_f}\right)\left(\frac{F}{232}\right)$$

Where: ΔP = pressure drop in lb/in.^{2}
V_f = specific volume of the flowing fluid in ft^3/lb.
F = flow in gpm.

The average water leak rates of the various seals, during eleven thermal cycles performed during six loop operating periods were:

1.	Seal cap to nozzle	1.58 ml/hr
2.	Nozzle to process tube	Zero leakage except for the first operating period when it was 22.7 ml/hr. The seal was then retightened and no further leakage was observed for the remaining five operating periods. It is felt that this initial leakage could be prevented if the gasket was formed to fit its seat prior to installation.
3.	Process tube to valve	Zero
4.	Valve stem	Zero
5.	Valve to jumper	This leakage was zero for four operating periods and 0.66 ml/hr and 1.39 ml/hr for two other operating periods.

EDEL-2 was tested to 2100 psig and 250 F. The system is operationally ready for thermal cycling of the Grayloc connector used at the nozzle to jumper connection. To minimize the testing period required for 50 thermal cycles, an air operated valve system will be used to alternately inject hot pressurized water and cold deionized water from EDEL-2 into the Grayloc connector.

Discharge Equipment. Essentially all fabrication is complete and all materials are on hand except for the hose assemblies, which have a promised shipping date of June 26, 1962. Minor modifications and/or alterations are anticipated during equipment testing.

Tool fabrication is approximately 70% complete. Completion of a powered cut-off tool is pending receipt of several off-plant procured items. Four new drawings on special hand tools were issued. Fabrication of metal components for the shielded viewing cart is now complete. Lead glass is being installed in the cart.

Plutonium Fuels Development

Plutonium-Bearing Fuel Elements for PRTR. During the month of June, seventeen 19-rod cluster fuel elements were completed through final assembly, including four Mark I-L Vi-Pac UO_2-PuO_2, twelve Mark I-M swaged UO_2-PuO_2, and one Mark I-M swaged MgO-PuO_2. Nine of these clusters were transferred to the PRTR on June 6 (two Vi-Pac, six swaged and the MgO-PuO_2). Most of the fuel element rods used in making up these clusters were incrementally loaded with UO_2 and PuO_2 in April and have been moving through the latter steps in the fabrication process such as etching, autoclaving and wire wrapping.

All of the fuel element clusters assembled in June used Zircaloy end brackets fitted with wide pads ($\frac{1}{2}$ inch x $\frac{1}{4}$ inch) on the spacing gussets. These pads were added at a late date in an effort to reduce the PRTR process tube wear which occurs at the contact points between spacing gussets and the process tube. At the time the decision was made to increase the contact area of the end brackets on all elements going into the PRTR, one cluster was already assembled with the standard end brackets, and some 15 sets of autoclaved end brackets were finished and ready for fuel element cluster assembly. It was necessary to reshape the gussets on the completed end brackets by machining on a lathe, weld the etched pads in place, remachine the outside diameter over the pads and reautoclave the entire end bracket. This was done over a weekend without delaying the transfer of elements to PRTR. The cluster which was assembled with the unpadded end brackets was cut apart (this was necessary in that it is a welded assembly) and the rods reassembled using end brackets with pads.

A new end bracket design with pads has been made (H-3-14511) and will be used until further information is gained and new recommendations are made.

Vibrational compaction produced 46 rods in an eight-day period with 160 increments each of coarse, medium, and fine UO_2 in each rod. Autoradiography has shown that 160 increment rods are much more uniform in PuO_2 than the previous 80-increment rods.

Modifications are being made so that up to seven rods may be compacted simultaneously. Equipment changes will allow multiple rod fabrication to be evaluated. One limit to the number of rods that can be compacted at one time is that of the 1750-pound transmitted force rating of the electro-dynamic shaker. Seven rods will weigh approximately 50 pounds and the associated rod chucks and table weight will add an additional 38 pounds. Thus, the available acceleration for fully loaded rods will be approximately 20 times the force of gravity. Single rods have been compacted at 60 to 80 g's so some compromise in operational procedure may be required.

Tubing for the swageable rods have been taper-reamed to give a uniform inside diameter in the weld area. Visual inspection of each tube after the first end cap is welded shows excellent weld penetration. It is felt that radiography will reveal the same results on the second end closure after swaging.

Experimental Elements for PRTR Irradiation. The Zircaloy-clad 19-rod $MgO-PuO_2$ fuel element has been completed and sent to the PRTR for irradiation. Irradiation will commence at the next reactor startup.

A Zircaloy clad ZrO_2-PuO_2 19-rod cluster element is being fabricated by swage compaction for irradiation testing in the PRTR. Swaging studies using CaO stabilized arc-fused ZrO_2 have been performed using the standard 0.680-inch ID Zircaloy tubing. A swage density over 88 percent of theoretical is achieved by using the following ZrO_2 particle size distribution:

Mesh Size	w/o
-20 + 60	47
-60 + 100	13
-100 + 325	24.5
-325	15.5

Seven full length UO_2-PuO_2 fuel rods were vibrationally compacted in 304-L thin-wall (0.008-inch thick) stainless steel cladding to fuel densities of 87 to 89 percent of theoretical. Satisfactory fabrication techniques have been developed which prevent collapsing and denting of the tubing during compaction. Difficulties are being experienced, however, with collapsing of the compacted rods during pressure testing. Different techniques for increasing the fuel density in the top two inches of the rod are being tried. It may be necessary to insert a sintered MgO plug next to the end cap for support of the tubing.

Further examination was made on a zirconium clad fuel plate for an extended surface fuel in which a bare plutonium-zirconium alloy core was inserted into the picture frame assembly. The plate was sheared to within $\frac{1}{4}$ inch of the core in all dimensions without encountering alpha contamination. Sections of the plate were destructively tested; no separation between the core and cladding could be made. The cast plutonium 15 w/o zirconium alloy was identified as primarily delta phase plutonium; an unidentified second phase was also detected. The alloy does not seem to harden appreciably on cold rolling to a 10:1 reduction; however, comparative hardness numbers have not been obtained.

Phoenix Experiment. The irradiation and reactivity measurements on the high-exposure aluminum-plutonium samples are continuing in the MTR and ARMF. Current status of the samples is as follows: the sample which contained plutonium with initially 6.25 percent Pu-240 (GEH-21-1) has received five cycles of irradiation and is cooling prior to ARMF measurements; the sample containing plutonium with 16.33 percent Pu-240 (GEH-21-3) has completed its fourth cycle of irradiation and ARMF measurements have been made; and the sample containing plutonium with 27.17 percent Pu-240 (GEH-21-19) is now being irradiated for its fourth cycle. Continuous reactivity measurements were made on sample GEH-21-3 immediately following its first cycle of irradiation to determine the reactivity effect caused by the decay of the short-lived fission products. These measurements will now be repeated following the fifth cycle of irradiation of this sample.

Reactivity measurements on the five aluminum-plutonium-boron samples have been completed in the PCTR and the results of these measurements are currently being analyzed by people in the Reactor Lattice Physics group. The purpose of these measurements is to determine boron concentration and homogeneity in these samples which will be used as poison standards in the ARMF.

Some of the underwater handling tools have been completed and an underwater work platform required for manipulating the Phoenix specimens is being fabricated.

Irradiation Testing. Radiometallurgical examination of the 42-inch long swage compacted uniformly enriched UO_2-PuO_2 seven-rod cluster (GEH-11-7) is continuing. More rods are being sectioned and examined to better investigate the effects of irradiation on this fuel element which had plutonium segregation in the UO_2.

Examination of the 42-inch long UO_2-PuO_2 cosine enriched seven-rod cluster (GEH-11-8) is also continuing in Radiometallurgy. A closer examination of the irradiated fuel element revealed that some loosening of the spiral wire wraps on the fuel rods has occurred. This is the first time that wire loosening has been observed on an oxide type fuel element. Autoradiographs made by exposing the irradiated fuel rods to glass revealed areas of localized fission product concentration, which resulted from segregation of the plutonium. This same observation was made on the uniformly enriched UO_2-PuO_2 element. Each fuel rod was sampled for fission gas and the results are given in the following table.

Rod Number	Total Gas Collected (ml at STP)
Center	51.3
24	37.4
20	37.6
17	46.6
23	49.2
18	40.0
21	39.7

The total gas release data from this experiment are similar to and corroborate the data obtained from the previous seven-rod cluster experiment (GEH-11-7), i.e., the least amount of gas is released from those fuel rods which were closest to the center of the reactor and, consequently, operated at higher temperatures. The most gas was obtained from the center rod which presumably operated at the lowest temperature since it was completely surrounded and shielded by the outer fuel rods. In-reactor sintering of the core, traps the gas and prevents it from being released under these operating temperatures since more gas is released from the lower operating temperature material. If the fuel temperatures were further increased so that center core melting or central voids and columnar grains were formed, the gas release from the higher operating temperature core material would then be greater than from the lower temperature material due to the sweeping action which occurs during recrystallization.

One UO_2-PuO_2 capsule (GEH-14-85) which contains high density UO_2 - 2.57 a/o PuO_2 pellets is being irradiated in the MTR. The exposure to date is about 8×10^{20} nvt. Capsule GEH-14-86, which contains high density UO_2 - 4.13 a/o PuO_2, was not recharged as tentatively planned. The Reactor Safeguards Committee would not approve recharging unless a hot cell leak test on the specimen is performed. At the time of discharge, a possible suspect area on one end of the capsule was noted by reactor personnel. Close visual examinations and photographs at HAPO disclosed no apparent suspect areas on the capsule immediately prior to its return to the MTR. Steps are currently under way to conduct the leak test at the NRTS.

Approval for the instrumentation modifications to the VH-4 Hydraulic Rabbit Facility was transmitted to the MTR. Arrangements are presently being made to obtain and install the equipment which as presently visualized would be a recording rotometer on the loop.

Flow tests were conducted with an assembled dummy unit for the ICARUS Experiment. The unit was moved from the simulated charging position to the mock in-reactor position with a flow of 15.4 gpm. With the unit in this location the upstream pressure was 58 psig and the downstream 54 psig. The specimen, as viewed through lucite tube sections, showed no tendency to vibrate or flutter. When the inlet coolant pressure was raised to 90 psig, the pressure at which the unit was set to trip, the transport vehicle portion of the unit released the fuel element and it dropped six feet in 1.6 seconds. All parts functioned satisfactorily. In discussions with the HAPO representative at the MTR, it appears that the next step is to conduct dummy runs in the VH-4 Hydraulic Rabbit Facility to more accurately determine trip settings.

Parts for the HELIOS Experiment bellows assembly were completed and shipped to the vendor for joining. The specifications for the thermocouple were modified. No word has been received from the vendor yet on whether the modified design can be readily supplied.

Uranium Fuels Development

PRTR Fuel Elements. Grooves were found in two PRTR process tubes that had contained UO_2 fuel elements since reactor startup. The grooves were related to spirally wrapped spacer wires on single rods and on the fuel bundles. The two elements were removed from the reactor for additional examination. Examination of the two process tubes that had contained the Mark II-C, nested tubular fuel element revealed no appreciable wear and the fuel element was in excellent condition. The accumulated exposure for this element was approximately 1300 MWD/T_U.

Fused UO_2 originally containing uranium nitrides was characterized after heat treating 12 hours in moist 1750 C hydrogen. Metallographic examination and nitrogen analyses by a modified Kjeldahl technique revealed that uranium nitrides (UN, U_2N_3, UN_2) had been removed.

High Energy Rate Compaction. The design has been completed for modifying a 22.5 KVA r atmosphere furnace for heating capsules prior to compaction. Sixteen capsules, four inches in diameter and six inches long, will be heated simultaneously. A transfer mechanism will remove an individual capsule from the furnace and deliver it to the compaction dies. Other improvements in this process for preparing high density UO_2 particles include: (1) elimination of iron contamination of the UO_2 by changing from Type 304 to Type 310 stainless steel containers; (2) reduction of the heating time, from 75 minutes to 60 minutes, by increasing the bulk density of the UO_2 powder, before compaction, from 44 to 53 percent T.D.; and (3) quenching of the high energy impacted UO_2 in water to prevent surface oxidation during cooling and to improve the fracture characteristics of the material during the subsequent crushing operation. Experiments using tungsten carbide tooling and evacuated capsules also are in progress.

Fuel Cladding Evaluation. A fuel element clad in AISI-406 stainless steel, which failed during irradiation, was examined in the ETR hot cells. No evidence of failure was found. The examination included a leak test in which the rods were immersed in liquid nitrogen and then in alcohol; a sample of the alcohol was analyzed for fission products, but none were found. The element was returned to Hanford for destructive examination.

Cladding Studies. A single unit, 19-rod cluster fuel element cladding assembly was designed for fabrication by the high voltage electron beam welding process.

Materials Development

Effect of Dissolution of the Oxide on the Oxidation Rate of Zr-2. Work is continuing to investigate the effect of vacuum heating of autoclaved films on the protective character of the oxide film in reducing the rate of subsequent oxidation. Zr-2 specimens were prepared by annealing at 700 C for 120 minutes, etching and autoclaving to form a 20 mg/dm^2 film (13,500 Å). Autoclaving before and after vacuum heating was carried out in water at 400 C and 1500 psi. Controlled oxide film dissolution into the metal was performed in vacuum at 600 and 700 C. Heating the film for times up to 20 minutes at 600 C did not affect the oxidation rate, while heating for longer than 20 minutes resulted in loss of the protective character of the film. Dissolution of approximately 1500 Å of the film was required to affect the subsequent oxidation rate. Dissolution of 2000 Å of the film resulted in an oxidation rate about equal to that on the unfilmed surface. Complete dissolution of the film resulted in a rate 2-1/3 times the rate on the original etched surface.

Deionization System for Fuel Element Rupture Loop. All tests conducted to date with the makeup deionization system indicate that the system performance will meet the design criteria. Final evaluation is being delayed until filtered Columbia River water is available for use as the influent stream.

A literature survey was conducted to evaluate the radiation stability of the cleanup system ion exchangers and the possibility of obtaining significant resin degradation from rupture products. The resin should be able to withstand an accumulative radiation dose of about 10^7 RAD without suffering significant damage. At radiation doses of 10^9 RAD, decomposition of the anion resin appears to be complete. A preliminary review of the irradiation dose expected from the rupture products of a uranium oxide fuel element indicate that the dose could exceed that required to decompose the resin in a relatively short time period. If the particulate rupture products were removed mechanically before the resin has an opportunity to collect them, the problem would be less severe.

PRTR Tube Monitoring. The 85 PRTR pressure tubes were visually inspected during May and June to determine the extent of fretting corrosion occurring at the contact points between the fuel elements and the pressure tubes. During visual examination the depth and location of each corrosion mark was measured. One tube had 78 marks with a maximum depth of 22 mils (channel 1257). The maximum depth measured was 26 mils, which had occurred during 4.6 months of operation (channel 1356). Both tubes were removed from the reactor for destructive testing. The tube from channel 1243 with a 14-mil deep mark was removed from the reactor prior to obtaining burst test results from the tubes with deeper corrosion marks. This tube is being retained in the PRTR storage basin and may be reused in the reactor.

A general summary of inspection results according to fuel element type is given in the following table.

Fuel Type	Number of Tubes with Marks of this Depth: 0 - 5 mils	6 - 10 mils	Greater than 10 mils
UO_2	26	3	1
UO_2 + Moxtyl	6	3	4
UO_2 + PuAl	13	1	1
PuAl	19	5	3
Total	64	12	9

Present data are being analyzed and additional data on flow, vibration, fuel element dimensions and hanger types are being obtained to determine what reactor operating variables affect the fretting corrosion. Inside diameters and gas gaps were measured on six old tubes and the three replacement tubes. No unusual changes in gas gap were noted.

Post-Irradiation Evaluation. Two sections of the pressure tube from channel 1257 were burst at room temperature. One section from the annealed portion of the tube contained the lower fuel element support marks, maximum depth of 9 mils, one single rod mark 9 mils deep, and a bundle wrap mark with a maximum depth of 11 mils. The tube burst at a hoop stress of 107,000 psi compared to an average value of 97,000 psi for unirradiated annealed tube sections. One section from the cold worked portion of this tube contained the following fretting corrosion marks. Single rod wrap marks varying from 10 to 17 mils deep, bundle wrap marks with a maximum penetration of 14 mils, and two areas where the fuel rod contacted the tube wall and caused a penetration of 2-4 mils over a length of several inches. This section burst at 130,000 psi hoop stress in comparison to 110,000 psi for unirradiated cold worked sections.

A section from the cold worked portion of the pressure tube from channel 1356 burst at 120,000 psi hoop stress. This section contained the following fretting corrosion marks: bundle wrap mark extending for 13 inches along tube varied in depth from 7 mils to 26 mils; single rod wrap marks up to 13 mils deep, and upper fuel element support marks with depths of 16, 20, and 24 mils. It was concluded from an analysis of these results and previous burst test data that the pressure tubes in-reactor have sufficient strength and ductility to permit safe operation of the reactor.

2. PLUTONIUM UTILIZATION STUDIES

Plutonium Oxides

Additional thermal expansion data have been obtained on PuO_2 up to 1620 C using the high temperature diffractometer attachment. The samples were run on a molybdenum filament in air at about one millimeter of Hg. The expansion was uniform to about 1275 C, and then a sharp change in slope was seen on the $\Delta a/a_o$ plot. This is probably due to a loss of oxygen in the sample and is consistent with the results seen in other experiments. A sample of silver was also run and the high temperature lattice parameters agreed well with those found in the literature, thus providing confidence in the temperature measuring system.

Plutonium Carbides

Comparison of experimental and calculated densities of arc cast non-stoichiometric single-phase PuC alloys has shown that the defect lattice occurs with a carbon deficiency. That is, the defect phase exists with four plutonium atoms per unit cell and the number of carbon atoms decreases with concentration reaching a minimum of 3.2 atoms per unit cell at 39.8 a/o C. The fact that PuC exists with a carbon deficiency was reported previously; however, extension of the previous data to compositions below 40 a/o C shows some evidence of plutonium solubility in the defect PuC structure. Indications of a solubility are the changing lattice parameter of PuC in the region between 20 to 39.8 a/o C. This occurs in a two-phase alpha plutonium plus PuC field and is contrary to normal phase equilibria. In addition, densities in this region are slightly below those calculated which also indicates plutonium in solution since there exists a lower quantity of a 19.50 g/cc phase. Slight intensity shifts in the NaCl type PuC_{1-x} structure below 40 a/o C point toward additional scattering, again possible due to plutonium in solution.

Plutonium Nitride

A 14-gram rod of alpha plutonium was converted to plutonium hydride, PuH_3, by reacting it with high purity hydrogen at 150 C for one hour. During this hydriding process the plutonium rod disintegrated into fine particles. The plutonium hydride was then heated for two hours in nitrogen at approximately 600 C. After grinding to -200 mesh powder, the product was analyzed by x-ray diffraction. X-ray results show the powder consists of plutonium mononitride, PuN, plus a slight trace of plutonium dioxide, PuO_2. The oxidation probably occurred during handling in air atmosphere. A small sample of this material was heated on a tungsten ribbon to determine its melting point under one atmosphere of flowing nitrogen. At 2270 C the portion of the specimen adjacent to the tungsten ribbon looked as if it had sintered. This was probably due to the melting of the finely dispersed PuO_2 impurity. No other change occurred until 2675 C was reached; volatilization without melting began at this temperature. The temperature was then raised to 2900 C, resulting only in a more rapid diminution of the specimen. Although the melting point for PuN under one atmosphere of argon was not discernible during heating, examination of the specimen after heating showed that the particles had coalesced to form a dense piece. The temperature at which PuN either decomposes or volatilizes under one atmosphere of argon is 2675 C. The melted PuN could not be separated from the tungsten ribbon, thus indicating a possible high temperature reaction between the two. X-ray diffraction analyses of the PuN showed only PuO_2 and tungsten as the impurities. A PuN lattice expansion of approximately 0.1% was also observed by x-ray diffraction. This could be a result of nitrogen loss from the PuN cell during the melting point determination.

3. UO_2 FUELS RESEARCH

Single Crystal UO_2 Studies

Post-irradiation studies of a single crystal of UO_2, irradiated to 5000 MWD/T, revealed no irradiation-induced changes that would seriously affect the usefulness of UO_2 as a reactor fuel. The UO_2 single crystal exhibited a notable capability of retaining fission gas. Less than 0.03 percent of the xenon and krypton diffused from the specimen during irradiation. No significant alteration of microstructure, and no change in high temperature properties were observed. An increase in hardness and a slight dilation of the crystal lattice that occurred during irradiation are attributed to irradiation induced lattice point defects. The release of sorbed gases, primarily hydrogen, was ten times greater during irradiation at 500 C than during a four-hour pre-irradiation anneal in vacuum at 1000 C. Details of this work were summarized in a paper, "Irradiation of UO_2 Single Crystals," presented at the June 18-22, 1962, ANS Meeting.

UO_2 Thermal Conductivity

Measurements on a large, UO_2 single crystal revealed that the thermal conductivity increased with temperature above approximately 700 C. The conductivity passed through a minimum (0.052 watt/cm - °C) at approximately 700 C, and increased to approximately the room temperature value (0.075 watt/cm -°C) at 1200 C. These data provide the first experimental confirmation of earlier predictions based on theoretical studies and post-irradiation fuel examination.

The same crystal was subsequently irradiated to approximately 10^{14} fissions/cc. Initial post-irradiation thermal conductivity by measurements at BMI showed a ten to twenty percent decrease in thermal conductivity at temperatures to 750 C. No annealing recovery of thermal conductivity occurred. Further measurements to 1200 C are in progress.

High Temperature Ductility of UO_2

High temperature, torsional deformation tests of UO_2 were performed at temperatures greater than 2000 C to determine the effects of temperature and strain rate. These data would help interpret fuel relocation phenomena and formation of gas bubbles and large voids in UO_2 operating as a reactor fuel at high temperatures. Preliminary tests revealed that a permanent 90-degree twist can be easily produced in UO_2 at 2300 C, with a twisting rate of approximately three degrees per second. Cracks, which formed in some specimens during heating, showed evidence of vapor phase mass transfer down the thermal gradient. The hot sides of the cracks were smooth and polished, while many small crystals of UO_2 were deposited on the cool sides.

Resistance heated tungsten radiator strips with ends held by heated tungsten collets, surround 1/10 x 1/4 x 1-1/4 inch sintered UO_2 plates in these experiments. When the plate reached the desired temperature, one collet was rotated with respect to the other to transmit a shear stress to the UO_2.

High Exposure UO_2

A 170-X photomosaic of the entire transverse cross section of a high exposure fuel capsule (GEH-14-177) was prepared. Examination of the mosaic revealed partial bonding of originally discrete particles near the cladding by grain growth across particle boundaries, even though bulk fuel temperatures in that region did not exceed approximately 500-600 C during irradiation.

UO_2 Relocation Experiment

Preliminary examination of an irradiated fuel capsule containing tungsten marker wires to provide quantitative relocation data revealed a narrow split ($\sim 1\frac{1}{2}$ inches long x 1/16 inch wide) in the aluminum cladding. The split was not associated with the weld areas or with any discernible mechanical defects. Destructive ceramographic examination is expected to provide relocation data; the fact that the test was terminated after a short time at full power means that the effects of power fluctuations and thermal cycling should not complicate interpretation of the results.

Single Crystal UO_2 "Standards"

Five, 0.185-inch OD diameter fused UO_2 spheres and one 4-gram sample of electrodeposited (LiCl-KCl bath) UO_2 were characterized and sent to the Radiation Laboratory, University of California, Berkeley, for use in fundamental UO_2 studies.

Solid Solution Properties of UO_2-U

A small amount of uranium powder was mixed with UO_2 in an inert atmosphere glove box. The mixture will be high energy impacted and heat treated to study the solid solution properties.

Electron Microscopy

Small pieces chipped from the edge of a UO_2 single crystal were examined by reflection electron microscopy. The chips will be replaced in their original positions on the crystal and irradiated for very short times, prior to re-examination of the same surfaces. An auxiliary column for the electron microscope was installed.

4. BASIC SWELLING PROGRAM

Irradiation Program

Two general swelling capsules still under irradiation have reached their goal exposure and will be discharged at the next reactor shutdown. Two additional capsules are complete. These capsules will be charged for irradiation at the earliest opportunity. Two capsules previously irradiated are being shipped from the reactor to Radiometallurgy for disassembly and examination of the specimens. The post-irradiation examination of the split, hollow, uranium cylinders recovered from the two disassembled capsules in Radiometallurgy is continuing. Details of this examination follow in a succeeding section under "Post-Irradiation Examination." Additional assembly of general swelling capsules has been delayed pending the receipt of electrical resistance heaters previously ordered.

The assembly of a prototype metallographic capsule is almost complete. The heat transfer characteristics and the integrity of modifications will be determined with intensive laboratory testing. The capsule contains a uranium specimen that has one surface metallographically prepared and pre-characterized. It is intended that this capsule be thermally cycled a large number of times between ambient temperature and about 600 C to determine both the surface roughening that occurs on cycling and the NaK uranium interaction.

Post-Irradiation Examination

Efforts to polish and etch the specimen that was irradiated to 0.4 percent B.U., at a nominal control temperature of 575 C have not been successful due to the warped and brittle nature of the sample. Additional attempts are being made to examine the microstructure. The samples annealed in the laboratory capsule which duplicated the in-reactor thermal history of this capsule are currently undergoing metallographic examination. Nothing unusual has yet been observed.

Post-irradiation annealing of Zircaloy-2 clad coaxial U-U diffusion couples irradiated to two burnup levels (GEH-14-334 and 336) has been extended to 800 C and 990 C. The outer shell of enriched uranium had sustained a nominal burnup of 0.2 a/o and 0.4 a/o during the two exposures; the depleted central core achieved burnups lower by a factor of 20. Light microscopy of the specimen annealed at 800 C for 100 hours showed extensive interaction between the clad and the uranium. Large grains are now present in the cladding. The grain structure in the uranium is clearly discernible and extends across the U-U interface. Severe circumferential cracking exists in the enriched uranium at and near the clad interface but the U-U interface is intact. Limited cracking is observed throughout the remainder of the samples. Replicas are being prepared for electron microscope examination. Replicas stripped from specimens annealed at 700 C await examination. The 990 C anneal was interrupted after 45 hours due to a leak that developed in the vacuum system. The specimen will be examined with this heat treatment rather than reheat for the additional time to complete the 100 hours.

5. IRRADIATION DAMAGE TO REACTOR METALS

Alloy Selection

An alloy whose mechanical properties indicate that it may have potential use in high temperature nuclear applications is Alloy R-27 developed by the Allegheny Ludlum Steel Corporation. A cursory examination of the effects of various oxidizing environments and irradiation at high temperatures is now in progress. Corrosion and oxidation specimens have been prepared. These specimens will be tested in 1800 F CO_2 and 680 F water for periods of time to 100 hours. In addition, tensile specimens are being prepared and will be irradiated in a gaseous environment at 1200 F.

New alloys with high temperature strength suitable for use as reactor structural materials should be tested to determine the effects of irradiation on mechanical and physical properties of the alloys. Hastelloy-C has good tensile, creep, and stress-to-rupture properties to 1600 F. This material, usually thought of as a casting alloy, can be easily formed into plate, bar and sheet. Its nominal composition is as follows: Ni, 54; Co, 2.50; Cr, 15.0; Mo, 16.0; W, 4.0; Fe, 5.50; Si, 1.0; Mn, 1.0; C, 0.08. The cobalt content can be reduced to less than 0.1 percent by proper selection of nickel going into the heat.

In-Reactor Measurement of Mechanical Properties

An in-reactor creep test on a 20 percent cold worked Zircaloy-2 specimen at 250 C and 30,000 psi stress is in progress. The test has been running more than 1000 hours during which time all capsule systems have functioned properly. A minor difficulty involving the effect of reactor water pressure changes on the temperature of the strain measuring system has been corrected. Correction of this difficulty was accomplished by removing cooling water from the capsule interior. The cooling water is usually needed to remove excess gamma heat from the capsule; however, nuclear activity in the vicinity of this capsule is low enough so as to eliminate the need for internal cooling water.

Creep rates and strains at 250 C and 30,000 psi are very low, making analysis of the details of the creep curves difficult. The general form of the in-reactor test at 250 C is similar to that of the 310 C test in that creep rates with the reactor on are very low but increase when the reactor shuts down. The magnitude of the increases in creep rate during reactor down conditions is much lower in the 250 C test than in the 310 C test. In fact, the rates during reactor down periods at 250 C are barely detectable in the relatively short time afforded by shutdown periods.

Further analysis of the 250 C in-reactor creep test revealed an abrupt increase in creep rate at 670 hours. The abrupt change in rate decreased rapidly with time giving a rate of 760 hours equivalent to that which existed before the onset of the transient. The strain associated with the transient amounted to 0.03 percent. Detailed examination of all available data concerning capsule and reactor operation has not revealed any experimental condition which might account for the transient.

The total creep strains of the 250 C - 30,000 psi ex-reactor and in-reactor tests are 0.107 percent and 0.140 percent, respectively. The creep rate in-reactor is too low for accurate measurement. The ex-reactor rate is 2.5×10^{-7} in/in/hr.

The 350 C - 30,000 psi stress in-reactor creep test has not been started because reactor operation has not allowed access to the equipment for application of stress to the specimen. This condition will be corrected shortly. The 350 - 30,000 psi test will be started immediately after correction.

Irradiation Effects in Structural Materials

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

During the month 59 Zircaloy-2 tensile specimens irradiated in the G-7 ETR hot water loop were tested. Of these specimens, 27 were tested at room temperature and 32 were tested at 300 C. The raw data from these tests were programmed for electronic data processing. A rapid technique has been perfected for the remote tensile testing of irradiated specimens at elevated temperatures. By employing a molten salt bath for temperature control of the specimen during the test, approximately seven specimens can be tested per shift compared to two or three by conventional furnace techniques.

Cold worked, Zircaloy-2 tensile specimens representing the transverse direction and irradiated to about 10^{21} nvt at 545 F have exhibited unusual features in yielding and fracture. It is characteristic of these specimens that immediately upon yielding the strain becomes non-uniform and the load falls off with strain to fracture. Thus, the uniform plastic strain is generally less than one percent. In some instances delayed yielding has been observed, which is manifested by a sudden drop in load beyond the initial yield point. Many of the specimens in this series also exhibit a dual mode of fracture, consisting of an internal ductile tear and a macroscopic shear. The tear occurs first and is associated with a rapid drop in load. Then, the fracture is completed by shear, which is accompanied by a linear decrease in load with strain. Necking associated with this fracture is both extensive and localized. The nature of the fracturing mechanism is being further investigated by metallographic examination.

Titanium, 99.45 percent or greater purity, was received for the monitoring of fast neutron flux. The order consisted of 3000 feet of 0.020-inch wire and ten feet of 0.5-inch rod. Maximum certified iron content in both items was 0.08 percent by weight. Attempts will be made to further purify the rod for future monitoring applications. Activation data for iron, nickel, and titanium flux monitors were obtained for G-7 loop experiments conducted during ETR Cycles 39, 40 and 41. These data are being programmed into a computer code written to characterize the neutron flux spectrum for selected positions in the G-7 loop facility.

Notched tensile specimens of unirradiated Zircaloy-2 and type 347 stainless steel were tested during the month. The stainless steel specimens contained 0, 10, 20, and 40 percent cold work and represented both the longitudinal and transverse directions of cold work. The ratio of notch stress to ultimate tensile stress, known as the notch sensitivity, was very close to 1.0 for these specimens. Materials are considered notch sensitive if the above ratio is less than one. This ratio for Zircaloy-2, corresponding to the same cold work levels, is greater than 1.3 in all cases. Tests were also conducted at room temperature on Zircaloy-2 specimens hydrided to 100 and 5000 ppm hydrogen, respectively. Whereas, the former specimen exhibited little or no change in properties over the unhydrided control specimen, the latter specimen fractured in the process of mounting it in the tensile grips. Specimens containing intermediate hydrogen contents are being prepared for testing.

Damage Mechanisms

The damage sustained by iron during irradiation as a function of neutron dose and interstitial impurity content will be studied in this program.

Optical metallography of high purity iron obtained as stock for the introduction of impurities is continuing. High purity iron foils, obtained from Johnson-Mathey, have been thinned electrolytically for transmission electron microscopy. For acceptable results the distortion of the image due to magnetic interactions between specimen and objective lenses will have to be eliminated or reduced. The effect of specimen size and geometry is currently being investigated. Equipment essential to the processing, heat treating, and testing of high purity metals is being assembled.

Testing has proceeded on a number of ingot iron specimens representing six levels of neutron exposure. The exposures range from 1×10^{17} nvt (fast) to 5×10^{18} nvt. The values of upper and lower yield strength and total elongation appear to approach saturation at the highest exposure. The maximum increase in upper yield strength was 21,000 psi. Elongation decreased from an unirradiated value of 37.5 percent to 9.0 percent at 5×10^{18} nvt. The strain-aging characteristics of ingot iron do not seem to have been affected by irradiation. Further strain-aging tests are in progress to verify this observation.

An experiment is being designed to supply information on the extent that neutron damage in a metal is affected by irradiation temperature. Iron, nickel, and zirconium will be irradiated at various temperatures to establish whether saturation of damage occurs below a critical temperature.

Inconel and Inconel-X sheet heat treated according to a specific schedule is being prepared for study by transmission electron microscopy. The oxide film which formed on the specimen surfaces during heat treatment is difficult to remove. Pitting and preferential attack at precipitates occurs during electrolytic thinning.

Aluminum foils, 0.003-inch thick, which were quenched from 620 C into iced brine to develop a large number of prismatic loops, have been irradiated to $\sim 2 \times 10^{18}$ nvt (fast) and examined by transmission electron microscopy. These foils are far more difficult to thin than irradiated aluminum which had not been quenched prior to irradiation. In the specimens examined there is a very large reduction in the number of quenched in prismatic loops. Foils irradiated to higher exposures will be examined shortly.

6. GAS-GRAPHITE STUDIES

EGCR Graphite Irradiation

The fourth capsule, H-3-4, in the series of irradiations of EGCR graphite continues to operate satisfactorily in the GETR. The estimated maximum sample exposure now exceeds 1×10^{22} nvt, $E > 0.18$ Mev. Sample temperatures are in the normal 450-825 C range and all thermocouples are functioning properly.

Flux Intensity Test

The GEH-13-8 graphite irradiation capsule designed to study the effect of flux intensity on property changes at a controlled sample temperature has successfully completed its third cycle of irradiations in the Engineering Test Reactor. The sample temperature of the four positions having electrical heaters are being controlled at 650 C.

Irradiation of Graphites Containing Additives

Results have been compiled for the first Hanford hot test hole irradiation of graphites supplied by Speer Carbon Company under contract DDR-118. The accumulated exposure at approximately 600 C was 1745 MWD/AT_K. All of the graphites show expansions transverse to the extrusion direction ranging from +0.001 to +0.013 percent and contractions parallel to extrusion in the range 0.001 to 0.010 percent. Comparison of additive effects based on these small dimensional changes is inconclusive and further irradiation is in progress.

Graphites in which the effect of additives on dimensional behavior was first noted were recently discharged with a total exposure of 9500 MWD/AT_K. Transverse contraction of a graphite prepared with one percent Fe_2O_3 additive was 0.09 percent, while a graphite prepared from the same mix without additive contracted 0.13 percent. In spite of this difference, the contraction rates between 2500 and 9500 MWD/AT_K were very similar: 0.014%/1000 MWD/AT_K for the graphite containing additive and 0.016%/1000 MWD/AT_K for the graphite without an additive. The large amount of initial growth which occurred in the graphite prepared with the additive accounts for much of the difference in contraction observed at higher exposure.

Irradiation of Raw-Coke Graphite

Graphite samples prepared by Great Lakes Carbon Corporation with a process utilizing a raw-coke filler were discharged with an accumulated exposure of 6900 MWD/AT_K at approximately 600 C. Growth of 0.048 ± 0.001 percent occurred in the direction parallel to the force applied during molding. Growths of 0.021 ± 0.006 and 0.026 ± 0.005 percent also occurred in two directions perpendicular to the molding direction. Growth in all directions has not previously been observed in hot test hole irradiations of equivalent exposure.

Graphite Compression Test

A comparison of the x-ray diffraction data obtained from the 150 psi compression test boat number P-5 after 1.52×10^{21} nvt ($E > 0.18$ Mev)* has been made. No significant difference was found between the x-ray diffraction data of the compressed and uncompressed samples. This result is in agreement with data obtained at much lower exposures.

Neutron Exposure Units

It has long been realized that MWD/AT is not a sufficient exposure unit for comparing the irradiation damage sustained in different test facilities. Dimensional changes due to irradiation in the Hanford reactors and in the GETR test reactor have been shown to be in good agreement when compared on the basis of fast neutron exposure. To aid those familiar with the MWD/AT exposure data in making the transition to fast neutron exposures, the following relationships are offered:

10^{20} nvt ($E > 0.18$) $\simeq$ 425 MWD/AT in a K reactor H.T.H.

$\simeq$ 735 MWD/AT in a C reactor H.T.H.

$\simeq$ 1150 MWD/AT in an old reactor C.T.H.

Microwave Activation of Gases

Analysis of a deposit formed during microwave activation of carbon monoxide gives rise to an empirical formula of C_5O. It appears possible, however, that some of the carbon came from decomposition of black wax in the system and that the true CO deposition was of carbon suboxide polymers, $(C_3O_2)x$ and $(C_2O)x$. X-ray d-spacings of 2.851 and 2.493 were determined for the deposit.

*K Reactor H.T.H. Exposure, 6485 MWD/AT.

A graphite sample placed in a CO glow discharge was extensively "sooted" by a black deposit. The surface area (BET) of the sooted sample had doubled. Vacuum cleaning of the soot essentially restored the surface area of the sample to its original value. Another sample was exposed to a helium glow discharge for a similar period of time (8 hours); no significant change in surface area was observed. Neither sample showed changes in helium density after the microwave treatment.

Elastic Modulus Determination

Preliminary investigations with metal samples preparatory to graphite determinations have been completed. The dynamic moduli as determined by an Elastomat machine for round samples of five different metals agree with the accepted values with the exception of copper.

To determine the effect of the stepped quarter-round shape now being used in the GETR tests, four full length quarter-round samples of 304-L stainless steel were obtained and resonant frequencies determined. The elastic moduli computed from these frequencies with analytically determined equations agreed within two percent with those determined by the same means for a round sample of the same material. The same full-length quarter rounds were then cut down to the stepped quarter-round configuration, and resonant frequencies again found. With this information the effect of the further complication of the shape can be determined. The analytically determined torsional resonant frequency of this shape agreed within less than two percent with the actual measured frequency.

Graphite samples will be similarly tested to determine the general applicability of the derived equations.

7. GRAPHITE RADIATION DAMAGE STUDIES

National Carbon Company Research and Development Contract

Initial data were obtained from samples of the controlled orientation series with ratios for the transverse-to-parallel coefficient of thermal expansion from 0.8 to 6.2, and from the particle-size series to an exposure of 1326 MWD/AT at temperatures above 600 C. Because of the relatively short exposure no significant differences were apparent among the grades which all showed a slight expansion in the transverse direction.

Samples of ZT-4130, a hot worked material with an apparent density of about 1.95 g/cm^3, were discharged after an exposure of 3120 MWD/AT. In the transverse direction no net contraction occurred. At the same exposure samples of CSF have displayed a contraction of at least 0.05 percent. Thus, this material may prove to be more stable than CSF.

8. ALUMINUM CORROSION AND ALLOY DEVELOPMENT

Corrosion Testing

Testing of A288 and X-8001 aluminum, A212 C/S, Zr-2, and 306 S/S continued at 300 C and a pH of 4.5 adjusted with chromic acid. After 2690 hours the corrosion rate of all aluminum alloys is approximately 1.1 mil/mo. The corrosion rate of the A288 alloy underwent an increase from 0.5 mil/mo during the earlier portion of the exposure to 1.1 mil/mo after about 1400 hours. The corrosion rate of the A212 C/S after 2810 hours is 0.0054 mil/mo; that of the 304 S/S, less than 0.001 mil/mo; that of the Zr-2, 0.004 mil/mo. Some pitting of the A212 C/S has been observed in crevice areas. In general, this pitting is on the order of 1-2 mils deep and scattered in nature.

The test in H-1 Loop studying the in-reactor corrosion and crud deposition on coupons of X-8001, A288, 304 S/S, and Zr-2 continued during the month. The test has accumulated about 372 hours of exposure at 500-550 F in high purity deionized water. The test is scheduled to last 1000 hours.

Dynamic Corrosion of X-8001 and KY (1.8 Fe, 1.2 Ni) Aluminum Alloys

In the current test the effect of prefilming aluminum samples at 360 C is being investigated. X-8001 and KY alloys were prefilmed for 14 and 30 days. Dynamic corrosion testing was carried out at 330 C with a flow rate of 25 fps and a refreshment rate of 9.4 gph/80 cm^2 of aluminum area. After one month, average corrosion penetrations on unfilmed samples are 2.14 and 1.64 mils for X-8001 and KY, respectively. Prefilming resulted in rate reductions of 27% in the case of X-8001, 30% for KY prefilmed for 30 days and 15% for KY prefilmed for 14 days. Penetrations for prefilmed samples include the corrosion increment from the autoclaving cycle. The test will be continued to 90 days.

9. **AEC-AECL PROGRAM**

Tests were run to establish the parameters for an ultrasonic test which will give equal response to longitudinal defects on the inside and outside of thin wall tubing. Notches 3 mils deep by 30 mils long were electro-jet machined on both the inner and outer surfaces of 0.680-inch ID by 0.035-inch wall tube. The ultrasonic responses to these notches were measured as a function of entry angle with 10 MC lithium sulfate line focused, spot focused and flat rectangular transducers.

The line focused transducer gave the best results of the three types tested. The curves of signal amplitude plotted as a function of distance the transducer is offset from the axis of the tube (equivalent to changing the entry angle) indicate two points at which the signals from inside surface and outside surface defects are equal in amplitude, 70 mils offset and 115 mils offset. The slopes of both inside and outside surface response curves at the 70-mil offset point are flatter than the slopes at the 115-mil offset point, indicating that adjustment for offset is less critical. The 70-mil offset position is the recommended operating point for this transducer.

The curves of response versus offset distance obtained when using the spot focused crystal show three points where the response from inside and outside surface defects are equal. However, the slopes of the curves at these points are very steep indicating the tube-transducer alignment is too critical for practical usage. The signal-to-noise ratio of this crystal is superior to that of the line focused crystal; however, the line focused crystal is adequate for this test.

When using the flat rectangular crystal, the amplitude of the response from the outside surface defect was larger than the response from an inside surface defect by a factor of about 14 making it unsuitable for this test.

The boiling burnout runs for the electrically heated 19-rod bundle test section with 0.015-inch spacing between rods were completed with the determination of four additional points. This makes a total of 19 experimental burnout points obtained with this test section. In addition, 17 boiling burnout points were obtained with a 19-rod bundle test section with 0.050-inch spacing between rods. Both test sections were run in a 3.25-inch ID horizontal coolant tube.

The 0.050-inch spaced test section was constructed in a manner quite similar to the previous test sections for this program. The rods were made from Inconel tubing, 0.587-inch OD, 0.0085-inch wall thickness, and had a heated length of 19.5 inches. Twelve of the rods were wrapped with 0.050-inch wire to maintain spacing and promote flow mixing. The thermocouple installation was modified from previous test section and made use of special "temperature sensing plugs". The plug consisted of a thermocouple attached in the center of a short copper cylinder and was inserted into the downstream end of each Inconel tube. Each plug was slotted radially to prevent a circumferential heat shift when a point on the rod attained high temperatures associated with film boiling. The plugs proved very satisfactory for detection of boiling burnout at any point at the downstream end of the test section.

The tests with the 0.050-inch spaced test section were performed in a manner similar to that used with previous test sections. Flow and inlet water temperature were held constant while the power to the test section was increased in steps until boiling burnout was reached. The onset of boiling burnout was defined by a temperature excursion on one or more of the surface thermocouples. A surface temperature increase in excess of 50 F, compared with increases of approximately 2 F per power increase step, was deemed to be an excursion. The tests were performed at a pressure of 1200 psig.

The testing with the 0.015-inch spaced bundle was stopped after several of the thermocouples had failed and several small leaks had developed in the section. The testing with the 0.050-inch spaced bundle was stopped by failure of two of the rods of the bundle. The bundle was being disassembled for repair at the month's end.

There were several observations made during the course of these experiments which appear significant. A total of seven or possibly eight different rods indicated burnout at one time or the other during the tests with the 0.015-inch spaced bundle, these being six or seven of the inner seven rods and one of the outer rods. The seven inner rods of the 0.050-inch spaced bundle showed burnout at one time or the other. On several occasions with both test sections more than one thermocouple indicated burnout, occasionally as many as four at one time. There was no pattern as to which thermocouple or thermocouples might indicate burnout for a given run except that the burnout tended to occur on one of the upper rods during the low coolant flow rate runs. This indicates some stratification.

The boiling burnout heat fluxes obtained with these test sections are reduced from those obtained at the same bulk coolant conditions with a test section which had a 0.074-inch spacing between rods, the reduction being more significant with the 0.015-inch spaced test section. The 0.074-inch spaced test section had boiling burnout heat fluxes at the lower edge of a band encompassing a large number of experimental fluxes obtained with tubular and annulur test sections. The reduction in the burnout heat fluxes was most pronounced at low coolant flow rates. At a coolant flow rate of 500,000 lb/hr-sq ft, reductions from the 0.074-inch fluxes of about one-fifth and one-half were found with the 0.015 and 0.050-inch spaced test sections, respectively. At a coolant flow rate of 3,000,000 lb/hr-sq ft, the 0.015-inch spaced test section had burnout heat fluxes about one-half of those of the 0.074-inch spaced section, while the 0.050-inch spaced burnout fluxes were but slightly reduced.

The burnout heat fluxes with these test sections showed a strong dependence upon the coolant flow rate. For example, the heat fluxes obtained with the 0.015-inch spaced test section at a coolant flow rate of 500,000 lb/hr-sq ft were quite low (about 150,000 Btu/hr-sq ft) and were rather independent of the bulk coolant enthalpy from 11 percent quality to 150 F subcooled. The 0.050-inch spaced test section at the same coolant rate were about 300,000 Btu/hr-sq ft and were slightly dependent upon bulk coolant enthalpy from about 18 percent quality to 115 F subcooled. On the other hand, at a coolant flow rate of 3,000,000 lb/hr-sq ft, the 0.015-inch spaced bundle had burnout heat fluxes ranging from 622,000 to 953,000 Btu/hr-sq ft at bulk coolant conditions of 5 percent quality and 146 F subcooled, respectively. At the same coolant flow rate the 0.050-inch spaced bundle had burnout heat fluxes ranging from 750,000 to 1,150,000 Btu/hr-sq ft at coolant conditions of 5 percent quality and 30 F subcooled, respectively. For both test sections the burnout heat fluxes at coolant flow rates of 4,000,000 lb/hr-sq ft were not much higher than those obtained at 3,000,000 lb/hr-sq ft indicating that the strong effect of the coolant flow rate on the burnout level may become less pronounced at higher flow rates. The foregoing points out that reducing the rod spacing from 0.074 inch to 0.015 reduces the boiling burnout heat fluxes significantly, particularly at the lower flow rates. Reducing the rod spacing from 0.074 inch to 0.050 inch reduces the boiling burnout heat fluxes somewhat, but significantly only at the low flow rates.

10. REACTOR STUDIES PROGRAM

Advanced Reactor Concepts Studies

For convenience in calculation of performance of a boiling metal cooled fast reactor for spacecraft missions, a conventional fuel pin type core was selected as the first model to be studied. Critical mass calculations based on Pu-239 loading indicate that a cylindrical core 40 cm in diameter and height should be adequate for a 20,000-hour life. To decrease the control span required, alternate fuel loadings of Pu-239, -240, -241 are now being investigated. With 3/16-inch diameter fuel pins spaced on a 0.233-inch triangular lattice, the initial average heat flux is 5×10^5 Btu/hr-ft^2 with a maximum of 10^6 Btu/hr-ft^2. Cases involving flattening by redistribution of fuel are not being considered in these preliminary estimates. Multigroup diffusion theory calculations have resulted in a tentative plutonium oxide fueled fast core design of approximately 20 cm radius and 40 cm length containing 35 v/o fuel, 50 v/o coolant, and 15 v/o hardware. A 10 cm beryllium reflector has been assumed. Variation of the reflector thickness has shown that a reflector control span sufficient to accommodate a 20,000-hour endurance is attainable. Burnup calculations have been initiated for the above reference design with combinations of Pu-239 and Pu-240 fuel.

Potassium as a coolant is also being considered in addition to rubidium because of better knowledge of cross-section values and the expectation that heat transfer and hydraulic data on boiling potassium will be obtained at an earlier date from work being performed at other sites. Recently, NASA, BNL, ORNL, and GE Flight Propulsion Laboratory have agreed to concentrate their loop work on potassium in order to exchange and compare data. Little effort has been expended in determining the properties of rubidium, although it appears to offer advantages in certain temperature ranges. In order to achieve comparable pressures and specific volumes in potassium and rubidium, the reactor inlet and outlet temperatures for potassium are 770 C and 1117 C, respectively, versus 670 C and 1000 C for rubidium.

D. RADIATION EFFECTS ON METALS - 5000 PROGRAM

Single crystal and polycrystalline molybdenum containing interstitial and excess carbon as impurity are being studied to establish the combined effect of neutron irradiation and carbon impurity level on the properties of the metal. This program includes tensile tests, electrical resistivity measurements, deformation studies, electron microscope studies, x-ray diffraction analyses, microhardness tests, length change measurements, and some pre- and post-irradiation damage recovery studies.

A variety of specimens have been prepared, characterized, and encapsulated in a helium environment for neutron irradiation to three exposure levels, 10^{18}, 10^{19}, and 10^{20} nvt ($E > 1$ Mev). These specimens will augment data obtainable from specimens already irradiated, or currently under irradiation. The specimens are comprised of (1) 79 single crystal tensile specimens, ~ 1.5 inches long by 0.100 inch diameter in the reduced section; the lengths of eighteen of these specimens have been accurately measured and will be remeasured after irradiation, (2) 18 single crystal specimens each with a 110 surface, (3) 24 polycrystalline tensile specimens with 0.180-inch diameter gage sections, (4) 6 polycrystalline length change specimens which are two inches in length, and (5) 6 polycrystalline stored energy specimens. The single crystals included in groups (1) and (2) contain carbon as interstitial impurity in the ranges 10-20, 100-200, or 400-500 ppm. The material used in groups (3), (4) and (5), annealed at 1050 C for 16 hours, contained less than 250 ppm total impurity, of which 25 ppm is carbon.

Johnson-Mathey high purity polycrystalline foils, 0.003-inch thick, irradiated to $\sim 1 \times 10^{18}$ nvt, showed no change in their original microstructure. The irradiated foils are therefore being annealed at successively higher temperatures to increase the mobility of the defects and permit clustering. Two hours of annealing at 200, 300 and 400 C resulted in no increase in number of loops or change in dislocation networks present in either as-rolled or stress-relieved states. Foils will therefore be annealed at higher temperatures. The irradiated foils have been examined by x-ray diffraction.

The lattice parameter of the as-rolled material increased 0.023 percent, from 3.1471 A to 3.1478 A as a result of the irradiation. The line width increased very slightly from 0.510 to 0.520 degrees 2 θ. For the annealed material, the lattice parameter increased 0.016 percent to 3.1476 A. The line width was unchanged, namely 0.470 degrees 2 θ.

Single crystal specimens containing carbon have been chemically thinned to needle points, ~ 200 A diameter at the tip, for field ion microscope studies. Equipment is being assembled and patterns will be obtained shortly.

An unirradiated molybdenum single crystal (specimen number 15s4) has been tested in tension. An interesting feature of the resolved shear stress-glide strain curve was the absence of a third stage (parabolic hardening) of deformation. The carbon content of specimen 15s4 was 10 ppm.

The single crystal tensile specimen was examined by x-ray diffraction after testing. The original specimen had a [166] axis. At the point of failure the [111] direction had rotated until it was nearly parallel to the tensile axis. This is in accordance with the predicted behavior for a sample with this orientation. Slip apparently took place along one or both of the planes (211) and $(\bar{2}11)$, in the $[\bar{1}11]$ and [111]directions. The tensile axis rotated about $[01\bar{1}]$, which is the line of intersection of the above two planes. Failure occurred in such a way that a symmetrical chisel-edge parallel to $[01\bar{1}]$ formed during deformation and ultimate fracture. Testing of additional specimens is in progress.

E. CUSTOMER WORK

1. RADIOMETALLURGY EXAMINATIONS

The examination has been completed on the I&E failure from 4667 KE. The point of initial water entry could not be found (RM 444). The third low exposure hole failure from D Reactor is being examined. Visually the welds appear to be in good condition, but further tests will be made. A bump was discovered on the spire adjacent to the rupture. The bump appeared to have been caused during fabrication. Further examination will be required to determine whether this was the cause of the failure (RM 449).

A high exposure, Zr-2 clad I&E coextruded element is being examined for clues that might suggest the failure mechanism of the first Zr-2 clad I&E failure. Cause of failure has not yet been determined (RM 446).

No water entry sites were found in the spires of two dingot uranium fuel elements which failed by cracking transversely. (RM 437) The failure of an enriched, X-8001 clad I&E element in tube 2259 C was caused by localized corrosion which penetrated the cladding. Examination indicated the corrosion might have been caused by small bubbles of steam which formed on the surface of the element during a heat cycle in the reactor (RM 452).

Three sections were removed from an 1100 aluminum alloy process tube where an ultrasonic testing device indicated defects greater than 12 mils deep existed. Fluorescent penetrant dye test revealed that the defects existed at the points indicated. Metallurgical examination to determine the depths of the defects and integrity of the metal will be conducted (RM 453).

"Groove Corrosion" penetrated the cladding to the AlSi braze in two overbore elements which were irradiated in 4475 C, neither of the elements had ruptured (RM 455).

2. EQUIPMENT PROJECTS

CGH-857 Physical and Mechanical Properties Testing Cell

A number of difficulties have arisen in the fabrication of the cell castings. Errors in machining the cell casting have resulted in out-of-squareness, plug holes in the wrong locations, plug holes bored oversize, undercutting on square windows, etc. Some of these errors can be corrected, but it may be necessary to re-cast the south casting and the door casting. The delivery date for the castings of June 30 is now impossible.

CGH-858 High Level Utility Cell

The preparatory work before startup of the utility cell is about 40% complete. Numerous parts have been designed and are in various stages of fabrication by plant forces.

High Temperature Tensile Testing Machine E 75

The original drawings submitted by the vendor were unacceptable. The revised drawings have been approved with the exception of a few items.

Modification to Room Temperature Tensile Testing Cell

The floor penetrations have been completed and the modified base casting has been delivered to the building. The new cell top has not been completed.

Instron Tester (CGH-857)

All portions of this unit with the exception of the in-cell leads have been delivered to the 327 Building. Mr. A. E. Cozens of Instron will be here the second week in July to check the equipment and instruct the operating personnel in its use.

Micro Sampling Equipment

Mockup of the micro sampling equipment has been completed and final testing is in progress. Successful collection of the drillings from 0.020-inch diameter x 0.010-inch deep holes has been accomplished by using a vacuum to collect the UO_2 particles on a $\frac{1}{2}$ inch diameter 0.80 μ plastic filter disk. The weight of each collected sample was about 20 μg.

"A" Cell Measuring Equipment

Installation of the new diameter measuring equipment for "A" cell was completed and the unit was placed in operation. Time required for diameter measurements has been reduced about a factor of two and measurements are accurate within ± 0.001 inch.

3. METALLOGRAPHY LABORATORIES

The closure welds of three Inconel temperature detectors designed for service in the NPR were examined. The welds on all three detectors were exceptionally clean but appeared to have less than the optimum penetration. It was recommended that the specifications be checked to determine if adequate weld penetration was achieved by the vendor.

A metallographic examination was conducted to determine the origin of two small metal fragments which had become lodged in two of the front face venturi screens of the 105 F Reactor. The examination disclosed that one fragment was a portion of a stainless steel weld nugget that probably became dislodged from some upstream piping. The other fragment was a piece of carbon steel, the origin of which is in doubt.

An examination was conducted, using electron microscopy techniques, of the area adjacent to the fracture of Zr-2 tensile specimens fabricated from flat stock. The replica technique was used in an attempt to show if a suspected porosity gradient could be detected in the region of the tensile break. Presence of pores, observed with optical microscopy, was confirmed with the electron microscope. A pore size gradient, a function of distance from the break, was not confirmed, however, but varying pore diameters were observed. The majority of observed pores were in a region located at the extremity of the tensile fracture and extending along the sample for a distance of 0.050 inch.

Some work with replication technique was accomplished during the month. A 1:1 dilution of acetone with methyl ethyl ketone was found to result in a Faxfilm dissolution time four times longer than that required when using acetone alone. This mixture of solvents could be useful with replicas that have a tendency to break up due to the vigorous action of the more rapid solvent. Methyl ethyl ketone used alone caused the Faxfilm to swell and did not dissolve it during a sixty-hour immersion. Polyethylene will dissolve in warm toluene and benzene but on cooling to room temperature the plastic drops out of solution and is suspended as a slurry. Polystyrene was found to be soluble in carbon tetrachloride or benzene. Film prepared from liquid polystyrene required a room temperature drying time that was prohibitive for replica work. A ten weight percent solution of Dow Saran F-120 in methyl ethyl ketone remained elastic after an overnight dry at room temperature.

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

4. N-REACTOR CHARGING MACHINE

Modifications

Installation of the remote control panel assembly and connecting wiring is 70% complete. About 60 hours of electrical craft time were expended in additions and modifications to the control circuitry.

Fabrication of two of the four new limit switch assemblies is complete. These assemblies contain the limit switches which control the machine vertical travel when loading and unloading magazines. Two switches will be installed and tested before the remaining two are fabricated.

The foot pedals which control the rate of vertical lift and cross travel were modified to correct their tendency to stick.

The nozzles on the prototype process tube were removed after internal inspection revealed some abnormal scale. One new nozzle was installed on the process tube; the other end was left without a nozzle. During inspection of the new nozzle, some abnormal tool marks were observed, apparently caused by the rolling operation. This matter is being investigated further.

Testing

During the month two tests were conducted in an attempt to isolate the cause of scratching when fuel is charged into the process tube. The purpose of the first test was to determine if scratching could be the result of column action between fuel elements without any contribution from the charging machine or the nozzle assembly and attendant rolling depressions. This test consisted of charging 18 fuel elements by hand into the end of the process tube which had no nozzle on it. The fuel was pushed out the nozzleless end using push poles. A flow of 10 gpm of filtered water was used during both charging and discharging. The tube was then dried and inspected. Severe scratching of the tube was observed. Serious damage to the fuel element shoes was also apparent.

The purpose of the second test was to eliminate the effect of column action between elements. The test was performed by pulling one fuel element through the tube with a cord. The fuel element was inserted into the nozzle end of the tube using a gasket paper shim to prevent the fuel element shoes from contacting the nozzle during insertion. Each end of the tube was then plugged and the tube was filled with filtered water. The fuel element was pulled through the tube. Some scratching was observed in the tube and some damage was evident on the fuel element feet. Neither was as severe or abundant as in the previous test.

Previous testing of autoclaved fuel element support shoes against an autoclaved nozzle surface scratched the nozzle surface. The autoclaved condition of fuel element shoes is intended to represent their condition after reactor exposure. Thus, this scratching is representative of the fuel being discharged from the reactor through the outlet nozzle. Additional tests this month to confirm further the earlier results, resulted in only one very fine scratch on the newly autoclaved nozzle section. Some rather severe scratching was observed on the fuel element feet, however, The nozzle section has been re-autoclaved for further testing.

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PHYSICS AND INSTRUMENT RESEARCH AND DEVELOPMENT OPERATION

MONTHLY REPORT

JUNE 1962

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

Exponential Experiments for NPR

Final analysis of all previous experiments without control rods in the NPR mockup has been completed.

The method used in the analysis was to determine a set of extrapolation distances as described last month; these values were then used to obtain a final buckling for each experiment. A refinement has been made in obtaining the values for the extrapolation distances. The new values for the extrapolation distances to be used in analyzing diffusion length experiments are 2.39, 1.42, and 2.21 inches for side-to-side, front-to-rear, and vertical extrapolations respectively.

A summary of the experiments analyzed is shown in the following table:

Fuel Enrichment	Water in Coolant Channels Yes	No	Flooding	$B^2(10^{-6}cm^{-2})$*	Relaxation Length (cm)	Remarks
No fuel		X	-	-150 ± 4	42.02	No process tubes
No fuel		X	-	-335 ± 4	36.49	With process tubes
Natural		X	0	-105 ± 6	43.64	Averages of three measurements
Natural	X		0	- 95 ± 5	44.06	Averages of three measurements
.947% U^{235}		X	0	- 28 ± 5	50.49	
.947% U^{235}	X		0	+116 ± 5	57.36	
.947% U^{235}	X		1	+122 ± 5	57.92	
.947% U^{235}	X		2	+ 96 ± 5	55.58	

Flooding Code: 0 = No flooding
1 = Approximately equivalent to having a cylinder of water 41.5 mils thick surrounding the process tube.

2. = Flooding type 1 plus the equivalent of a layer of water approximately .29" thick lying uniformly in the horizontal steam vents.

* The errors include errors due to ± .25" uncertainty in all three extrapolation distances plus the error in fitting the vertical traverse to an exponential.

The diffusion lengths are:

1. Without process tubes   81.60 ± 1.04 cm
2. With process tubes   54.64 ± 0.36 cm

There are three experiments that are of lesser importance since they were performed using extra thin process tubes (3.055" O.D. x 2.879" I.D.) to simulate large flooding. The internal diameter of the NPR graphite process hole was reduced so that this large amount of flooding is no longer physically possible in the actual reactor. A summary is shown below:

Fuel Enrichment	Water in Coolant Channels Yes	No	Flooding	*Corrected $B^2(10^{-6}cm^{-2})$	*Corrected Relaxation Length (cm)
.947	X		3	134 ± 5	59.08
.947	X		4	119 ± 5	57.62
.947	X		5	124 ± 5	58.08

Flooding Code: 3 = thin process tubes
4 = thin process tubes plus type 1 flooding
5 = thin process tubes plus the equivalent of a layer of water approximately .15" thick lying uniformly in the horizontal steam vents

* Corrected for Aluminum removed from the internal diameter of the regular 3.055 x 2.700 process tube as quoted in the December 1961 monthly report. Errors quoted are the same as for the other experiments and do not include errors in the aluminum correction.

Measurements have been completed on the following experiments in the NPR mockup with wet enriched fuel elements. They are as follows:

1. A vertical traverse with six NPR control rods inserted and *polyethylene wrapped around process tubes to simulate flooding.

2. Vertical and horizontal traverses with zero and six NPR control rods inserted plus both *polyethylene wrapped process tubes and two pieces of masonite in each steam vent.

A new IBM 7090 code (BVTθCθ) which is described in the April 1962 monthly report was used in the analysis of the NPR mockup experiments without control rods.

This program was requested to eliminate a source of error caused by guessing the effective height of the exponential pile. An uncertainty of one-quarter inch in the effective height can cause an error in buckling of two to six microbucks for an eight foot pile. This program allows one to measure the effective height from vertical traverse data and to study the effects that lattice geometry and fuel variations have on extrapolation lengths.

Optimization of Retubed Lattices

Two horizontal layers in the "C" pile mockup contains holes for special horizontal flux traverses. The effect of these holes is to lower the bare counts in the region due to thermal leakage and to create a flux transient in the vertical direction similar to that caused by a reflector just above the region. Previous measurements will be repeated when the holes are plugged.

Computational Programming Services

Three exponential pile data codes - VTθCL, CθFIT2, and BVTθCθ - are now in production status. It is desirable from the standpoint of machine economy, and for ease of handling by the Data Processing Control Operation to reduce the three to one program deck. A code, ICEDT, is being prepared which will process intermixed cases previously handled by all three exponential codes.

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* Three and two-thirds layers of .010" sheet.

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A TRIP descriptive document (HW-72616) has been published and distributed.

PRθBC, the collision probabilities code has been completed.

Spatial Resonance Self-Shielding

Work has begun on a FORTRAN program which will compute spatial resonance self-shielding factors for group cross sections. Absorption cross sections in each energy group will be computed from the Breit-Wigner formula using either resolved or unresolved resonance parameters. The effect of Doppler broadening will be approximated by lowering the peak of each resonance to the Doppler value, and increasing the width to maintain the same area. The cross sections at discrete energies spanning each resonance will be self-shielded by interpolating in a table of monoenergetic flux depression factors, $g(\sigma_a)$, obtained from a suitable transport code such as P_3, S_N, or SHUSH-HFN, wherein all cross sections are held constant at an appropriate average value while the absorption cross section of the isotope to be self-shielded is allowed to take on values necessary to determine the flux depression factor over the entire range. Since the slowly varying component of the absorption cross section will, in general, have a non-unity g value, the resonance self-shielding factor is defined as

$$g_r = \frac{\int g(\sigma_a)\ \sigma_a(u)\ du}{\left[\int g(\sigma_a)\ du\right]\left[\int \sigma_a(u)\ du\right]},$$

where the integration is over specified energy bands.

S_x Consultation Service

Data input for the S_x transport code for four separate reactor systems is now in various stages of completion. Two of these systems are problems for Materials Development Operation concerning high energy neutron damage. The third problem is a fuel-moderator interface angular distribution study, and the fourth is a continuation of the critical mass problem for dissolving plutonium. An informal paper giving a simpler discussion of the input to the S_x code has been written. It is hoped that in the future this paper will enable the user to prepare the basic input data without requiring the help of Theoretical Physics, although the consultation service may continue on methods and applications.

Instrumentation

Tests continued on one prototype gamma spectrometer of twenty-four being fabricated for use in the NPR Fuel Failure Monitor. Tests on the count-rate-meter portion indicated that saturation effects occurred for input count rates of about three times full scale with a consequent meter indication reduction to less than full scale for a continued increase in count rate. The difficulty was caused by the use of an inadequate voltage breakdown transistor which was replaced by a different type to solve the problem. The high voltage supply and high voltage limit board, which had been returned to GE-APED for changes, were received and installed. Preliminary tests appear satisfactory.

Assistance was given to Electrical and Instrumentation Design, CE&UO for the preparation of the Gamma Energy Monitor portion of the acceptance test procedures for the NPR Fuel Failure Monitor System.

Vendor prints of the proposed intermediate range startup monitors for NPR were reviewed and comments were made regarding items not in accord with the original specification. The comments were forwarded to Instrumentation and Electrical Design, IPD.

At the request of Electrical and Instrumentation Design, CE&UO, investigations were started regarding possible methods of controlling the current output from the NPR neutron-sensitive ionization chambers. Calculations indicated the neutron flux incident on the chambers located in the top shield will be higher than necessary. This will make mandatory the use of neutron absorbers, or other approaches, to obtain optimum performance for the system. For the chambers located below the reactor, the calculations indicated the incident neutron flux level might be too low to obtain optimum performance; therefore, these chambers should have an increased sensitivity so that absorbers might be used for optimization. Other calculations indicated that radiation leakage through instrument openings into the flux monitor room will be within the design criteria and no shield plugs will be necessary.

Assistance was rendered to Facilities Engineering, IPD, regarding the possible use of titanium in control rods for K-Reactor. The problem involves establishment of purity specifications with consideration of activation and a metallurgical evaluation of physical changes due to the reactor environment.

Studies of pulse shape in organic scintillators were resumed. A three-dimensional family of curves was made for Stilbene. These curves are a plot of pulse height versus electron energy versus number of counts for

γ and neutron radiation. The resulting solid figure would resemble two steep ridges running approximately parallel, but converging at 0,0. One ridge corresponds to the γ (electron)--the other to the fast neutron (proton) detection. These plots will be compared with those of other materials to determine relative separation and convergence.

Systems Studies

Analysis of the rod position versus neutron flux data obtained earlier at 105-KW was started and is still in progress. Additional tests at 105-KW will be needed to verify the results of the first tests and to determine the effects of various flux configurations on the rod position versus flux attenuation characteristics. It appears that the simultaneous measurement of localized neutron flux power density spectra at various locations within the reactor core may yield useful information on reactor dynamics. A test using three or more of the neutron in-core monitors at 105-KW is being planned.

Published papers in the field of Dynamic Optimization of Control Systems usually deal with processes which are functions of a single space variable. A reactor, however, is not such a process, but is subject to spatial oscillations. A way was devised of extending current theories to cases of more than one space variable. Details are being worked out in a simplified system to verify the theory. This will provide a means for determining scientifically what interconnections there should be between controllers on adjacent control rods in a reactor, whether those controllers are automatic or manual.

One meeting of the NPR Failure Sequence Analysis Group was attended. The primary loop flow control system was discussed and assistance was given in formulating recommendations to be included in a forthcoming document.

The 11-node reactor instrumentation analog simulation was programmed on the EASE and GEDA analog computers toward the end of May. The purpose of the simulation is to determine proper instrument settings for the protection of the reactors against nuclear excursions at various power levels. The instruments under investigation are those which monitor flux, reactor outlet temperature, and linear rate of rise of outlet temperature. The problem has been running accurately; reproducibility has been consistently excellent. The first few runs were made for comparison with the results of the previous instrumentation study which was made using a six-node reactor model. The two models exhibit appreciably different behavior, indicating that a significant increase in accuracy is obtained by going to the 11-node model.

A review was made of the proposed Acceptance Test Procedure for the NPR Integrated Data and Temperature Monitor System. Comments are being forwarded to NPR Project Section. It appears that the entire area of interference among different sections of the system has been omitted from the procedure. The electrical distribution system was analyzed with regard to the effect of power failure on the operation of the steam generator by-pass valves.

SEPARATIONS

Experiments with Plutonium Solutions

Criticality experiments were continued with plutonium nitrate solutions in a 14-inch diameterstainless steel sphere, fully reflected with water. Plutonium concentrations were in the range of ~ 39-46 g Pu/l at an acid molarity of ~ four.

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

The chemical analyses of the solutions used in the current experiments have not been completed; until the final results of the analyses are received, comparisons can be made with the earlier data only in a qualitative way.

The preliminary results of these measurements, however, indicate agreement with the early P-11 experiments, and the critical mass values appear to correspond roughly to those in which a four-inch concrete reflector was used; a concrete reflector of four-inch thickness is then the equivalent of an effectively infinite water reflector.

Further difficulties were encountered with leakage of Pu solution back through air operated valves. The safety rod system also became inoperative because of a broken stretch wire which resulted in loss of current to the magnet coil; a new stretch wire was installed and this unit now operates correctly.

A new spherical critical assembly vessel of 11.6-inch diameter was received during the month. This vessel was especially designed with an off-center mounting of the control and safety rods, and contains provisions for a re-entrant tube to facilitate flux measurements along the axial diameter of the unit. It will thus be possible to determine buckling values, and to make other measurements, such as the spectral index (effective neutron temperature) from foil activations within the sphere.

Measurement of k_∞ in the PCTR for Dilute Solutions

An experiment to study the effect of stainless steel (as used in containment vessels for solutions) on the measured values of k_∞ in the PCTR was begun during the latter part of June. The experiment is being conducted with dilute uranyl fluoride (UO_2F_2) solution.

Mass Spectrometry

The mass spectrometer for this program provided isotopic analyses on samples for the Isotopic Analysis Program during the first three weeks of the month. The resolution of this instrument became progressively worse during the month and it is now shut down. A thorough examination of the analyzer alignment of the spectrometer is in progress.

Recuplex Criticality Incident - Calculated Values for Criticality from Multi-Group Diffusion Theory

As a further aid in the post analysis of the Recuplex criticality incident, calculated values for criticality in vessel K-9 were determined from the 9-Zoom and the HFN codes. The critical radii of bare spheres were calculated for $Pu(NO_3)_4$ solutions with nitric acid molarities of one and two

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TABLE I

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

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

Equipment Number	Date	Reflector	Pu Concentration (g/l)	Acid Molarity	Sp. Gr.	H_2O (g/l)	Total NO (g/l)	H/Pu Atomic Ratio	Critical Volume (liters)	Critical Mass (Kg Pu)
1142090	6-5-62	Water	46.0	4.22	1.207	840	306	-	21.1 +.02 −.04	0.97
1142091	6-7-62	Water	42.1	4.22	1.203	-	300*	-	22.0 +.06 −.08	0.9[illegible]
1142092	6-8-62	Water	38.0	4.22	1.201	-	303*	-	23.4 +.09 −.13	0.89
1142093	6-11-62	Water	39.2	4.20	1.198	-	313*	-	23.1 +.07 −.09	0.91
1142094	6-12-62	Water	39.2	4.20	1.198	-	313*	-	23.1 +.05 −.06	0.91
1142095	6-14-62	Water plus 0.036" S.S.	38.8	4.14	1.194	-	313*	-	23.7 +.12 −.19	0.92

* Chemical analysis incomplete and subject to change.

[redacted]

[redacted]

in order to cover the possible cases of interest; the acid molarity of the solution in K-9 was ~ 1.0 and the total nitrate concentration was ~ 120 g NO_3/l. The effect of Pu^{240} was also accounted for in these calculations.

The critical radii of the bare spheres were used to obtain buckling values as a function of the solution concentration. These values were then in turn used to compute the critical solution height as a function of Pu concentration in vessel K-9, through use of the following relationship:

$$\text{Material Buckling} = \frac{(2.4048)^2}{(R_c + \lambda_r)^2} + \frac{\pi^2}{(h_c + \lambda_o + \lambda_b)^2},$$

with,

$$R_c = 22.07 \text{ cm}$$
$$\lambda_r = 3.0 \text{ cm}$$
$$\lambda_o = 2.2 \text{ cm}$$
$$\lambda_b = 3.0 \text{ cm}$$

The extrapolation length was assumed to be 2.2 cm + the thickness of the vessel wall (0.8 cm).

Estimates of criticality were also made on the basis of an aqueous reflector on top of the plutonium solution (for this case $\lambda_o = 6$ cm). The 6 cm value would be obtained for a solution reflector of $\sim$3-4 inches thickness.

With the system unreflected on top and for a Pu concentration of 35 g/l with one molar nitric solution, the critical volume is calculated as ~ 44 liters. This value is in fair agreement with the measured value of 43 ± 1 liters.

Although the calculated critical volume ($\sim$44 liters) is in qualitative agreement with the volume estimated from measurement there are some apparent discrepancies elsewhere. The minimum critical mass is calculated as $\sim$ 1.38 Kg, which is slightly too large, since 1.33 Kg was in fact actually critical in K-9. From the calculations the critical volume (for minimum mass) would be $\sim$ 55 liters of solution. This would tend to imply that the system did not go through the minimum mass value on evaporation and subsequent shutdown. This assumption is predicated on the basis of the "high water" marks - since none were observed as high as 55 liters (highest mark at $\sim$46 liters). The solution was slightly on the over concentrated side of the minimum in the mass curve at the beginning of criticality - if the concentration were uniform to start with.

Because of the uncertainty in the measured critical volume, further attempts to obtain better agreement between the computed and measured value would probably not produce significant results. However, if we assume the maximum difference in the error uncertainty between the calculated and measured values (in this case ~2 liters) and then reduce the minimum value by a proportionate amount, since the calculated value is too high, the minimum mass in K-9 becomes ~1310 g Pu for a concentration of ~25 g Pu/l; the critical volume would then be ~52 liters for minimum mass.

On this basis the system would also have been delayed critical at a volume of ~48 liters with a Pu concentration of 27.5 g Pu/l, and supercritical throughout the volume range up to ~55 liters.

Let us now make an estimate on the assumption that water was evaporated from the solution increasing the nitrate concentration to ~120 g/l at 38 liters - starting with 48 liters. Correcting for the lower nitrate concentration at the higher volume would reduce the volume by < 1 liter to ~47 liters.

The calculations with a top reflector (layer of solution 3-4 inches in thickness) indicate that criticality could also be obtained with a volume of 38 liters for the concentration of 35 g Pu/l. In other words, the solution in the vessel at the time of draining could have been made critical by adding a hydrogenous reflector to the top of the solution as it existed.

Some estimated criticality parameters for vessel K-9 are summarized in the following table.

SUMMARY OF ESTIMATED CRITICAL PARAMETERS FOR K-9

Estimates Based on Calculations

Minimum Critical Mass	~	1310 g Pu
Minimum Critical Concentration	~	25 g Pu/l
(Critical Volume)	~	52 liters

Critical Mass ~ 1.54 Kg at 35 g Pu/l
Critical Volume ~ 44 liters at 35 g Pu/l

Volume at Delayed Critical ~ 48 liters at 27.5 g Pu/l
Supercritical for Volumes > 48 liters and < 55 liters

Critical Mass (With Top Reflector) ~ 1.33 Kg at 35 g Pu/l
Critical Volume (With Top Reflector) ~ 38 liters at 35 g Pu/l

Estimates from Measurements

Critical Mass ~ 1.51 Kg at 35 g Pu/l
Critical Volume ~ 43 ± 1 liter at 35 g Pu/l

Nuclear Safety in HLO

Four nuclear safety specifications were approved and issued--three for Critical Mass Physics and one for the Plutonium Metallurgy Operation. The titles of these are as follows:

C-9 (Supplement A) Rules for Make-up, Handling, and Transporting 20 g/l UO_2F_2 (93% U^{235}) Solution in k_∞ Measurement Vessels.

C-12 (Rev.) Rules for Storage and Transporting of 93% U^{235} Enriched Uranium Solutions

C-13 Rules for the Storage, Handling, and Transporting of 2.1 and 2.3% U^{235} Enriched Uranyl Hexahydrate (UNH)

J-3 Rules for the Fabrication of Plutonium Carbide

Nuclear Safety in CPD

Participation as a member of the Recuplex and Project 880 hazards analysis groups continued throughout the month. Seven procedures pertaining to the deactivation of Recuplex were reviewed.

Nuclear Safety in FPD

Two temporary nuclear safety specifications were reviewed and approved for the Engineering Operation. The titles are as follows:

Storage and Shipment of Uranium Oxide Scrap. This specification applies to 0.96% U^{235} enriched uranium oxide generated in the scrap metal burning operation.

Uranium Transportation Offsite. This specification applies to the transporting of small quantities of fissile materials by common carrier.

Nuclear Safety in Transportation

The shipment of 90 Kg of 2.3% U^{235} enriched uranium in the form of UNH powder to Oak Ridge was reviewed and approved(1). This uranium was shipped by common carrier in four approved birdcages.

Nuclear Safety - Training and Education

Talks on the subject of nuclear safety were given to the following three groups: Chemical Development, Plutonium Metallurgy, and Plutonium Fuels Development.

Instrumentation and Systems Studies

The radioactive waste disposal heat transfer, a study concerned with the determination of the maximum sphere of influence of the heat produced by a buried cylinder of radioactive waste, was continued. Difficulties have been encountered in designing a satisfactory simulation. These difficulties are associated with the given boundary conditions. A satisfactory simulation has not yet been devised. Several forms for an analytical solution were derived.

The selsyn driving gear reduction unit for the control rod drive at the Critical Mass Laboratory was fabricated and assembled. All gears and clutches were shaft-keyed to eliminate slippage.

The newly installed instrument lines for the C-Column Control experiments were tested for noise and found to have a spurious signal level well below the minimum for signal interference in the data logging system. One line appeared to be noisy, but investigation showed the signal degradation to be caused by a faulty turbine flow meter transducer.

NEUTRON CROSS SECTION PROGRAM

Quasi-Elastic Scattering of Neutrons from Water

The series of measurements of quasi-elastic neutron scattering from water has been essentially completed. These measurements were made with improved energy resolution at a neutron energy of 0.15 ev from room temperature water. Measurements were made at two scattering angles for samples of different water thickness and for the vanadium standard. These measurements were significantly delayed by reactor maintenance work. Analysis of the results of the measurements is in progress.

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(1) Letter from P. F. Gast to F. J. Zelley, "Nuclear Safety Approval for 2.3% U^{235} Enriched UNH Shipment," June 26, 1962.

Inelastic Scattering of Neutrons from Water

The computer program S-X calculations of multiple collision and beam attenuation corrections for the scattering from the vanadium standard have been completed and applied to these measurements. Estimates have also been made of the much more difficult case of inelastic neutron scattering from water. These corrections are indicated to be as large as 15 percent. Because the problem which is being calculated is not a good representation of the actual physical problem the values obtained for the corrections are not very reliable and will not be applied to the data.

Rotating-Crystal Spectrometer

A rotating-crystal slow neutron spectrometer system to perform neutron scattering measurements is being studied. The 1024 channel time-of-flight analyzer for energy analysis and data storage has been essentially completed by Instrument Research and Development. The general features of the spectrometer system are presently being studied preliminary to detailed design of the system components. These considerations include the lengths of the incident and scattered neutron beam flight paths and the number and disposition of neutron detectors and the relative motions required of the various components.

Experimental studies are in progress of the feasibility of using scintillation counters for neutron detectors. A Li I detector enriched in Li^6 offers the potential advantage of high neutron detection efficiency with small timing uncertainty compared to a BF_3 system. Experiments are in progress to attempt to determine if a system of two or more photomultipliers viewing a thin Li I scintillator can be operated in coincidence and shielded to give a suitably low background of multiplier noise and gamma sensitivity while retaining the desired characteristics.

Np^{237}(n,2n) Cross Sections

The (n,2n) reaction on Np^{237} leads to the nucleus Pu^{236} which would be an attractive energy source because of its 2.85 yr alpha half life. At the request of IPD an evaluation was made of the Np^{237}(n,2n) cross section for which only a single measurement at 14.5 Mev has been reported. The (n,2n) cross section as a function of neutron energy E_n(Mev) is given approximately by

$$\sigma_{n,2n} = \sigma_n^* \left[1 - (1 + E_n - E_T) \exp - (E_n - E_T) \right]$$

where $E_T = (6.91 \pm 0.16)$ Mev is the calculated reaction threshold energy for Np^{237}. The quantity σ_n^* was evaluated from optical model calculations of the non-elastic cross sections of neighboring nuclei, extrapolations of the measured fission cross section, and estimates of the effects of inelastic scattering. A value of $\sigma_n^* = (0.48 {+0.09 \atop -0.29})$ barn was obtained for Np^{237} for $E_n < 13$ Mev. For $E_n \gtrsim 13$ Mev the value of $\sigma_{n,2n}$ is expected to decrease because of competition from the onset of (n,2n'fiss) and (n,3n) reactions. This analysis is consistent with the single measurement of $\sigma_{n,2n} = (0.39 \pm 0.07)$ barn at 14.5 Mev.

Fast Neutron Cross Sections

A series of total cross section measurements using the $Li^7(d,n)$ reaction neutron spectrum and time-of-flight were performed during the month. Measurements were made on samples of Bi, In, Cu, Na, and Li. The data on bismuth were marred by instrument failure and may not be adequate. The series of measurements was interrupted during the measurements on Li because of the failure of the accelerating column of the Van de Graaff.

Several changes in the time-of-flight system were incorporated in the above measurements. A major change was the use of a quadrature deflection system so that only one deuteron burst was produced per RF cycle. The time analyzer channel widths were reduced to 0.7 nsec and unchopped beam currents of 25 to 30 µamp were successfully employed.

Work continued on the fabrication and satisfactory encasement of total cross section samples of strontium, barium, and calcium. Work is also in progress on the fabrication of a new 2 inch diameter scintillation counter assembly for possible application as a detector of the arrival time of the deuteron beam on the neutron-producing target.

Work continues on the use of the data-reduction program BIGNED to handle the total cross section data. Problems have been encountered in developing a criterion for adjusting the data to the proper time zero scale.

Instrumentation

The final stages of debugging the 1024-channel time-of-flight analyzer was completed. At the present time the analyzer is able to accept simulated time-of-flight pulses and store them in memory. The stored data can then be displayed on a scope, or printed out. Data exchange between memory sections including adding, subtracting or erasing, is also possible. Long term tests of accuracy and stability are planned next.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE

Graphite Lattice Parameters for Low Exposure Pu-Al Fuel

Attempts to analyze the low exposure, plutonium-aluminum fueled graphite lattices using Program S have yet to be successful. Extensive calculations have been done on the $10\frac{1}{2}$" poisoned lattice. The best value of k_∞ calculated to date is 0.91. There is reason to believe, however, that the fully converged value of k_∞ will be somewhat higher. Originally, a 1 milli-k convergence criterion was used and several runs were made that had converged within this limit. Indications were, however, that the rate of convergence at this stage was approximately 1/2 milli-k per iteration. Consequently the convergence criterion was changed to 1/10 milli-k. A fully converged case has not been obtained as yet after 22 iterations. It was found also that performing a double S_3 calculation improved the calculated value of k_∞ by about 50 milli-k over a double S_2.

Effective Resonance Integral of Pu^{240}

The analysis of the experiment to measure the effective resonance integral of Pu^{240} relative to the dilute resonance integral for a series of Pu-Al rods is nearly complete.

The tentative results for the effective resonance integrals for the Pu-Al rods have an uncertainty of five to ten percent. This results from an uncertainty of about the same size in the sensitivity of the PCTR to neutron absorption in Pu^{240}.

The first draft of a summary of this work has been prepared for submission to the American Nuclear Society for the winter meeting.

The Critical Facility

The HOO-AEC has requested additional hazards analyses of specific fuel loadings which are to be measured in the PRCF. Assistance has been given in planning calculations which will be run on the analog computer. In addition, a series of reactor calculations has been started on various PRCF loadings using the IBM program SWAP. This work is, in part, in support of the hazards evaluation work on the quasi-uniform loading (Pu-Al-UO_2 supercell loading). In this work typical PRCF cells are being used rather than typical PRTR cells used in some earlier analyses. To date it has been shown that the latter changes have increased cell k_∞'s by 110 mk and 160 mk for UO_2 and Pu-Al cells, respectively. The effect on K_{eff} for various fuel loadings in the PRCF has not been evaluated as yet. Preliminary calculations have been made on the quasi-uniform loading with control and safety rods in their "normal" positions (212, 812, 503, 515, 806, 206).

The final drafts of the PRCF startup procedures are 95% complete and are being prepared for distribution, acceptance, and approvals. The experimental procedure for the reflector experiment remains to be written. With respect to the latter, paraffin has been ordered by PRCFO to provide a six-inch-thick by nine-foot-high reflector. The paraffin is to be commercial grade with melting point classification of 140 to 142°C. The blocks are to be 2 inches by 12 inches x 18 inches. This reflector size is approximately equivalent to five inches of water, which is effectively infinite, and will therefore satisfy the requirements for hazards evaluation of reactor cell flooding.

Process specifications which will govern the operation of the PRCF have been reviewed as well as detailed procedures which will be used for the startup experiments.

The two flux traversing rigs have arrived and are being debugged.

Six Hanford 1/2" aluminum jacketed BF_3 tubes were received. At present the tubes are being calibrated using a RaBe source in the standard pile. These pulses are being recorded by the 256-channel analyzer in the form of differential and integral plots of pulse count vs. bias voltage for a given time interval. Four of these tubes are to be used in the power level calibration of the PRCF by measuring the absolute value of the neutron flux. The two remaining tubes are to be used as spares for this project.

Measurements on Phoenix Fuel Standards

The reactivity measurements necessary to determine boron and plutonium contents have been completed. These data from the PCTR are now being analyzed.

Several fuel loadings were tried in order to maximize the sensitivity of the reactor and still maintain a highly thermal spectrum in the test position. When the driver fuel was moved from a circle at 67 cm radius to one of 49 cm radius, the cadmium ratio of gold (0.005") changed from 43 to 20. The worth of Cu increased from 13.1 g/¢ to 11.3 g/¢. An attempt to increase the reactor sensitivity further by putting lucite bars into the center of each of the driver fuel elements failed. The worth of Cu decreased to 12.1 g/¢.

To measure the boron content the worths of the Phoenix Fuel Standards and copper were measured in the center of the graphite core with the fuel loaded in the 49 cm radius circle.

To measure the Pu content of the Phoenix Standards the relative worths of the thermal and fast neutrons were varied. In graphite the relative worth of fast to slow neutrons was 0.8. With a 1" thick water jacket the relative worth was 1.4. With a 2" thick water jacket it remained at 1.4. With the 1" jacket with .001 g/cc boron dissolved in the H_2O, the relative worth was 2.5. The reactivity coefficients of all samples were then measured inside the 1" water jacket with the .001 g/cc boron.

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

Data from exponential experiments and from approach-to-critical experiments with 1.8 w/o Pu-Al fuel in H_2O have been analyzed to yield values for λ, the extrapolation distance plus the reflector savings. This was accomplished by assuming λ was equal for both the radial and axial direction and equating the buckling for exponential experiment to that from the critical experiment. The values are tabulated in Table I. The average 3.14 in. or 7.97 cm is not much different from the value 7.7 cm which was assumed in previous calculations.

TABLE I

VALUES OF λ, THE EXTRAPOLATION DISTANCE PLUS THE REFLECTOR SAVINGS

Lattice Spacing (in.)	λ (in.)
0.75	3.50
0.80	3.25
0.85	2.99
0.90	3.14
0.95	2.80

Results of chemical and isotopic analyses of the Pu-Al have been compiled for each of the loadings. Information about the history of the various alloys has been compiled also. This information will make it possible to obtain more analyses if they are needed.

Experiments which will use zoned loadings have been planned. First the critical loading will be determined for $\sim$ 209 rods of which 6 w/o of the Pu is Pu^{240} surrounded by rods which contain 5 w/o Pu^{240}. Secondly the critical loading will be found for the same number of rods of 5 w/o Pu^{240} surrounded by 6 w/o Pu^{240}. The lattice framework and instrumentation has been checked out for these 2 zone loadings.

Technical Shops is preparing an estimate for a set of templates with 0.66 inch lattice spacing and 1027 holes for the 16 w/o Pu^{240} fuel rods. The diameter of these holes is not large enough for plastic sleeves around the fuel rods. This is likely to be the first job on the automatic drilling machine just received by Technical Shops.

PRTR Fuel Irradiation Experiments

Data from lutetium foils which were irradiated in the PRTR are being analyzed. Analyses of the data which were taken in order to normalize counts at various distances from the NaI(Tl) crystals have been completed. The remaining data have been prepared for analyses with the program LULU.

The results of an eleven group HFN calculation of the slowing down flux, in a 1.8 w/o Pu-Al fuel element, between 0.18 ev and 0.68 ev have been fit by a polynomial equation. This equation will be used as a subroutine in the program ACE and the resonance integrals for lutetium between 0.18 ev and 0.68 ev will be calculated. The resonance integrals in turn will be used to correct the data from the lutetium foils.

Heterogeneous Pu-Al-H_2O Reactor Physics Calculations

A three-group, one-dimensional diffusion theory analysis of heterogeneous 1.82 w/o Pu-Al rods in H_2O is completed. A comparison of theoretical estimates of critical mass to experimentally determined values has been made. The critical systems were composed of Pu-Al alloy rods, 44 inches long, 0.500 inch in diameter, immersed in a tank of light water with infinite reflection on all sides. The alloy contained 1.82 w/o plutonium which had an effective average composition of 94.37% Pu-238, and 5.63% Pu-240. The alloy is clad in 0.030 inch thick Zircaloy-2 with an O.D. of 0.566 inch.

The computation was done with SWAP code on the IBM-7090. The neutron cross sections, temperature and spectrum corrections, and axial flux extrapolation distances were all determined in a manner described in the April and May Monthly Reports.

Calculations of the critical number of rods were made for lattice spacings in the range 0.75-0.95 inch with effective Pu-240 concentrations of 5.1, 5.63, and 6.0 w/o. The results are compared in Figure I to the measured values for the 5.63 w/o Pu-240 systems. The systems depicted in Figure I are all uniform loadings. Table I lists the values of effective multiplication calculated for measured critical systems of 5.63 w/o Pu-240 rods.

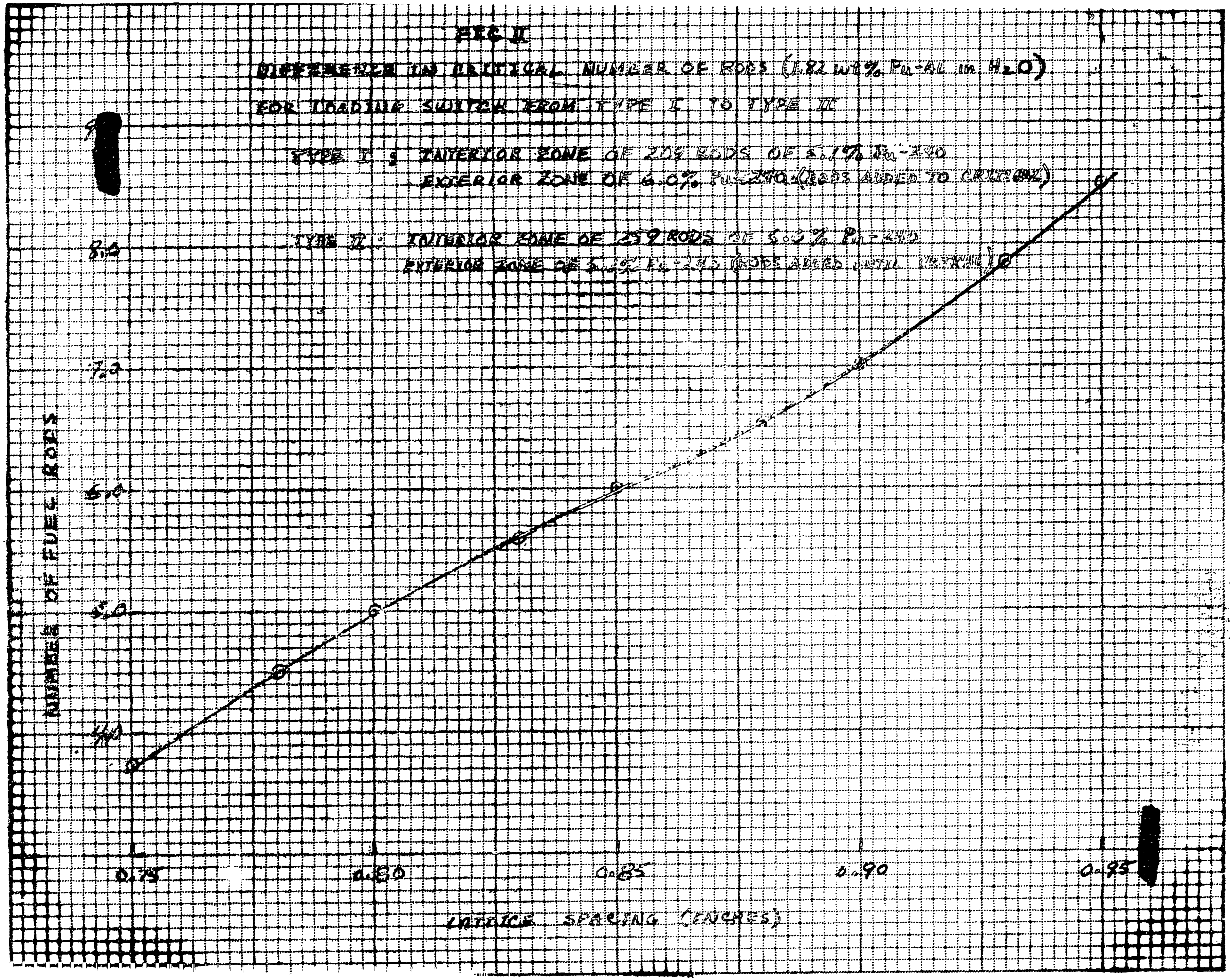
FIG II
DIFFERENCE IN CRITICAL NUMBER OF RODS (1.82 wt% Pu-Al in H_2O)
FOR LOADING SWITCH FROM TYPE I TO TYPE II
TYPE I: INTERIOR ZONE OF 259 RODS OF 5.1% Pu-240
EXTERIOR ZONE OF 6.0% Pu-240 (RODS ADDED TO CRITICAL)
TYPE II: INTERIOR ZONE OF 259 RODS OF 5.0% Pu-240
EXTERIOR ZONE OF 5.1% Pu-240 (RODS ADDED UNTIL CRITICAL)
[redacted]
8.0
7.0
6.0
5.0
4.0
NUMBER OF FUEL RODS
0.75
0.80
0.85
0.90
0.95
[redacted]
LATTICE SPACING (INCHES)

TABLE I

Lattice Spacing	Measured Critical No. of Rods	Calculated k_{eff}
0.75	563	1.0021
0.78	526*	0.9985
0.80	510	0.9966
0.83	496*	0.9938
0.85	494	0.9927
0.88	502*	0.9920
0.90	516	0.9921
0.93	550*	0.9942
0.95	578	0.9941

* Interpolation from curve.

The values presented in Figure II anticipate experimental determinations with two-zone loadings of rods with 5.1 and 6.0 w/o Pu-240. For a given lattice spacing, the loading switch yields a lattice with a different proportion of each rod type. Also, the Pu-240 concentration difference is enhanced by the change in statistical weight of the position occupied by the switched zones.

Status of PRTR Fuel Irradiation Experiments

Low Exposure Pu-Al Elements (PRTR Test 13)

Element 5093 in location 1544 was discharged to the basin on June 4, 1962, having accumulated 56.1 MWD exposure.

Mark I UO_2 Elements (PRTR Test 37)

The six UO_2 physics test elements were installed on May 12, 1962, as indicated below:

Fuel Element	Location
1096	1449
1097	1649
1098	1552
1099	1548
1100	1651
1101	1451

[redacted]

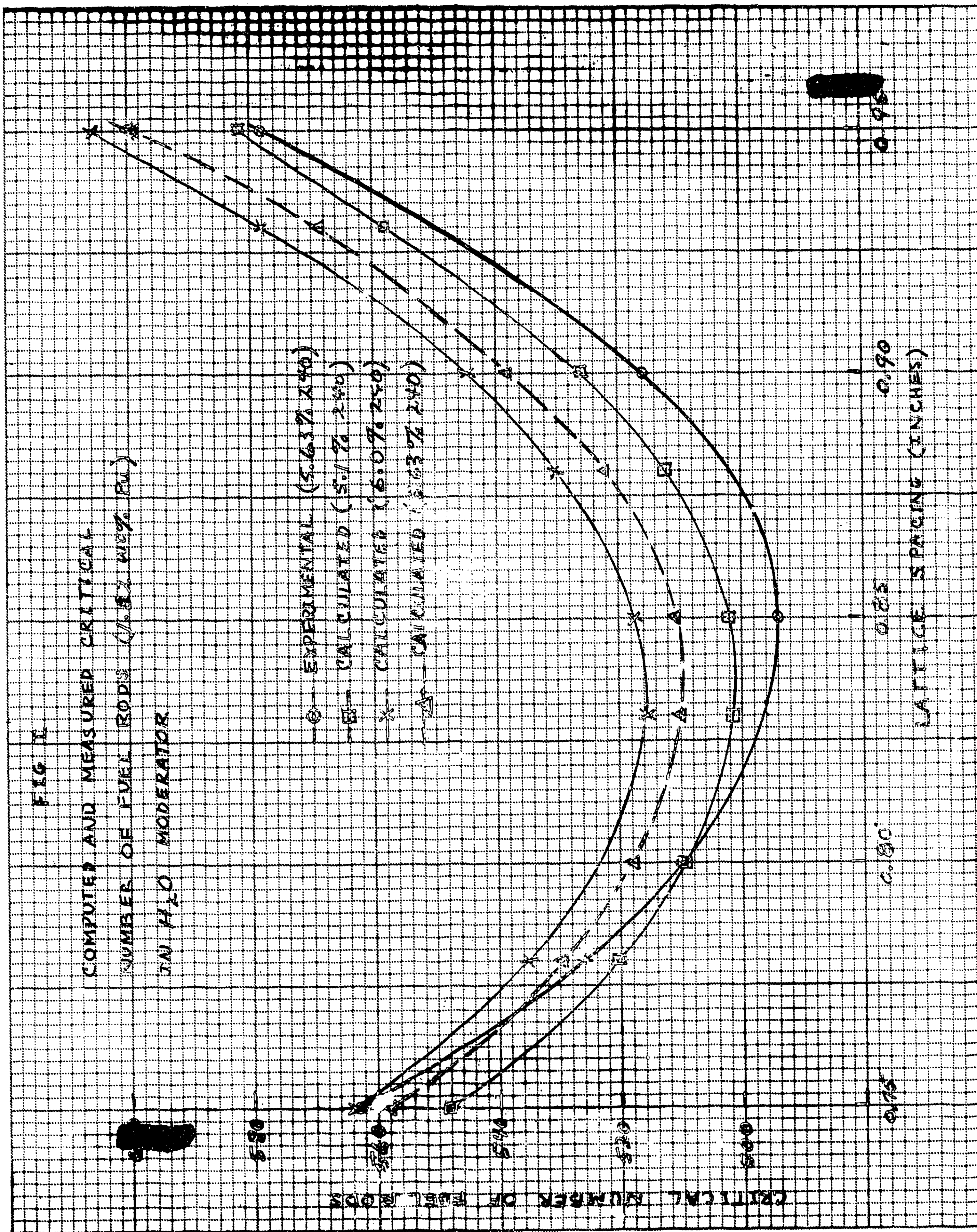
FIG II
COMPUTED AND MEASURED CRITICAL
NUMBER OF FUEL RODS
IN H2O MODERATOR
EXPERIMENTAL (5.63% 240)
CALCULATED (5.1% 240)
CALCULATED (6.0% 240)
CALCULATED (5.63% 240)
LATTICE SPACING (INCHES)
0.75
0.80
0.85
0.90
0.95
CRITICAL NUMBER OF FUEL RODS
500
520
540
560
580

Element 5075

Analysis of PRTR Pu-Al fuel element 5075, which was exposed to 49.5 MWD, has been completed by Analytical Laboratories. Comparisons have been made of burnup in 13 samples by coulometric titration of plutonium, plutonium alpha counting, Cs-137 analysis, and mass spectrometer data. Good agreement between the Cs-137 and mass spectrometer data, the latter being interpreted by MELEAGER burnup calculations, is obtained if the Cs-137 yield of plutonium is taken as 0.0568, and the half life as 28.4 years(2) or 0.0583 if the half life is 29.15 years(3). The distributions of burnup radially and longitudinally as determined by the Cs-137 and mass spectrometer data from the 13 samples are shaped as expected; however, the coulometric and alpha data, while in fair agreement between themselves, are in poor agreement with the former methods. Since the sample solutions were not optically clear, the possibility exists that a true solution of plutonium was not obtained by the reagents used at Radiometallurgy.

PRTR Theory-Experiment Correlation

To accurately predict the power distributions in the PRTR, an effort is under way to derive a meaningful model of the PRTR. This derived model will be used in trying to match the power distributions in the experimental "Banked Shim Rod Test."

A consistent set of multi-group cross sections for use in either a transport or diffusion analysis is being compiled. The cross sections pertinent to the PRTR are currently being updated in the RBU Basic Library. The program "Barns"(4) will be used to obtain 1/E weighted average cross sections, and these average cross sections to be used in updating the GAM-1(5) nuclear data tape. Also, the program TEMPEST(6) will be updated

(2) Rider, B. F., J. H. Russell, Jr., D. W. Harris, and J. P. Peterson, Jr., "The Determination of Uranium Burnup in MWD/T," GEAP-3373, March 17, 1960.
(3) Rider, B. F., J. P. Peterson, Jr., and C. P. Ruiz, "Half-Life of Cs-137," Transactions of the American Nuclear Society, 5, No. 1, p. 196 (1962).
(4) Schlosser, J. E., "Barns - A Program to Obtain Cross Sections from the RBU Basic Library," HW-72117, December 27, 1961.
(5) Joanow, G. D. and J. S. Dudek, "GAM-1 - A Consistent Pl Multi-Group Code for the Calculation of Fast Neutron Spectra and Multi-Group Constants," GA-1850, June 28, 1961.
(6) Shudde, R. H. and J. Dyer, "TEMPEST - A Neutron Thermalization Code," NAA Program Description, September, 1960.

[redacted]

by the point cross sections contained in the RBU Library. From the combination of these three programs, a basic set of cross sections for the PRTR will be compiled.

Several methods to obtain a cylindrical model of the PRTR 19-rod cluster have been preliminarily investigated using IDIOT(7). From these preliminary effects, it seems advisable to investigate these methods further. A two-dimensional analysis (x-y) of a UO_2 - 19-rod cluster would aid in inferring the best method for cylindricizing, for the radial flux, and reactivity could be compared to the experimental results(8) for this cell. The three methods investigated are, briefly: (1) conserving volume of materials in the cluster, (2) conserving radius to and thickness of cladding and fuel, (3) same as (2) except for homogenizing the fuel, cladding, and moderator contained between the inner and outer radii of the annulus.

Advanced Concepts - Critical Mass Calculations for Rubidium Cooled Cores

Rubidium's thermodynamic properties make it interesting as a reactor coolant for high temperature, direct cycle power systems. Such systems might be useful for application as space power units.

A series of critical mass calculations for fast spectrum, rubidium cooled cores have been performed. The HFN multi-group diffusion theory code for the IBM-7090 was used for the calculations. The Yiftah, Okrent, and Moldauer (YOM) 16-group structure was used with the cross section data assembled from several sources. The Pu-239, Pu-240, U-235, O, and Ta cross sections were taken from Fast Reactor Cross Sections by Yiftah, Okrent, and Moldauer. The cross sections for N and Be were taken from the GAM code using the P_1 option and the YOM weighting flux and group structure. The rubidium cross sections were from the RBU Basic Library using the YOM weighting flux. Only transfers to the next lower group were allowed. The rubidium transfer cross section was assumed to be $\xi \Sigma_s/\Delta u$. Three types of fuel were considered in this study: PuO_2, UO_2, and PuN. The coolant volume fraction was used as a parameter and criticality was obtained by adjusting the fuel and structure (Ta) volume fractions. All calculations were made using spherical geometry for the core surrounded by a 10 cm Be reflector.

(7) Richey, C. R., "IDIOT - A Lattice Parameter Code for the IBM-7090," HW-63411, January 7, 1960.
(8) Lilley, J. R., "Correlation of Data on Heavy Water Moderated Cluster Lattices," HW-60275, May 19, 1959.

The required critical fuel volume fraction for PuN was found to be the least for the three fuels considered, while it was greatest for U-235 O_2. The ratio of critical fuel volume fractions appears to be reasonably constant over a wide range of core radii (10 cm - 70 cm). For the three fuels considered, the approximate fuel volume ratios are:

$$PuN : PuO_2 : UO_2 = 1 : \sim 1.5 : \sim 2.0$$

The results of the calculations should be used with caution, particularly for the smaller radii range where the validity of diffusion theory is questionable.

Code Development

CALX

SIGMA-3, the program which prepares a cross section tape for use in the CALX multi-group cell burnup code, is operating satisfactorily. The "merging library" which SIGMA-3 uses to correlate GAM and TEMPEST results is complete, but has not been checked in detail. Some minor bugs remain in the multi-group flux and reactivity calculation.

Coding of the burnup portion of CALX has revealed some unanticipated problems. Most arise because CALX differs considerably, in both concept and implementation, from its predecessor, MELEAGER. This means that the existing MELEAGER decks must be so extensively modified, in writing CALX, that it has become impossible to maintain the absolute input/output compatibility which Programming Operation requested. Present attempts are aimed at preserving as much input/output compatibility as possible. This will yield a code (called CALX-I) which will read a fair amount of unused data, but will minimize the number of changes required in the Programming Operation economics codes when they are converted from using MELEAGER to using CALX. A future version (CALX-II) will have more sensible input and output formats.

All foreseeable changes in converting the MELEAGER main program and Runge-Kutta numerical integration subroutine have been coded. The subroutines which handle recycle systems and start-of-life reactivity adjustment are presently being changed. After these are done, debugging of the burnup analysis can start.

RBU

Further investigation of the double mass variation approach in treating the chemical binding effects of moderating scattering centers indicates that this method is capable of describing the binding effect on both average energy exchange and scattering angle at thermal energies. At this point the problem appears to be a theoretical one of obtaining the mass variations, since the Monte Carlo program is now prepared to handle the double mass variation.

To generate the desired averages, one introduces two mass parameters in the gas kernel. One parameter is used in selecting a suitable target velocity vector, and the other is used exclusively in the collision mechanics.

Taking the zero'th moment, the first moment in energy, and the first Legendre moment of the gas kernel, one obtains for the average cosine of the scattering angle, and the average energy exchange in the lab system the following forms:

$$\overline{\mu(E_0)} = f\left[E_0, M_1(E_0), M_2(E_0)\right]$$

$$\overline{\Delta E(E_0)} \doteq g\left[E_0, M_1(E_0), M_2(E_0)\right]$$

where:

E_0: Initial neutron energy.

$M_1(E_0)$, $M_2(E_0)$: The mass variations.

Using the kernel codes one may calculate values of $\bar{\mu}_0(E_0)$ and $\overline{\Delta E}(E_0)$ and solve the above expressions for $M_1(E_0)$ and $M_2(E_0)$. Although the correct expectation values of μ_0 and ΔE should be generated in the Monte Carlo, the detailed structure of the angular flux distributions may not be preserved. In addition, detailed balance will not be satisfied for higher moments in energy.

Both the zero'th and first moment in energy have been obtained and the calculation of the first Legendre moment is 40 percent complete. It is anticipated that a machine program will be required to obtain $M_1(E_0)$ and $M_2(E_0)$ by iterative methods.

C6 - SPECTRUM Iteration Scheme

The C6 age slowing-down calculation and the SPECTRUM thermal diffusion codes are to be tied together in an iteration scheme to converge on the thermal cross sections and slowing-down source to the thermal group.

Since C6 uses age theory, an output scattering matrix to generate input to the SPECTRUM code is necessary. The functions describing the probability of scatter from a group to any lower group for a given scattering nucleus have been obtained and separately programmed.

xt these functions must be included as a subroutine in the C6 program. Questions of the proper weighting functions for scattering in a homogenized region must still be investigated.

Instrumentation and Systems Studies

All of the equipment necessary to perform the PRTR Test No. 35 (PRTR Kinetics) was installed and/or checked at the PRTR site during the month. The test, which was originally scheduled for June 8 has been postponed. Postponement of the test will result in the need for additional rechecking of equipment just prior to the next scheduled test period. Non-Process Change No. 44 was prepared to obtain permission to install an electrical transducer on the PRTR pressurizer level transmitter. It will be used to study the relationship between pressurizer level indications and degasser effluent flow.

The PRTR nuclear excursion study was intended to determine the nuclear accident potential of the reactor using a core consisting entirely of mixed-crystal fuel elements. Analog runs were made to determine the maximum power level excursions produced by various reactivity disturbances with the reactor status varying from subcritical to normal operating power level.

One experimental PRTR Liquid Effluent Gamma Monitor of the scintillation transistorized type was installed for testing at PRTR. It is operating satisfactorily to date. The second generation monitor is about 50% completed in fabrication.

Technical assistance was rendered to PRTR Technical Planning and to Maintenance and Equipment Engineering regarding the PRTR Fuel Element Rupture Monitor and the PRP Critical Facility. A general improvement project was outlined for the rupture monitor and specific circuit changes and performance tests were recommended. All installation and physical modification effort is to be supplied by PRTR technician personnel. The initial problem is to increase the dependability of the monitors and to reduce the ambiguity of the indicated signals by removal of output interconnections of the several channels. Only in this manner can alarm signal validity be established by comparison methods.

The PRPCF assistance was in the form of testing of neutron proportional and fission counters and preamplifiers in systems which were not performing correctly. Guidance and instructive effort was rendered to PRTR personnel regarding the stated items.

In addition to assistance to PRTR on vibration measurements, Physical Testing also helped locate a new point of difficulty in the PRTR systems. During routine startup procedures metal pieces were found on one of the primary loop system screens. Suspecting that the piece came from vanes in a flow-straightening section of the loop, radiographs were made. The section involved was 14 inches in diameter and filled with heavy water. Using Co^{60} techniques satisfactory radiographs were obtained confirming that a metal piece had broken out.

NEUTRON FLUX MONITORS

Computer calculations, which were based upon the flux monitor computer program established to provide information concerning the optimum fissile and fertile radionuclide concentrations for the longest useful lifetime in a given reactor environment, were performed for various plutonium radionuclides and were found to give unrealistic values. It was determined that the plutonium radionuclides computer input data generated a matrix that was nearly singular during a particular computer inversion process; consequently, the incorrect answer was considered to be generated by noise rather than by the true input data. At present, portions of the computer program are being revised to utilize sixteen, rather than eight, significant figures during the stated inversion process, and since the technique is an important segment of the whole program, it is essential that the calculations generate valid optimum detector compositions for the radionuclides of interest.

In an effort to establish a proper experimental program for the detectors, it was determined that information must be obtained relative to flux levels, temperatures, and spectra to which the detector will be exposed during the irradiation tests. In addition, the detector output, as measured by the macroscopic fission cross-section, will have to be evaluated as a function of time. Since high ionizing field intensities will be encountered within the reactor core, investigation of possible chamber parameters was started in an attempt to optimize the chamber design.

The available HAPO test facilities for use in the irradiation testing of the experimental detectors were inspected.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

Modifications were made in the laboratory multiparameter eddy current testing equipment in preparation for its evaluation on metal test specimens. An improved null balance system was built, and tests show that it will simplify the adjustment of both conventional eddy current test equipment and the multiparameter equipment. Engineering design of a prototype multiparameter eddy current testing device was started. Assistance given to Physical Testing on tests of 1/4 inch diameter Inconel tubing at the NPR site was continued.

Modification of the multiparameter eddy current test equipment included fabricating two sizes of flat, probe type test coils, about 1/4 inch and 1/2 inch in diameter, respectively, and making and checking a null balance circuit for use with the new coils. A null balance system is being developed which is adaptable for use in the multiparameter test. Tests of a prototype unit show that it will facilitate the adjustment of single frequency eddy current equipment and should be an aid in both the laboratory and the field in setting up eddy current equipment. The new device permits adjustment of the instrument null with one control lever in contrast to the conventional method of using two controls to perform the same function. In addition, a voltage, or impedance plane plot of the test coil condition can be made manually for record purposes. An invention disclosure is being prepared.

The desired features of the multi-parameter tester have been listed. The various circuits to accomplish these objectives have been investigated. Where possible, several different circuits have been compared, so the most effective (and where possible, the most simple) circuit can be devised. Tentative circuits have been devised for use in the oscillators, buffer amplifier, nulling system, and adders. The exact components to be used in these circuits will be determined as more information is received from the manufacturers.

The problem of noise reduction has been considered. It was decided to attempt to remove the noise from the various A-C signals before driving the pickup coils rather than try to cancel the noise in the signals from the pickup coils. This approach should allow the use of simpler nulling circuits.

Assistance to Physical Testing included readjustment of the internal tester after moving it to the location of the second set of 581, 1/4 inch O.D. installed Inconel tubes to be tested. The commercial tester with cylindri-

The PRPCF assistance was in the form of testing of neutron proportional and fission counters and preamplifiers in systems which were not performing correctly. Guidance and instructive effort was rendered to PRTR personnel regarding the stated items.

In addition to assistance to PRTR on vibration measurements, Physical Testing also helped locate a new point of difficulty in the PRTR systems. During routine startup procedures metal pieces were found on one of the primary loop system screens. Suspecting that the piece came from vanes in a flow-straightening section of the loop, radiographs were made. The section involved was 14 inches in diameter and filled with heavy water. Using Co^{60} techniques satisfactory radiographs were obtained confirming that a metal piece had broken out.

NEUTRON FLUX MONITORS

Computer calculations, which were based upon the flux monitor computer program established to provide information concerning the optimum fissile and fertile radionuclide concentrations for the longest useful lifetime in a given reactor environment, were performed for various plutonium radionuclides and were found to give unrealistic values. It was determined that the plutonium radionuclides computer input data generated a matrix that was nearly singular during a particular computer inversion process; consequently, the incorrect answer was considered to be generated by noise rather than by the true input data. At present, portions of the computer program are being revised to utilize sixteen, rather than eight, significant figures during the stated inversion process, and since the technique is an important segment of the whole program, it is essential that the calculations generate valid optimum detector compositions for the radionuclides of interest.

In an effort to establish a proper experimental program for the detectors, it was determined that information must be obtained relative to flux levels, temperatures, and spectra to which the detector will be exposed during the irradiation tests. In addition, the detector output, as measured by the macroscopic fission cross-section, will have to be evaluated as a function of time. Since high ionizing field intensities will be encountered within the reactor core, investigation of possible chamber parameters was started in an attempt to optimize the chamber design.

The available HAPO test facilities for use in the irradiation testing of the experimental detectors were inspected.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

Modifications were made in the laboratory multiparameter eddy current testing equipment in preparation for its evaluation on metal test specimens. An improved null balance system was built, and tests show that it will simplify the adjustment of both conventional eddy current test equipment and the multiparameter equipment. Engineering design of a prototype multiparameter eddy current testing device was started. Assistance given to Physical Testing on tests of 1/4 inch diameter Inconel tubing at the NPR site was continued.

Modification of the multiparameter eddy current test equipment included fabricating two sizes of flat, probe type test coils, about 1/4 inch and 1/2 inch in diameter, respectively, and making and checking a null balance circuit for use with the new coils. A null balance system is being developed which is adaptable for use in the multiparameter test. Tests of a prototype unit show that it will facilitate the adjustment of single frequency eddy current equipment and should be an aid in both the laboratory and the field in setting up eddy current equipment. The new device permits adjustment of the instrument null with one control lever in contrast to the conventional method of using two controls to perform the same function. In addition, a voltage, or impedance plane plot of the test coil condition can be made manually for record purposes. An invention disclosure is being prepared.

The desired features of the multi-parameter tester have been listed. The various circuits to accomplish these objectives have been investigated. Where possible, several different circuits have been compared, so the most effective (and where possible, the most simple) circuit can be devised. Tentative circuits have been devised for use in the oscillators, buffer amplifier, nulling system, and adders. The exact components to be used in these circuits will be determined as more information is received from the manufacturers.

The problem of noise reduction has been considered. It was decided to attempt to remove the noise from the various A-C signals before driving the pickup coils rather than try to cancel the noise in the signals from the pickup coils. This approach should allow the use of simpler nulling circuits.

Assistance to Physical Testing included readjustment of the internal tester after moving it to the location of the second set of 581, 1/4 inch O.D. installed Inconel tubes to be tested. The commercial tester with cylindrical outside test coil was set up, checked in the laboratory, and then moved to 100-N Area for use in testing about 140,000 feet of 1/4 inch O.D. Inconel tubing prior to its installation. The original test coil design failed due to wear after about 14,000 feet of tubing was tested. Two additional probes were wired, and one was furnished with a plastic guide bushing which can be replaced when it becomes worn. A second, standby tester, with tube feeder is being readied for this test.

Heat Transfer Testing

Changes were made in the dual radiometers to improve their operation. Heat transfer defects in the nickel to uranium bonds of nickel plated fuel elements were detected. A passive analog simulator was fabricated and used to study the surface temperature behavior of nickel plated uranium fuel elements during heat transfer testing.

Heating of lens masks, installed on the radiometers to prevent cracking of the lenses due to heat from the arc, caused spurious signals due to emitted and scattered radiation. The mask closest to the arc was replaced with a water cooled lens barrel.

An electronic resistance-capacitance analog simulator has been fabricated for use in studying the surface temperature of the nickel during heat transfer testing under various conditions. Initial results from the simulator indicate that variations in thermal contact conductances less than 80 BTU/hr ^{o}F ft^2 should be detectable using the present test parameters. Conductances greater than this cause the surface temperature differences that appear during heating to drop almost to their equilibrium values before reaching the detector 20 milliseconds later. It would be necessary to observe the surface temperatures of the nickel within about 5 milliseconds after the heat application cycle to detect thermal contact conductance changes between 80 and 1600 BTU/hr ^{o}F ft^2. The rapidity of the surface temperature transients is due to the small heat capacitance of the thin nickel plate.

Experiments have shown that the present heat transfer testing system is sensitive to large variations in thermal contact conductance between the nickel and the uranium in nickel plated uranium fuel elements. Seven nickel plated uranium fuel elements were used in these experiments. The nickel plate varied between .0003 and .0008 inch thick. Defects in the nickel/uranium bonds in two of the fuel elements were produced by omitting one of the cleaning steps prior to the nickel plating step. These defects were easily detected, and were permanently recorded as melted and discolored areas in the nickel plate. No defects could be detected in the remaining five fuel elements, even though they were tested a number of times at high heat fluxes. No discolored or otherwise visibly damaged areas appeared. These fuel elements will be destructively examined to ascertain whether or not defects were present.

Zirconium Hydride Detection

Tests are being made to learn whether or not X-ray diffraction or electrode potential measurements can be used to determine hydride concentrations in Zircaloy-2. Hydrided Zircaloy-2 powder samples have been obtained for use in nuclear magnetic resonance studies.

Ultrasonic studies related to zirconium hydride detection are continuing. In addition, parallel studies are being carried out to determine whether or not other physical effects could be used. Previous work indicated that X-ray diffraction could be used to detect hydride at concentrations down to 300 ppm. However, difficulty in obtaining quantitative concentration estimates was encountered, apparently due to preferred orientation. X-ray diffraction studies using a rotating sample holder are being carried out by Chemical Separations, HLO, in response to a request from Physical Measurements, HLO.

Electrode potential measurements were made with the assistance of Chemical Separations, HLO, on Zircaloy-2 samples containing 0, 50, 300, and 500 ppm hydrogen. Initial experiments using 20% HNO_3, 1% HF, 79% H_2O electrolyte showed no difference in electrode potentials of the samples. Measurements are being made using other electrolytes.

Powdered Zircaloy-2 samples containing 496, 714, 2195, and 16,000 ppm have been obtained for nuclear magnetic resonance studies. Arrangements were made last year with the University of Washington and with Varian Associates to make these studies.

USAEC-ACEL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

As an alternate to the use of notches for standards and for calibrating the ultrasonic test of thin-walled fuel element sheath tubing, the use of drilled-holes is being investigated. An anomalous response was obtained in one instance in comparing a drilled-hole to a notch. The ultrasonic response from a six mil diameter hole drilled completely through a Zircaloy-2 plate was obtained using three different lithium-sulphate transducers operating at 10 Mc. With a line-focused and a flat rectangular transducer a regular decrease in ultrasonic response with increase in metal-path from the hole was found. Using a point focused transducer the ultrasonic response decreased in a saw-tooth manner. In contrast, using notches a saw-tooth behavior was encountered with the line-focused transducer. No explanation for this difference in behavior is apparent at the present time.

In the evaluation of production test parameters, the degree of instrument variations such as signal display non-linearity, gain setting non-linearity, and frequency bandwidths are important. Three types of calibration signals which simulate pulses were used to obtain instrument non-linearity curves. Comparison with pulses from actual Lamb-modes gave non-linearity curve agreement to within plus or minus 3%. It would appear that simulated pulses can be used to obtain corrections for instrument non-linearity. Curves of signal indications as a function of front panel gain setting were also obtained. These curves follow a logarithmic behavior which is typical of most pulse amplifiers.

While the optimum frequency bandwidth desirable for tubing tests with Lamb-waves is not known, the phase velocity-frequency thickness product curves show that, at constant entry angles near V1, a frequency bandwidth of about 2 Mc for a center frequency of 10 Mc should permit excitation of at least one Lamb-mode for all tubing wall thicknesses of about 0.020 inches and above. Experiments using two different instruments have proven this supposition. Further, analysis of pulse frequency components leading to calculated frequency thickness products provided excellent agreement with the theoretical Lamb-wave curves. The experiments indicate a 10 Mc bandwidth of about 2 Mc is, therefore, appropriate.

Fundamental studies have indicated that a type of Lamb-wave propagation must be encountered in the ultrasonic testing of thin-wall tubular components. In order to construct a physical picture of Lamb-wave propagation behavior, a theoretical treatment for separating the wave into longitudinal and shear components was developed. Plate end leakage observed with the Schlieren system during Lamb-wave propagation was satisfactorily explained qualitatively based on the wave separation method. Efforts to obtain quantitative amplitudes to show that the transmitted beam results from a Lamb-wave is complicated by the existence of critical angles which give surface waves. However, the presence of Lamb-waves and other wave types reflected from a plate end has been verified for certain incident Lamb-wave situations using the Schlieren system. Future experiments may be assisted through recognition of Lamb-waves and other waves as combinations of longitudinal and shear-wave components.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Investigation continued in the study of lateral dispersion as related to the wind direction spectrum observed at a point. The dispersion of the time mean plume generated from a continuous source, as measured in our experimental technique, is the result of both diffusion and transport of the instantaneous plume elements. Whether the dispersion at a given distance arises primarily from diffusion or from meander depends on the spectrum of turbulent eddy sizes relative to the plume dimensions, i.e., eddies smaller than the plume dimensions cause the diffusion while the larger eddies only contribute to meander. Since dispersion in the atmosphere is a Lagrangian phenomenon, it is essential to determine the relationship between the fluid-attached Lagrangian time scale and the fixed point Eulerian time scale applicable to the meteorological measurements.

Five experiments were conducted to determine the relationship between the travel time of a puff of tracer to a given distance and the travel time determined from the Eulerian wind velocity measurements at 1.5 meters' height. Records from the direct reading sampler permitted calculation of the travel time. As expected, considerable "stretching" of the puff was observed due to the deformation field resulting from wind velocity shear. A puff formed by a 15-second release required 105 to 210 seconds to pass the sampler 200 meters downwind. The ratios of Lagrangian to Eulerian travel time ranged from 0.82 to 1.10. These results indicate that the ratio is approximately one for stable atmospheres and short travel distances, but more data are required to test the validity of the assumption at greater distances.

In Air Force-supported programs, diffusion data from Vandenberg Series II experiments were transmitted to the Air Force. Off-site work on Vandenberg Series III experiments was completed on June 30.

A prototype-direct-reading sampler for zinc sulfide tracer material was field tested with encouraging results. Concentrations of less than 2×10^{-6} grams per cubic meter of air were detected.

In our rain scavenging work, effort was directed toward further delineating factors affecting the experimental measurements and designing experiments and techniques to either circumvent some of the confounding ones, or to determine their relative importance in the scavenging processes. Results obtained so far, although tentative, indicate that the collection efficiency of very small drops (less than 0.8 mm.) is much greater than expected. Since small drops are far more numerous than larger ones in natural rainfall, this deviation from theory would alter considerably interception predictions. Therefore, the most logical next step is to determine the relative scavenging efficiency of small and large drops, and to try to explain the deviations in the measurements from theory. Problems of air-isokinetic sampling, wettability of particulates, and deposition of tracer on the ground and vegetation can be circumvented temporarily.

A Research and Development Report titled, "The Hanford Raindrop Sampler and Selected Spectra," HW-73119, by R. J. Engelmann, was distributed during the month. An instrument is described that utilizes a sensitive paper to record individual raindrop size and particulate content. Means of reducing splash and edge-effect errors are described. Nine rain and one snow spectra are shown in demonstration of the detail that may be obtained in the small drop portion of the spectrum.

Dosimetry - 06 Program

Measurement of the radioactivity in Alaskan Eskimos was begun. More than two hundred people were counted at Kotzebue. The amounts of Cs-137 found ranged from what would be typical in the rest of the United States up to, in one case, 518 nc. The amounts in the body show a close relation to the diet. Most of the Cs-137 appears to come from eating flesh of such animals as caribou or reindeer. The only other isotopes found with any frequency were Zr-Nb-95. Tests have indicated that it is present as external contamination. The equipment was moved to Point Barrow for further studies.

Semi-annual maintenance was completed for the Van de Graaff. It operated satisfactorily during the month.

The investigations related to the Recuplex accident were completed.

Calibration of the polyethylene double moderator was completed.

Two neutron spectrometers were studied. Fast neutron peaks were observed with a high-pressure He^3(n,p)T proportional counter, but the chief use of this device may be as a high sensitivity thermal neutron detector. A solid state spectrometer utilizing Li^6 was calibrated and used to make a measurement of a PuBe spectrum. The device has good resolution but very low sensitivity.

Double-moderator neutron counting apparatus was installed in a covered pickup truck to permit measurement of near-background neutron fields at different locations.

The discrepancy between the calorimetric measurements of a plutonium source performed here and at Mound Laboratory was found to be due to an unsuspected heat leakage during one of our calibration steps. This leakage was eliminated. The results of the two laboratories now agree to within 0.4%.

Radiation Instruments

The manual keyboard entry was installed on the new RIDL 400-channel analyzer at the Whole Body Counting Facility. The keyboard permits the hand-punching of data into the punched paper tapes which are later processed by EDPO.

Investigations continued on the recharging type pocket dose meters. One dosimeter with a 0.0005 inch diameter stainless steel fiber was irradiated to a total dose of about 1000 r before any failure due to sticking of the fiber to the rod. By arranging a circuit and two 0.00025 inch diameter platinum-tungsten fibers properly, the mechanism of the sticking trouble was determined. The fibers were arranged in a manner that the second fiber would not touch the center rod of the dosimeter unless the first fiber failed to make contact. In this way, if the first fiber was failing by contact-sticking to the rod, the second would return to the rest position. Tests showed that the first fiber always made contact with the rod; thus, when sticking does occur, the fiber is making electrical contact to the rod. Tests are being continued with the 0.00025 inch diameter platinum-tungsten fibers as these have shown excellent promise to date. Six prototype recharging type dosimeters were received from off-site as fabricated with platinum fibers. Tests were just started with these.

Further work was done regarding transistor circuits to be used for noise pulse cancellation wherein the particular characteristics of noise pulses, as compared to normal scintillation signal phototube pulses, are being exploited in an attempt to provide self-cancellation.

Several tests were made using silicon surface barrier diode detectors, in conjunction with the multi-channel analyzer, for detection of filter-collected plutonium particulates in the air. Resolutions of about 1.5% were obtained for the Pu^{239} alpha energy values, and it appears that the approach will provide adequate discrimination to detect filter-collected airborne Pu^{239} under high radon-thoron background conditions.

Field tests were made with the continuous operation real-time or field-type airborne zinc sulfide particle detection system used for air movement and diffusion studies. A membrane filter was used in conjunction with the continuous unit to provide calibration and sensitivity data. This method permitted measurement of the total mass of ZnS particles which passed through the real-time detector chamber. Twenty sampling runs were made and the sensitivity was such to provide a 10% of full scale increase in chart reading for ZnS concentration of 1×10^{-7} grams per cubic foot; thus, it would be possible to detect concentrations of 3×10^{-8} grams per cubic foot on a continuous basis. In addition, tests were made with different air flow rates to determine the proper rate for the required sensitivity. The sensitivity increased by a factor of 2.4 for a flow rate increase from 1 CFM to 2.4 CFM.

The special transistorized line-operated instrument, which uses an underwater G.M. tube detector, was completed and installed at the Display Center in Richland. The project, devised in cooperation with Biology Operation, entails the use of a number of fish in a tank wherein certain fish have Co^{60} wire implanted in them. As the so-implanted fish swim closer to the G.M. tube, the output signal, a gated one kilocycle per second note, of the instrument changes rate. This audible signal is very easily heard throughout the Display Room.

Fabrication was nearly completed, except for installation of reed-relays which have not been received as yet, on the prototype scintillation transistorized six decade logarithmic response area radiation monitor. Fabrication is about 30% complete on a similar prototype instrument which employs four decaded combination logarithmic and linear (selectable) response ranges.

Assistance and advice was rendered to Chemical Effluent Technology, HLO, concerning preparation of an electrical circuit for a resistance analog of HAPO ground water aquifers. A method of measurement and circuits for highly accurate power supplies were provided.

The decade scalers portion of the Portable Mast System being developed for Atmospheric Physics, HLO, was completed and all decades were satisfactorily tested. A suitable circuit was developed to gate the scaler input data from the wind speed transducers. The gates are actuated from the main programmer unit and a total of six gate modules and four decade scalers are used. The gate modules, which use a binary for storage and the transistor gates, are used with a trigger circuit which translates the input sinusoidal waveform into proper pulses. The trigger circuits drive through the gate modules to the scales. A power supply was designed for use with the system and it is being fabricated. All system programmer tests continue to be satisfactory. The system is approximately 30% completed at this time.

Considerable revisions were made to the circuit drawings for the data stations of the Atmospheric Physics Telemetry System. Typing was completed on the report covering all system revisions, maintenance procedures, and general equipment descriptions.

The Ampex tape recorder electronics purchased by Atmospheric Physics Operation was tested and used successfully in conjunction with the Sanborn Electronics purchased by Systems Research Operation. The combination makes available a seven-track FM tape recording unit.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples as received during June. These analyses were performed using the single-filament ion-source mass spectrometer.

The mass spectrometer for this program has been out of service due to the lack of a satisfactory electron multiplier. A survey of past experience with the RCA 6810 multipliers showed that all of the usable ones have been constructed with a ceramic structural insulation while the unsatisfactory ones have been made with glass insulation. A ceramic insulated multiplier which had been in service in the single-filament ion-source spectrometer has now been installed in this instrument. A precision decade voltage divider with an accuracy better than 0.04 percent of its indicated ratio has been used to remeasure the linearity and accuracy of the vibrating reed electrometer and recorder parts of the data recording system.

Preliminary work was done towards setting up a solid state alpha particle detector system to be used for routine assay of loaded sample filaments.

No further progress has been made on the study of the post-acceleration scintillation detection system due to the necessity of extended maintenance of the two mass spectrometers.

TEST REACTOR OPERATIONS

Operation of the PCTR was routine during June although there were 5 unscheduled shutdowns. Four resulted from electronic failure and one was caused by incorrect bypassing technique.

The experiment to determine the possibility of boron and plutonium analysis by reactivity measurements was completed.

Calibration irradiations were made of two sets of dysprosium foils.

An experiment designed to determine the nuclearly safe concentration of uranyl fluoride was started. The effects of container material and thickness are being investigated.

The following maintenance items were completed or essentially completed:

1. The rear face inner flux leveling slug drive was reactivated.
2. The graphite for a permanent thermal column for the top of the reactor was machined.
3. The reactor room air conditioner was installed.

The TTR and the critical approach tank were not operated. Minor maintenance and preventive maintenance items were completed.

CUSTOMER WORK

Weather Forecasting and Meteorological Service

Consultation service was rendered on meteorological and climatological aspects of Carbon-14 release from the reactor areas to RPO and transmission line maintenance scheduling to CE&UO.

A draft of the procedure for first predictions of the dispersion of airborne material released from a serious radiation event was distributed for comment.

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

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	90	87.5
24-Hour General	60	87.1
Special	165	92.1

Temperatures during June averaged near normal with a generally cool first half and warm second half. Precipitation and cloudiness were much below normal and solar radiation much above. There was a high wind on the 25th.

Instrumentation

All laboratory tests were completed on the Automatic Conveyor Type Alpha-Beta-Gamma Laundry Monitor and the complete system is now partly installed at the 200-W Laundry Facility. The mechanical components are 50% installed. Work continued on the preparation of circuitry schematic and on instruction manual.

All testing was completed on the scintillation transistorized Columbia River Radiation Monitor designed for Environmental Studies and Evaluation, Radiation Protection Operation. The instrument, which uses a 3-inch diameter by 3-inch long terphenyl-in-polyvinyltoluene detector, provides an 8% of full scale reading increase for a six micro-roentgen change in dose rate. Three decaded ranges are provided with the middle range having a full-scale value equating to about 50 micro-roentgens per hour. A temperature change of 80°F, from +50°F to +130°F, gives a reading error of 8% of full scale or 0.1% error per degree F change. This is considered to be acceptable.

Circuitry was nearly completed for a transistorized alpha monitor with aural output, for use with air proportional type alpha probes. The unit was designed for Finished Products Chemical Technology, CPD.

Assistance and suggestions were conveyed to Chemical Development, HLO, personnel regarding methods of increasing directional sensitivity of a neutron counter used to scan process cells and columns.

All circuits have been completed for the coincidence-count type alpha air filter counter, for counting standard HAPO filters, as designed for Radiation Monitoring Operation, RPO. Testing was just started on the system.

The designed holding and positioning mechanism for use with a nine-inch diameter NaI crystal was completed and is operating satisfactorily for Analytical Laboratories, HLO.

Fabrication continued at an accelerated pace in the 328 Building Electronics Shop on one final "field model" coincidence-count type alpha particulate continuous air monitor. The scintillation transistorized instrument was designed for Radiation Protection Operation, HLO.

Consultation was rendered to Component Testing Operation, IPD, regarding possible instrumentation methods for digitization of data from chart recorders.

A dc to dc converter was fabricated for use in the sensitive gamma transistorized aerial survey monitor of Environmental Monitoring, RPO.

The controller manufacturer who supplied the 333 Building autoclave controls has been asked to suggest an instrument modification which will allow the complete removal of the controller reset mode. This feature was specified originally, and apparently was provided, but the circuit does not function properly. Elimination of this control mode will improve start-up operations without appreciably affecting the steady state performance of the system.

Work continued on the design of a solid-state scanner-programmer for use in Physical Metallurgy Operation's creep capsule data logging system. Emphasis has been on the development of circuitry for digitally controlling the micro-positioner in each capsule. The micro-positioner is the primary system for creep measurement and is used to calibrate the LVDT's used in the secondary system.

The cause of high temperature trips in the Physical Metallurgy Operation's swelling capsules on four occasions during the month was found. The Minneapolis-Honeywell protecto-vane pyrometers which are used as high temperature limiters are extremely sensitive to line voltage variations. The cause of the sensitivity to line voltage was pinpointed in the upscale thermocouple burnout circuit and the unusually high external impedance of the protecto-vane. Laboratory tests showed a 20% change in meter reading per 10% change in line voltage. A phone call to the manager of Electrical Utilities Operation, and to the Electrical Engineer, 100-K Maintenance, explained the timing of the high temperature trips. During the eight-week period over which the trips occurred, power conditions at 100-K were the worst in years. A 10% line voltage variation was very possible. Recommendations were made to purchase voltage regulating equipment which would remedy the situation.

Optics

The Optical Shop is making an elliptical mirror to be used as a 30 x objective in a high temperature microscope. The process involves selective grinding and polishing in five zones followed by testing to determine what further corrections are necessary. To date we have gone through five cycles of grinding, polishing, and testing. The work continues.

An experiment was conducted to explore the characteristics of "short wavelength" radiometers. It is believed that this type of radiometer is one of the most reliable temperature indicators. It would seem particularly adaptable to use in reactor environments.

The characteristics of commercially available lasers are being investigated. It is believed that a pulsed laser would be an ideal source of radiant energy to use in measuring thermal conductivity of materials at high temperatures.

A spare 6-inch model optical traverse mechanism has been assembled and a new 12-inch model has been fabricated and tested.

During the four-week period (June 3 to July 1) a total of 392 man-hours work was performed. The work included:

1. Fabrication of twenty glass bearings for two different waste storage tank pumps.
2. Fabrication of components for two optical traverse mechanisms.
3. Repair of a metallograph for 327 Bldg.
4. Repair of a macrocamera for Bob Emmer, FPD.
5. Fabrication of components for an electrical readout traverse mechanism.
6. Repair of one microscope for Biology.
7. Modification of two periscopes for use in viewing VSR Channels.
8. Copper plating twenty fuel element components for Fuels Fabrication Development, HLO.
9. Adjustment of a camera and cutting of 40 slate blocks for Biology Operation.
10. Repair of crane periscope heads for Purex, B-Bldg., and Redox.
11. Aluminizing one bell jar for Plutonium Fuels Development.

Physical Testing

Service testing work proceeded routinely, with almost the entire gamut of nondestructive and supplementary tests being used. A total of 3,281 tests were made on 3,082 items, representing some 35,209 feet of material. Work was done for 23 different HAPO components representing most of the operating departments and service organizations, and other AEC contractors. Advice was given on 47 different occasions on general testing theory and applications. Two training courses, one on Report Writing and one on Fundamentals of Eddy Current Testing, were conducted by Physical Testing Operation as part of a continued effort aimed at increasing the value of service to current and prospective customers. The report writing course was adapted to the particular memorandum used by Physical Testing Operation in reporting results. The eddy current course consisted of lecture and laboratory sessions featuring practical experiments to illustrate eddy current applications. Course instructors in both instances employed on-site personnel. The testing and treatment of NPR process tubes was resumed on June 18 after an extended NPR work stoppage.

Small diameter, inconel instrument lines were successfully eddy current tested at the 100-N site for IPD. Two penetrations of some 580 tubes each through 11 foot of concrete were tested, using an inside eddy current probe. A total of 47 tubes were found exceeding the test standard. The tubes were assigned a priority listing to be used in the event of an emergency need only. Testing was started, using an encircling coil eddy current test, on some 150,000 feet of tubing yet to be installed.

Considerable assistance was given to PRTR operations in measuring vibration in the reactor process tubes. Process tube damage, found during periodic tube monitoring, is postulated as being caused, among other things, by a vibration of the fuel element bundle and/or piping system. Cold vibration measurements have been made on both the top and bottom faces under various flow conditions and with different combinations of primary circulating pumps. A limited number of tubes will be monitored hot. A possible long range development could be the insertion of a vibration-measuring instrument package either in the fuel element tube hanger, or in a fuel element itself.

In conjunction with fatigue testing being conducted by IPD on NPR primary loop piping to determine serviceability, discontinuities detected by ultrasonics prior to testing were observed ultrasonically at intervals during testing. Changes were detected in ultrasonic response as the tests proceeded. Whether the increase in response prior to failure is growth of the discontinuity, change in angle of the discontinuity, or initiation and growth of a fatigue crack remains to be determined by destructive analysis. Positive correlations were obtained between the point of failure and prior ultrasonic indications, though not all ultrasonic indications cause failure.

Work was continued in assisting CPD in establishing quality levels of process vessel fabrication. After re-examination of a vessel fabricated and tested off-site revealed objectionable discontinuities, repair work was initiated. In the course of repairing the weldments, removal of metal has substantiated the occurrence of discontinuities detected radiographically.

The fabricator is being back-charged for the cost of repairs. Reliable detection of such discontinuities are essential to CPD to reduce the cost of process equipment failures.

An evaluation of a LogEtronic printer and enlarger for duplicating radiographs has been initiated at the request of Commission personnel. The equipment has the capability of exactly copying radiographs or optionally changing the scale of contrast. It is possible with the use of such automatic dodging equipment to enhance radiographs of low density, thereby displaying detail that may not be apparent in the original. Conversely, it is also possible to reduce radiographs of high density so that a readable radiograph is obtained.

Preliminary examinations were completed on process tube number 2371 H which was removed from H-reactor as a result of defect indications obtained from the functional ultrasonic process tube test. Three six-inch sections which gave indications approximately comparable to a 0.020 inch deep by 0.125 inch long transverse I.D. notch standard were examined visually and with dye penetrant. The sample which had the largest ultrasonic indication had a transverse I.D. scratch and several O.D. pits which had extensive bleed-back indicating considerable depth. The other two samples had deep appearing transverse O.D. scratches and some pitting, all of which had bleed-back. These samples will be sectioned for examination of defect depth.

Eight fission product transient samples were bond tested for Plutonium Fuels Development. One sample had an unbond indication comparable to a 0.25 inch diameter unbond standard.

Analog Computer Facility Operation

The major computer problems considered during the month were:

1. NPR Simulator
2. Reactor Instrumentation Studies
3. Radioactive Waste Disposal Heat Transfer
4. PRTR Nuclear Excursion Study

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

GEDA	EASE	
154	154	Hours Up
22	22	Hours Scheduled Downtime
0	0	Hours Unscheduled Downtime
0	0	Hours Idle
176	176	Hours Total

The over-all maintenance of the computers and support equipment has been good. The 22 hours of scheduled downtime is for one hour per day for routine adjustments and preventative maintenance. Equipment reliability for the month of June was higher than ever before. The reactor instrumentation study involves an 11-node reactor model and is the most complex problem attempted. The completely satisfactory equipment operation with this complex simulation opens the way for new methods of computer time and operation administration which can greatly extend the usefulness of the equipment.

Instrument Evaluation

A special scintillation transistorized combination logarithmic and linear response area monitor, which covers a dose rate range from 0 to 50 r/hr in four decades, was completed in evaluation, calibrated, and delivered to Fuels Development Operation, HLO, 326 Bldg., for routine use. Calibration curves, operating instructions, and prints were delivered.

Evaluation tests continued on some four ampere-hour capacity rechargeable nickel-cadmium batteries for use in portable radiation monitoring instruments.

Attempts were made, via visits to Instrument Laboratory, Inc., Seattle, to secure fabrication completion and tests on 65 "Scintran" line-operated radiation monitors and on 30 portable transistorized neutron monitors which were ordered by Radiation Protection Operation for delivery before July 1, 1962.

Background gamma level compensation tests were conducted on the prototype background compensated beta-gamma hand and shoe monitor.

Two alpha-only scintillation transistorized hand counters, which also have external probes for alpha and beta-gamma contamination monitoring, were completed, evaluated, and delivered to Biology Operation and to J. A. Jones Co., respectively. The alpha readout level to background ratio, for a 500 d/m alpha source distributed over a 4 inch by 8 inch area, was about 5 to 1.

Evaluation tests were conducted on one previously-fabricated alpha-only scintillation transistorized hand counter for personnel in the 308 Building. All four hand probes were found to be performing properly and the difficulty seems to be one of circuitry maladjustment.

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CHEMICAL RESEARCH AND DEVELOPMENT OPERATION

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Treatment of NPR Decontamination Wastes

Experiments were conducted to determine the effect of triethanolamine (an additional complexing agent) on scavenging of Co-60 and other radionuclides from a dibasic ammonium citrate cleaning solution. Ferrous sulfate with caustic produced better scavenging of Co-60 than potassium permanganate with caustic. However, more iron was required to remove Co-60 from the citrate cleaner containing triethanolamine than from the citrate cleaner not containing it. For example, 800 ppm Fe^{+2} was required to remove 95 percent Co-60, compared to only 100 ppm Fe^{+2} without the triethanolamine. Furthermore, closer control of the caustic addition was necessary when triethanolamine was present. Maximum Co-60 removal was attained in the pH range 10.2 to 11.8. This is in contrast to increasingly effective removal of Co-60 with increased amounts of caustic when triethanolamine is absent.

Ground Water Temperature Studies

Temperature profiles of the ground water in the region surrounding the 100-B Area indicate thermally hot water is spreading out in all directions from a source area beneath the 100-B retention basins and from beneath the header pipes leading into these basins. The aquifer carrying hot water away from this site appears to be 100 to 150 feet thick, with the hottest zone near or at the water table. Isotopic analyses of ground water samples from this same general area show a pattern similar to that indicated by the temperature data. The greatest distance at which radionuclides have been detected in the ground water from the 100-B Area, however, is approximately 2.5 miles to the east; whereas, the temperature data trace the warm water as far as ten to twelve miles east of the 100-B Area.

Ground water flow rates, calculated on the basis of Cr-51 concentrations in the reactor effluent and in the ground water at varying distances from the 100-B Area, show the average rate of ground water movement to be between 25 and 35 ft/day. Based on this velocity, a gradient of five ft/mile, and a porosity of 20 percent, the average field permeability of the sediments underlying this region are approximately 50,000 gpd/ft^2. This high permeability value tends to confirm the belief that a buried river channel exists in this location and also supports the theory that the Columbia River recharges the ground water in this region through the buried channel.

Initial ground water characterization studies conducted at 100-K Area indicate significantly less noticeable effects of local effluent discharge than was noted at 100-B Area. Ground water temperature was near normal, 17 C, and radioisotope concentrations in the ground water were near background except in the vicinity of the disposal trench. Evidently, appreciably less effluent enters the ground water through leakage and disposal at 100-K Area in contrast to 100-B Area, or the warm ground water moves directly to the Columbia River when it reaches the water table with very little lateral spreading. Further studies of the 100-K Area ground water flow system are in progress.

Effluent Monitoring

Operation of the As-76 monitor continued without malfunction through the month of June. Iodine interferences were measured on the effluents from three reactors to determine the consistency of the interferences during normal reactor operation. Wide variations were not detected; the contribution of iodine in the As-76 measurement being about 30 percent at F, and 20 percent at both H and D. If the iodine interferences continue to be consistent, the need for an iodine removal method may be obviated.

Efficiency of Charcoal in Reactor Confinement Halogen Traps

A sample of charcoal was examined which had been supported in the filtered exhaust air from 105-F Building since the reactor confinement installation was completed. Iodine removal efficiency of a 1-1/4 inch length bed of this charcoal was found to be 97 percent for a linear velocity of 20 ft/min and 95 percent for air at 100 ft/min. These efficiencies, though somewhat lower than for fresh BPL charcoal, suggest that aging of charcoal in the filtered air streams has not seriously reduced the iodine removal efficiency.

Alum Production Studies

Analyses of product alum samples from IPD water treatment plants yielded the following average results: In the 100-DR alum plant, consisting of two stirred vessels in series, 92 percent of the acid-soluble bauxite was reacted at an operating temperature of 132 C and a total apparent liquid holdup time (no allowance for foaming) of 13 minutes. In the 100-H plant, consisting of a single stirred vessel, 85 percent of the acid-soluble bauxite was reacted at an operating temperature of 130 C and an apparent liquid holdup time of ten minutes. The higher percentage conversion found for the 100-DR plant may be due in part to continuation of the reaction while the rather large samples were cooling to room temperature. The 100-H samples were immediately diluted with water. Pilot plant runs reported earlier (cf. HW-73514 C) gave 80 to 95 percent conversion under similar conditions.

Reactor Effluent Water Radioisotope Studies - Silicate Addition

The silicate addition test on a chemically cleaned tube was interrupted by circumstances which required addition of normal process water during a five-day period. When the addition of silicate was resumed without another tube cleaning operation, radioisotope reduction of only 20-30 percent was obtained. Since laboratory evidence indicated that the adjustment of pH of the silicate to 6.6 resulted in a reduction of silicate ion available to the tube surface, a new test was initiated without pH adjustment. This results in a pH of 7 in the tube under test, compared with 6.6 in the control tube. After six days of operation the As-76 and P-32 concentrations were a factor of two, or slightly more, lower in the effluent from the addition tube and continued at this level for about two weeks. After an outage the silicate addition rate was increased to 20 ppm and a separate nitric acid addition begun to keep the pH of the water at 7. Initial results show that the As-76 (the only isotope for which data have been obtained to date) had, after one week, already decreased to about a factor of 3 lower than the control tube.

Deionized Cooling Water Test

Deionized cooling water has been continuously fed to a reactor tube containing the regular film resulting from process water since May 10, 1962. A reduction was noticed in all effluent radioisotopes until the first week of June when the activities increased rapidly and then began to decrease again. This increase coincided with an unscheduled period of two hours operation when regular process water was substituted for the deionized water. Because of this and numerous other short duration process water substitutions the maximum attainable reductions have not been determined. However, P-32, Np-239 and La-140 concentrations were reduced to one-fifth pre-test levels; Cr-51, As-76, Mn-56 and Cu-64 concentrations were reduced to one-tenth or less of pre-test levels. Some additional reductions should result as the test is continued.

X-Ray Fluorescence of Nickel and Uranium

This is a very sensitive technique for measuring plating thickness of copper on zirconium or nickel on uranium. Calibration with nickel foils showed that the attenuation of four fluorescent lines from the uranium base metal, each with a different calibration slope, could be used to determine nickel plating thickness in the range 2.5 to 50 microns (2.0 mils) with a sensitivity of about ± 0.5 micron. In the range 0.025 to 5 microns, the fluorescence intensity of two nickel lines will provide very high sensitivity for plating thickness.

Since this method is very sensitive to change in nickel thickness over the uranium it may be able to determine what portion of the nickel is caused to diffuse into the uranium during various heat treatments for diffusion bonding. It would thus serve as a tool to help optimize the bonding process.

SEPARATIONS PROCESSES

Disposal to Ground

Tritium analyses of river water collected from five to eight miles upstream of the 300 Area showed no concentrations in excess of the 1×10^{-5} μc/cc detection limit. Wells in this region do show low but positive concentrations of tritium in the ground water. Additional riverbank water samples have been scheduled.

The B-Swamp, located one mile east of 200-E Area, was found to contain tritium up to 3×10^{-3} μc/cc. This concentration was also noted in the Purex chemical sewer which has been the major supplier of water to the B-Swamp for the past several months. The source of the tritium is probably the vacuum acid fractionator overheads which are discharged to the chemical sewer. The tritium in the swamp probably accounts for the measurable concentrations of this radionuclide in wells east and north of the swamp site which routinely show less than detectable concentrations of other beta-emitting isotopes.

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Tritium in Gaseous Effluents

Redox stack effluent was sampled and analyzed for tritium as HT and as HTO. The sampling was carried out during metal dejacketing and a first dissolving cut. About 0.3 curie of tritium per day was emitted of which 0.24 curie was HTO and the remaining 0.06 curie was HT.

Iodine-131 in Airborne Effluents

The reaction between hexone adsorbed on charcoal and NO_2 subsequently passing through the bed was further studied. When the charcoal had been pre-exposed to hexone vapors, then a 10 percent NO_2 in air mixture passed through, a reaction occurred, raising the bed to 540 C. At 5 percent NO_2 a bed temperature of 40 C was reached, about that reached when 5 percent NO_2 is passed through fresh charcoal.

Charcoal which had been exposed to 10 percent NO_2 and then to hexone vapors failed to show any unusual heat generation. These concentrations of hexone and NO_2 are much higher than would be expected in Redox streams being considered for I-131 removal with charcoal; hence, the likelihood of a charcoal bed being destroyed from this source is very small.

Charging of 11-Inch Slugs to a Multi-Purpose Dissolver

An increase in the length of the fuel elements to about 11 inches for use in the production reactors is being considered. There has been some concern about the possible jamming of these longer slugs in the annular crib of the multi-purpose dissolver.

Two test chargings of dummy steel slugs 1.5 inches in diameter and 11 inches long were made to the carbon steel mock-up of the Redox multi-purpose dissolver slug crib. The annular slug crib has a 70-inch outer diameter and a 10.5 inch wide annulus. No jamming was observed during the dumping of 17 buckets of the steel slugs. The measured void volume (57 percent) and the slug distribution pattern were comparable to those previously observed with the nominal 8-inch slugs.

C-Column Studies

Relocation of the experimental C-Column and the associated process control instrumentation has been completed. Experimental capabilities of this pilot-scale solvent extraction column have been substantially expanded by the incorporation of a 200-channel data logging system. The new system, now installed and operating, is capable of logging all data points pertinent to operating variables in the column in 40 seconds. In addition to providing basic data for developing a mathematical model of the process, detailed studies of column stability and flooding characteristics are now possible with the greater rapidity of data logging. Output data are punched on paper tape, resulting in reduced costs of data processing.

The Gaussian non-linear least squares code was modified to allow control of the size of parameter changes on an iteration. This change will restrict the parameter values tried as possible solutions to a range that is known to be reasonable from an engineering standpoint.

Measurements of Stresses in Resin Beds

The program of stress measurements in moving beds of anion exchange resins was essentially completed during the month, although additional stress measurements will be made on beds of glass beads or other incompressible solids to permit comparison with published work of a related nature.

WASTE TREATMENT

In-Tank Solidification

Solids content and viscosity of evaporating "old coating removal waste" were determined as a function of concentration factor at temperatures expected during evaporation with hot gas sparge. Maximum solids content was about six volume percent when air only was used as sparge. The solids settled rapidly until a concentration factor of about 3.3 was reached and then tended to remain suspended because of increased viscosity. Viscosity of the solutions increased rapidly in the final stages of concentration and was in the 50 to 100 centipoise range at termination of concentration.

The addition of sufficient carbon dioxide to the coating removal waste to neutralize the free caustic increased the maximum solids content to 30 to 40 volume percent. Viscosity of the concentrated solution also increased and was in the 300 to 400 centipoises range at termination of concentration.

Blending of coating removal waste with other wastes for in-tank solidification is currently being studied. Initial results with synthetic solutions indicate stored Uranium Recovery Process wastes may not require blending and can be solidified by evaporation without additives. Solidification of simulated Purex stored waste supernatant liquid (TK-103-A) was achieved without blending. However, the solids content of the terminal hot solution was high and blending at a 1:1 volume ratio with coating removal waste was definitely beneficial. Organic wash waste is difficult to solidify completely by in-tank concentration procedures; blending with other wastes such as coating removal waste appears necessary if complete solidification is to be attained.

Other studies continued in the 4-foot diameter by 15-inch deep hydraulic and thermal model of the in-tank solidification process. In a second run, 313 Building coating waste was adjusted to approximately Purex Plant composition by addition of sodium nitrate, and concentrated in the model by a factor of 5.25. At this point, circulation and heating ceased and the entire contents of the model solidified on cooling. The cake remained dry and solid with no indications of hygroscopicity. Solids deposition was remarkably uniform (6 to 7-1/2 inches in depth across the tank bottom) with a slight ridge located approximately 60 percent of the tank radius from the center point.

A third run is in progress in the scale model using 313 Building coating waste and employing time lapse photography to follow the course of solidification. Chemical composition of the waste appears to have a major bearing on the nature of solidification process.

<u>Ultrasonic Depth Sounding of Waste Storage Tanks</u>

Development of instrumentation and techniques for determining sludge layer and tank bottom contours continued. Purex Waste Tank 105-A was used for testing; no sludge layer was present. The speed of sound in this waste tank supernate at 57 C was determined to be 5760 feet per second, using a one-megacycle, high temperature barium titanate transducer. The time for a pulse to travel from the transducer to a metal plate and back to the transducer was measured; remote removal of the plate then permitted measurement of the distance to the tank bottom. Use of lower frequency ultra-sound is recognized as having higher penetrability of sludge layers, and procurement of appropriate electronic equipment is underway.

Cesium Removal from Formaldehyde-Treated Waste

Additional inorganic exchangers were tested for extraction of cesium from acid FTW waste. Linde AW-300 and AW-400 were found to be relatively ineffective, probably due to the acidity of the waste. Preheated clinoptilolite was found to be about the same as unheated clinoptilolite.

Cesium extraction was accomplished with several exchange materials from synthetic FTW waste that had been adjusted to pH 3.5 with sodium hydroxide and sodium citrate. This composition simulates the FTW after treatments used in solvent extraction for strontium and rare earths and is equivalent to 130 gallons of waste/ton of uranium. The column volumes to 50 percent breakthrough were 136, 131, 133, 171 and 280 for clinoptilolite, preheated clinoptilolite, Zeolon, AW-500 and AW-400, respectively. The kinetics of the preheated clinoptilolite, AW-500 and AW-400 were better than for the other two materials, indicating that about twice the flow rates could be used with these materials than with unheated clinoptilolite.

A scouting experiment was performed to determine the sorption of fission products by AW-500 from a simulated FTW solution containing citrate ion at pH 3.5. Preliminary results indicate sorption of some Zr-95 - Nb-95 in addition to Cs-137. Sorption of Ce-144, Sr-89,90 and Ru-103,106 was negligible.

The source of an insoluble material in ammonium carbonate eluates from cesium loaded clinoptilolite was investigated. The insoluble material, which is largely silica, represents less than one percent of the volume of the eluate from a column of clinoptilolite that has been loaded with cesium from simulated FTW. The solids were found to be produced by contacting the clinoptilolite with dilute acid followed with ammonium carbonate. Analysis of the acid after contacting the clinoptilolite disclosed the presence of trace aluminum, indicating a solvent effect of the acid on clinoptilolite. The silica and other insoluble material are removed by the action of the ammonium carbonate.

Elution of Cesium and Strontium from Zeolites After Heating

Studies continued on the elution of cesium and strontium from inorganic exchangers which had been heated to 600 C after loading. Twenty column volumes of 2 $\underline{M}$ $NaNO_3$ removed 86 percent of the cesium from Linde AW-500, 96 percent from clinoptilolite, and 38 percent from Decalso. A typical trivalent cation salt solution, 2 $\underline{M}$ $La(NO_3)_3$, removed 22 percent of the cesium from Decalso in 20 column volumes.

One molar nitric acid solution removed 90 percent of the cesium from AW-500 and 99 percent of the strontium from Linde 4A in 20 column volumes.

TRANSURANIC ELEMENT AND FISSION PRODUCT RECOVERY

Recovery of Neptunium and Plutonium from Purex FTW Solutions

Work was continued toward development of a flowsheet for recovery of neptunium from Purex plant FTW solutions. Studies with synthetic FTW of estimated 1965 composition showed that the presence of 0.01 M EDTA, HEDTA or citric acid do not impair seriously the extraction of Np(IV) by 0.04 M D2EHPA-Soltrol at an ℓ/v of 10. These complexants have been shown previously to inhibit extraction of tracer Zr-95 and/or Nb-95. Other studies showed that Np(V) in FTW solutions can be reduced to Np(IV) by hydrazine alone--ferrous sulfamate is not required. Reduction by either hydrazine alone or hydrazine-ferrous sulfamate is more rapid at 60 C than at room temperature. About 97 percent of the Np(IV) extracted into 0.04 M D2EHPA-Soltrol is stripped by a single contact of the organic with an equal volume of 0.25-0.5 M oxalic acid. Coextracted zirconium, niobium and iron do not appear to strip as efficiently as neptunium; some decontamination of the niobium during stripping, therefore, may be possible.

Work was initiated to explore the possibility of recovering both neptunium and plutonium from FTW by single batch contact with a D2EHPA solution. Batch contact studies were performed with a synthetic "1965" FTW made 2.4×10^{-5} M in Pu(IV). A single batch contact with one-tenth volume of 0.04 M D2EHPA-Soltrol extracted over 95 percent of the Pu(IV). The presence of 0.01 M EDTA or citric acid did not significantly decrease plutonium extraction. A single contact of the organic phase with an equal volume of 0.25-0.50 M oxalic acid removed 96-98 percent of the plutonium.

A batch contact was made between centrifuged Purex plant 1WW (used in lieu of currently unavailable FTW) and one-tenth volume of 0.04 M D2EHPA-Soltrol. Neither a valence adjustment step nor addition of complexing agents was made in this first run. Phase disengagement by gravity settling was excellent and no solids were noted at the interface (considerable interface scum was observed in similar experiments with synthetic FTW and tracer Zr-Nb-95). Extraction of zirconium-niobium was less by a factor of about five than has been obtained with synthetic FTW and ORNL Zr-Nb-95 tracer. Other analyses are not yet available.

Solvent Extraction of Cesium from Purex Stored Waste Supernatant

Miniature pulse column runs have been made to test the extraction of cesium from Purex plant stored waste supernatant liquid by dipicrylamine (DPA)-diluent solvents. The most promising operating conditions tested to date involved simulated supernatant liquid (103-A) diluted with an equal volume of water as feed and 0.01 M DPA in 50 percent nitrobenzene-50 percent tetralin as solvent. At an ℓ/v of one, cesium extraction was about 98 percent. Hydraulic performance was satisfactory. These runs are preliminary to a test with actual 103-A supernatant liquid at a later date.

Solvent Extraction of Fission Products from Purex FTW

Additional mixer settler extraction runs were made with feed solutions prepared from simulated 1965 Purex FTW to complete demonstration of Study Flowsheet No. 1. Citric acid is used both as complexant and buffering agent. Runs were made with the feed 0.19 M in citric acid and at pH 4.0. Extractant was 0.2 M D2EHPA-0.2 M SBP-Soltrol. Scrub was 0.25 M citric acid at pH 2.9. Feed to scrub to extractant flow ratios were 1.0:0.21:0.98. With seven extraction and five scrub stages, decontamination factors for ruthenium, chromium and uranium were 1.3×10^4, 1300 and 100, respectively. Nickel was below spectrographic detection limit in the organic.

Organic from the above runs was used in mixer-settler partitioning runs. Partitioning agent was 0.03 M HNO_3 at a volume flow one-fourth that of the organic. Seven stages were used. Aqueous product pH was 1.8; strontium loss (not stripped) was 0.3 percent; the cerium decontamination factor was about 500. Partitioning with dilute nitric acid rather than citric acid appears promising.

Cesium Absorption on Duolite C-3

A fixed bed absorption test was completed for the removal of cesium from a solution simulating the supernatant liquid in the Purex 103-A Tank. A feed stock composed of 0.1 M NaOH, 4.5×10^{-4} M CsOH, and traced with Cs-134 was processed in a 33-inch high by 4-inch diameter bed at a rate of 3.13 liters per hour per liter of resin. The bed of 16-50 mesh Duolite C-3 resin processed 5 and 15 bed volumes to 5 and 50 percent cesium breakthrough, respectively. These values compared favorably to other experiments in smaller beds.

Isolation of Cesium by Volatilization of Ammonium Salts

A cesium isolation method has been proposed involving the elution of cesium from an ion exchange resin with a volatile ammonium salt and the subsequent separation by evaporation. Two tests of the evaporation step were made using cesium chloride and ammonium acetate or ammonium carbonate. The solutions were two molar in the ammonium salt and 0.06 to 0.12 molar in cesium chloride.

In both runs, essentially complete volatilization of the ammonium salt was achieved. The weight of the residue approximated the weight of cesium chloride added. No difficulties were encountered with the ammonium acetate. The temperature of the boiling solution rose steadily from 103 to 160 C. With ammonium carbonate initial boiling (or reaction) was observed at 74 C and the final temperature was 100 C. The ammonium carbonate solution foamed badly. Also, crystals of ammonium carbonate appeared in the condenser. The crystals were removed by reducing the cooling water flow until the cooling water outlet temperature reached 50 C.

Product Forms

Studies of strontium carbonate fluxing with lithium fluoride were completed. Fifteen weight percent LiF forms a stable melt at 900 C. $SrCO_3$ that had been precipitated, filtered, and dried in the prototype loading station equipment was heated in a closed bomb to determine the pressure build-up with temperature. The pressure reached 50 psig at 900 C but returned to zero when the bomb cooled to room temperature. This pressure rise is conceivably caused by partial decomposition of $CaCO_3$ impurity.

Studies continued on the use of molecular-sieve type synthetic zeolites as a means of packaging cesium and strontium. Design was finished, and fabrication begun on equipment to be used in the study of transient heat effects during hot gas dehydration of a bed of fission-product-bearing zeolites. The unit will provide simulated internal heat generation, in a container 3-feet long by 8-inches in diameter, at up to 5100 Btu/hr per cubic foot.

Work continued on the design of a satisfactory test to determine the relative attrition resistance of various forms of synthetic alumino-silicate materials. A test is sought which will reflect conditions expected during shipment. Testing of dry sieves is complicated by the moisture absorption characteristics of the material, and by difficulty of removing the attrition products from the surface of the samples.

Fission Product Binding Materials

Search for materials which will fuse with the synthetic zeolites Linde 3-A and Linde 13-X was continued. Lead borohydrate, calcium iodide, barium peroxide, sodium peroxide, calcium carbonate, zinc oxide, lead metal and sodium hydroxide were tried at levels of from 2 to 25 weight percent. None of these shows promise as a fusing agent at temperatures up to 900-1000 C. Preliminary experiments with mixtures of lithium salts and sodium hydroxide are promising and will be continued.

Thermobalance data for Linde 3-A indicates that it, as received, contains three molecules of water per molecule. Weight loss begins at a temperature about 70 C and is essentially complete at 250 C.

Remote Welding

Studies of remote welding continued with 25 samples welded under various conditions to check tendency for blowout. The following is a summary of the blowout tests:

1. All samples could be closed with 1-inch overlap of weld joint without any blowout in the weld area.

2. As the void area under a raised flush face joint cap is increased from 3/8-inch to 1-1/2-inch in depth, the overlap before blowout changes from one revolution overlap to 1/4 revolution overlap.

3. No blowouts could be detected in the raised V groove samples for 1-1/2 revolutions overlap for void depths of 5/16 to 1-1/2 inches.

4. Two samples that were sand blasted prior to welding had more impurities in the weld puddle than the samples that were machined and degreased prior to welding.

Assistance to Purex Plant Head-End Fission Product Recovery

The work reported in the last monthly report and in Invention Report HW-74036 on the use of peroxide to effect the separation of rare earths from strontium was continued. Significant new findings included: (1) the observation that copper(II) has the same effect as iron, but that the use of copper in addition to iron does not enhance the separation and may even be detrimental;

(2) after addition of peroxide and digestion at pH 2, the solution can be adjusted to pH 1 without destroying the complexing action on cerium. Optimum point in the process for addition of peroxide is after initial pH adjustment and before addition of lead precipitant; (3) a series of tartrate "decomposition products" were tested and found to have no effect. These included: dihydroxytartaric acid, glycolaldehyde, glyceric acid, glycolic acid, glyoxal, ethylene glycol, glyoxylic acid, ketomalonic acid, tartronic acid, and oxalic acid; (4) addition of peroxide was found, at least under conditions tested, to inhibit the "second precipitation peak" in tartrate-complexed 1WW. Without peroxide, copious iron precipitation (ferric sulfate) occurs when insufficient tartrate is used. Use of somewhat more tartrate eliminates ferric sulfate precipitation, but a certain critical concentration results in a massive ferric tartrate precipitate, which is avoided by using even more tartrate, or by adding peroxide. It is hypothesized that peroxide prevents the tartrate precipitation either by destroying tartrate or by reducing the iron to the ferrous state.

At the end of the month, B-Cell experiments had been initiated to test the peroxide flowsheet with full level plant feed.

Removal of Cesium from Purex Supernate by Ferrocyanide Precipitation

Hot-cell runs to perfect a process for cesium removal from Purex tank farm supernate were continued. A total of five runs with full-level feed have been completed. No difficulty was encountered in achieving very nearly quantitative cesium removal (98-99 percent); however, silver carbonate metathesis (with solid silver carbonate) of the cesium nickel ferrocyanide was unsatisfactory, possibly because of difficulties experienced in solids transfer. A run using silver nitrate in dilute nitric acid as metathesis agent has been performed, but analyses have not been completed. Mercuric salts will also be evaluated as metathesis agents.

Cesium Solvent Extraction

Laboratory work was continued during the month on the selective scrubbing of sodium and on the determination of the aqueous-phase solubilities of dipicrylamine (DPA) and nitrobenzene.

Results were reported last month on the use of ammonium salts to scrub sodium from cesium-bearing dipicrylamine-nitrobenzene solutions. Experiments this month showed that weak acids can be used in the aqueous scrub to accomplish the same result. The weak acid provides a very small but controlled hydrogen ion concentration (i.e., constant pH) such that cesium is extracted while sodium is stripped. Buffer action of the slightly dissociated organic acids renders the process operable. Although the separation factor between cesium and sodium (about a hundred) is not as great as in the ammonia scrubbing case, it is adequate for operation in counter-current contactors (such as pulse columns or mixer-settlers). Additional details of both the ammonia and acid processes are given in Invention Report HW-74130.

The solubility of both dipicrylamine and nitrobenzene in the aqueous streams of the cesium solvent extraction process is of interest both from a reagent loss and a waste storage standpoint. Although the aqueous distribution of both compounds appeared visually to be quite low, quantitative measurements were desired. This required the development of analytical methods for the determination of small traces of nitrobenzene and dipicrylamine in both acidic and alkaline solutions. The method adopted is to extract the nitrobenzene from the aqueous sample with cyclohexane (which does not extract DPA) and to determine the nitrobenzene content of the organic phase spectrophotometrically by measuring the 250 mμ nitrobenzene peak (molar extinction coefficient about 8.3×10^3). After removal of the nitrobenzene, DPA is extracted with hexone and similarly measured. Molar extinction coefficient of DPA extracted from acid solution into hexone is $3 \times 3 \times 10^4$, at about 415 mμ.

The solubility of nitrobenzene in water was found to be 1.9 g/l, in good agreement with literature values. Its solubility is higher in nitric acid solutions and increases monotonically with acidity. Thus, the nitrobenzene solubility is about 6.5 g/l in 5 M HNO_3 and 18.5 g/l in 10 M HNO_3. In alkaline salt solutions, such as 103-A supernate, the solubility of nitrobenzene decreases steeply with increasing salt concentration, reaching a value of only about 0.15 g/l in full-strength 103-A supernate. Dipicrylamine concentrations, in aqueous solutions in contact with 0.01 M DPA-NB, were even lower. Preliminary results indicate DPA values of about 5 micrograms/ml in 1965 salt waste and about 2 micrograms/ml in 103-A supernate. These values correspond to aqueous DPA losses of less than 0.1 percent. A report will be issued on the solubility measurements as soon as the work is complete.

In other analytical development, preliminary data indicate that satisfactory analyses of DPA in nitrobenzene can be obtained by potentiometric titration of an aqueous phase in equilibrium with the organic phase.

Ion Exchange Studies

The development of an anion exchange process for the recovery of plutonium from RMC Line Task I oxalate supernates was reported last month. An experiment was initiated this month to determine the chemical resistance of the recommended Permutit SK resin to hot 10 M nitric acid; 10 M HNO_3 is being circulated continuously through a column jacketed to 58 C. After two weeks, no change is apparent in the appearance of the resin. The experiment will be continued for at least six weeks. At the end of that time the plutonium absorption capacity will be determined as a measure of resin damage.

In other ion-exchange experiments, a commercially available synthetic zirconium phosphate cation exchanger was found to exhibit cesium capacities somewhat higher than those of the alumino silicates. The zirconium phosphate resin elutes very easily with 0.1 M HNO_3, possibly a significant advantage.

Isotopic Fission Product Analyses

The isotopic analysis of cesium and rubidium, recovered from 103-A supernate by DPA extraction in the hot cells, was performed by mass spectrometry. Purpose was to learn how much the cesium had been diluted with natural cesium and also to explore the feasibility of determining both cesium and rubidium by isotopic dilution techniques. The results obtained:

Isotope	Atomic % Found	Atomic % Theoretical
Rb-85	29.23 ± 1.83	30.73
Rb-87	70.77 ± 1.83	69.27
Cs-133	46.57 ± 0.85	46.71
Cs-135	16.73 ± 0.11	11.00
Cs-137	36.69 ± 0.68	42.39*

*Neglects decay of Cs-137

These results indicate no significant dilution with natural cesium or rubidium and also that quantitative analysis of these elements is possible by diluting with a known amount of cesium and/or rubidium followed by redetermination of the spectra.

The isotopic composition of the strontium recovered in the recent CPD Hot Semiworks strontium purification run was also determined by mass spectrometry. Strontium-90 content was 53.2 atomic percent versus 56.1 percent in earlier product and 60 percent theoretical, implying increased dilution with natural strontium.

EQUIPMENT AND MATERIALS

B-Plant Remote Connector Gaskets

Evaluation was completed on several candidate gasket designs for use in B-Plant flat-face remote connectors to resist the extreme radiation fields anticipated in fission product and waste processing. The recommended replacement is an asbestos-filled, spiral-wound gasket specially prepared for this service.

Purex Concentrator De-Entrainment

A scaled-down section of a proposed Purex concentrator baffle compartment has been fabricated for investigation of de-entrainment efficiency. The baffles are composed of vertical 2-1/2-inch by 2-1/2-inch angle irons set in chevron fashion two inches on centers and placed normal to vapor flow. This mock-up is operated by blowing an air-water mixture against the baffles at vapor velocities similar to those expected in the Purex concentrator. At a vapor velocity of 20 ft/sec in the slots, the removal of the gross entrainment is effected. Fine mist (generated by a spray nozzle) is not appreciably removed by this arrangement. Improvements to the equipment are being made to extend the operating range to 30 ft/sec in the slots.

Corrosion of Titanium by Purex 1WW Containing Fluoride

A titanium tube has been exposed as a condensing surface for the vapor phase from a boiling synthetic Purex 1WW containing 0.04 $\underline{M}$ HF. Iron and aluminum content were in the range normally expected in 1WW, 0.4 $\underline{M}$ and 0.125 $\underline{M}$, respectively. Fluoride was ten times that normally expected. The exposure period was about 3 months. Visible corrosion of the titanium occurred where condensation took place. Metallographic examination of the area to estimate a rate is in progress. These results emphasize the need to prevent volatilization of fluoride (by adequate complexing) if titanium tube bundles are used in concentrators treating fluoride-containing wastes.

Corrosion by Perchloroethylene

Serious corrosion has been observed in the 234-5 Building perchloroethylene cooling system used with induction heaters. Samples of 304-L stainless steel, copper and 1020 mild steel were exposed to pure perchloroethylene and perchloroethylene containing one volume percent water. Significant corrosion was observed only on 1020 steel at the vapor-liquid interface of the perchloroethylene-water solution. A sample of 234-5 Building perchloroethylene contained about 0.5 volume percent water.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Plutonium, Uranium and Rare Earth Separations - The past two months' laboratory investigations of plutonium-uranium-rare earth separations have given the results described below:

1. Precipitation of PuO_2 (by O_2-Cl_2 sparge) from a melt containing one weight percent plutonium left about the same concentration of soluble plutonium in the melt as precipitation from a 0.1 weight percent plutonium melt (i.e., between 0.012 and 0.03 w/o Pu). This indicates that previously reported plutonium recoveries across a precipitation step were as low as they were (< 90 percent) because of a solubility effect.

2. The possibility of exercising some measure of control over plutonium separation during a "partition"-type UO_2 electrodeposition, by control of cell voltage and/or current density, was demonstrated. At a current density of 0.4 amp.cm^{-2} (applied potential of about -1.0 volt vs. a Ag/AgCl reference), UO_2 was deposited from a 2.6 LiCl-KCl melt at 600 C with a plutonium decontamination factor of about 50. At one-half the current density and potential, the decontamination factor averaged about 100.

3. Europium and promethium tracers were found to differ markedly in the extent of their separation from plutonium and uranium during PuO_2 precipitation and UO_2 electrodeposition, respectively. For example, the decontamination factors measured for PuO_2 precipitation from 2.6 KCl-LiCl at 575 C were from 60 to 110 for europium and 600 for promethium. For UO_2 electrodeposition from the same type melt at 600 C, the decontamination factors were 135 for europium and 1200 for promethium.

4. The effects of the presence of 0.3 w/o iron(III) chloride upon UO_2 electrodeposition and PuO_2 precipitation from 2.6 LiCl-KCl at 575 C were investigated. In the case of the electrodeposition

of UO_2, the iron lowered the current efficiency from 70 to 60 percent and lowered the measured decontamination factors for plutonium and europium by about 20 percent. The decontamination factor for iron (uranium basis) was 170. In the case of the PuO_2 precipitation, the iron reduced the europium decontamination factor by a factor of two. The separation factor for iron itself in the precipitation step was 4.3.

Electrochemistry of Plutonium in Molten Chloride Salt Solutions - Recent electrochemical studies of the reduction of plutonium(III) chloride to metal in molten chloride salt solutions have embraced cathode polarization and chronopotentiometric measurements under a variety of conditions. The work has given the following results:

1. Using a micro tungsten cathode in a KCl-LiCl eutectic at 600 C, voltammetric measurements gave added proof that the reduction of Pu(III) to metal occurs in two steps. The measured potentials for the second step at plutonium concentrations of 9.37 and 29.33 millimolal were 1.937 and 1.904 volts, respectively, vs. a Ag/AgCl (1 molal) reference electrode. From these data it was calculated that this step involves a two-electron change. Thus, the course of the overall reaction is

$$Pu(III) + e^- \rightarrow Pu(II)$$

$$Pu(II) + 2e^- \rightarrow Pu(0)$$

2. Preliminary cathode polarization measurements were made with a macro plutonium metal electrode over a range of Pu(III) concentrations from 0 to 0.8 molal, in a $BaCl_2$-KCl melt at 700 and 800 C. Aside from determining that reduction of the melt at 700 C occurs at a voltage of 2.2 volts (vs. the Ag/AgCl (0.1 molal) reference electrode), interpretation of the results will have to await additional data.

3. The diffusion coefficient of Pu(III) was determined at 600 C by chronopotentiometric measurements (with a tungsten electrode) for 2.6 LiCl-NaCl, 7.7 LiCl-KCl, and LiCl-1.5 KCl. The diffusion coefficient was essentially constant at about 3.0×10^{-5} $cm^2 sec^{-1}$ for the three systems.

X-Ray Diffraction Studies of Irradiated UO_2 - X-ray diffraction techniques were used to study the effect of neutron irradiation upon a UO_2 crystal lattice. Lattice constants were measured for two segments of a single UO_2 crystal, one segment unirradiated and the other exposed to 5,000 MWD/T at 400 C. They showed the irradiated UO_2 to have an expanded lattice with a spacing of 5.4723 Å, compared with 5.4710 Å for the unirradiated UO_2. The extrapolation method was used with the X-ray spectrometer, rather than the much lengthier photographic procedure. In spite of the fact that the irradiated sample was reading 5 R/hr from the front face of a special lead cell, and that the radiation-induced defects in the crystal caused a marked decrease in diffraction intensity, which had to be measured on top of a greatly increased background count, excellent results were obtained by averaging the results from seven scans in the back-reflection region. The method appears to be accurate to about ± 0.0001 Å.

Heating with Alternating Current - As a part of the efforts to develop suitable frozen wall techniques, alternating current heating studies were continued using four-liter carbon pots and uranium-free KCl-LiCl melts. Since earlier studies had demonstrated the ability of AC heating to maintain a satisfactory melt, recent efforts were devoted to means of initially melting the high electrical resistance solid salt.

Two methods have been used successfully to initiate meltings. The first is to strike an arc at the surface of the frozen melt between a movable electrode and an electrode frozen in the solid salt. A puddle of melt forms at the surface and then gradually enlarges until it contacts the pot wall. At that time the AC current is switched from the movable electrode to the pot wall. Simultaneously a quartz shroud originally placed on the fixed electrode is allowed to progress down through the melt continuously increasing the path length for the current as the remaining frozen salt melts. With this method and 1200 watts of power, approximately 15 minutes were required to melt ten pounds of salt.

A second method which does not require a movable electrode is to start with the salt damp enough to sustain an adequate current between the center electrode and pot wall. Sufficient moisture is adsorbed by the hydroscopic salt by exposure to the air for two or three days. This type of startup requires about 30 minutes with 1200 watts of power but subsequent drying of the melt is a problem.

Salt Cycle Process Instrumentation - Development of instrumentation for control of the Salt Cycle Process included work on a molten salt bath liquid level indicator and testing of the microelectrode voltage control and power supply.

The liquid level indicator, with a range of eleven inches, is comprised of a quartz float buoying a rod whose elevation is indicated by a Foxboro Dynalog recorder, using a rotameter coil as a position transducer.

Testing of the microelectrode voltage control and power supply has shown that power transistor heat sinks are necessary for realization of the maximum current demand of five amperes. The voltage control performs very satisfactorily, giving a maximum error of 0.015 volts over the range from 0.05 to 10 volts with an output current of two amperes. Testing of an operational amplifier as a high impedance differential voltage measuring device to determine cathode-to-reference electrode potential has not proved successful.

RADIOACTIVE RESIDUE FIXATION

Synthetic Zeolites

Equilibrium constants (K_c) and ion exchange capacities were determined experimentally in the systems strontium-sodium, cesium-sodium, and cesium-potassium for the zeolites Linde 4AXW, Linde 13X, Linde AW-300, Linde AW-400, Linde AW-500, Zeolon and clinoptilolite. Exchangeable cation fractions in the equilibrium solution were varied from 0.2 to 0.00001. The "equilibrium constants" were not constant at cation fractions in the equilibrium solutions of greater than 0.0001 but were a function of the fraction of zeolite loaded.

A method was devised to use the K_c to predict a zeolite column load for a given influent composition in the above systems. For example, with an influent containing 0.002 $\underline{N}$ Sr^{++} + 0.01 $\underline{N}$ Na^+ Linde 13X column capacity was 3.1 meq Sr^{++}/g, while a capacity of 3.3 meq Sr^{++}/g was predicted from the K_c curve. Linde 4A gave a column capacity of 3.4 meq Sr^{++}/g with the same influent solution, while 3.3 meq Sr^{++}/g was predicted.

Cesium-sodium K_c curves predicted a column capacity of 0.75 meq Cs^+/g for clinoptilolite with an influent containing 0.01 $\underline{N}$ Cs^+ + 1.0 $\underline{N}$ Na^+, while actual column capacity was 0.73 meq Cs^+/g. With 0.01 $\underline{N}$ Cs^+ + 0.5 $\underline{N}$ Na^+, clinoptilolite yielded a capacity of 1.01 meq Cs^+/g, while 1.02 meq Cs^+/g was predicted.

Condensate Treatment

Satisfactory removal of cesium, strontium, zirconium and cerium from Purex Tank Farm condensate waste has been accomplished with various sulfonated polystyrene cation resins when the pH of the feed has been adjusted to about 4. Work is continuing, however, to improve ruthenium removal and to understand better the ruthenium removal mechanism so that its removal can be predicted.

Clinoptilolite Beneficiation Studies

Petrographic examination of clinoptilolite from Hector, California, disclosed that the clinoptilolite crystals are uniformly in the micron-size range, up to a maximum seen of about 10 microns long. Impurities, consisting dominantly of clay (montmorillonitic), carbonate, unaltered glass, halite, biotite, feldspars, and magnetite, are up to a millimeter in diameter but are disseminated throughout the rock. The lack of appreciable agglomeration of impurities will probably preclude easy upgrading of clinoptilolite by simple mechanical beneficiation procedures. Mechanical release of impurities will require find grinding with consequently more expensive separation techniqies and reagglomeration of the clinoptilolite that would not be required if simpler crushing, grinding and screening alone were sufficient. Further studies are underway to relate laboratory-determined chemical characteristics of the clinoptilolite to visible physical and mineralogical characteristics.

Electrostatic Bubble Scrubber

The bubble scrubber which will be used in the A-Cell set-up was assembled and tested, using the power supply which will be employed in the cell. Testing was carried out both with the off-gas from "cold" calciner runs and with a sub-micron smoke generated by burning magnesium ribbon. Performance was very satisfactory and equalled or exceeded design expectations. With calciner-generated dust, the incremental decontamination factor across the bubble scrubber was a hundred or greater (limit of the analytical method). Using magnesium oxide, the decontamination factor was measured as a function of applied voltage and gas flow rate. At design flow rate (1.3 ft^3/min) and no applied voltage, i.e., operating as a simple scrubber, the decontamination factor was only 2 to 3. As the applied voltage was increased to 20 KV, the DF increased uniformly to a value of 235. Decreasing the flow rate increased the DF further, to a value of over 1000 (analytical limit) at a flow rate of 0.33 ft^3/min. Dust loadings in the existing gas were only 100 to 15 micrograms/ft^3 at the two flow rates.

These values are in the "ultra pure" range which is very difficult to achieve by conventional methods.

Radiation Resistant Elastomer

Two samples of ethylene propylene rubber (EPR) were irradiated in the cobalt source to an estimated dosage of 2×10^8 R and then examined and the hardness, elongation, and tensile strength measured and compared with unirradiated controls. One rubber in particular was little affected and appears very promising, particularly for applications where inertness to nitric acid and high temperature is important (the chemical characteristics of EPR are rather similar to polyethylene).

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

Significant differences were noted in vertical ground water potentials measured in well 699-69-45 north of Gable Mountain. The maximum potential, six feet greater than the minimum, occurred near the water table. Potential differences in this well were higher than those measured in the other two project wells which are similarly equipped with piezometers. These measurements assist in explaining anomalies in ground water contours evident in this region. Farm wells which terminate only slightly below the water table have consistently shown higher ground water elevations than well 699-69-45 previous to the installation of the piezometers.

Graphic transfer of the airborne magnetometer data was completed. Additional ground check points noted on the rectilinear chart were added to the flight line overlay to permit a more precise determination of aircraft speed and ground position. The magnetic values are now being transferred to an enlarged flight line overlay in preparation for constructing an equal anomaly contour map which will be correlated with known buried basalt elevations.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Carbon-14 Dioxide in Reactor Stack Gas

A series of continuous samples of reactor stack gas taken over a two week period from the two K reactors has been analyzed for $C^{14}O_2$ content. They indicate a consistently higher release rate for KE over KW by a factor of about 3, which was expected on the basis of the difference in pile gas composition. The KE release rate was 0.068 curie per day and the KW release rate was 0.024 curie per day.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition in Conduits

The experimental data obtained locally and data from the experiments of S.K. Friedlander[1] are being reviewed to determine whether an improved correlation can be shown. Turbulent gas flow equations will be applied with empirical terms derived to account for particles not following the motion of the gas. A FORTRAN program is being written to facilitate the calculations.

Columbia River Sediments

McNary Dam reservoir sediments were sampled at stations 600 yards, 2 miles, 4 miles, and 7 miles from the dam. Five equally spaced samples were taken across the river at each station and analyzed for Co-60 and Zn-65 content. Except for the station seven miles upstream, the Co-60 and Zn-65 content of the sediments on the Oregon side was greater than that on the Washington side by as much as a factor of two. Further studies will be needed to determine why this difference occurs, but it is perhaps related to the entrance of the Snake River at a point about 35 miles upstream.

W. H. Reas

Manager
Chemical Research and Development

WH Reas:cf

1. Friedlander, S.K., H.F. Johnston, Ind. Eng. Chem., 49, 1151 (1951)

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BIOLOGY OPERATION

A. ORGANIZATION AND PERSONNEL

Dr. Frank Mraz, an exchange scientist from the UT-AEC Agricultural Research Project at Oak Ridge, Tennessee, completed a one-year assignment in Biology on May 31.

Dr. Ahmet Noyan, an International Atomic Energy Agency Fellow from the University of Ankara, Turkey, completed a one-year assignment with Biology on May 14. He will study for six months in the Department of Physiology at the University of California in Berkeley before returning to Turkey.

Dr. Jack Van't Hof, a Biological Scientist in the Plant Nutrition and Microbiology Sub-Section, terminated on June 14, 1962, to accept employment with Brookhaven National Laboratory.

Miss E. Jane Coleman, a Biological Scientist in Biological Analyses, terminated on June 8, 1962.

Patricia L. Hackett and Beatrice J. McClanahan returned from educational leaves of absence to join the Metabolism and Experimental Animal Farm Operations, respectively, on June 15.

Capt. John W. Cable, an Air Force Veterinarian, on tour of duty with the Experimental Animal Farm, received orders to report to the University of Rochester AEC Project, Rochester, New York, on June 8, 1962.

Mr. Ward Whicker, graduate student from Colorado State University, Fort Collins, Colorado, was assigned on June 25 to the Radioecology Operation as a part of an AEC grant and program to disseminate knowledge and techniques in radiobiology.

GENERAL

The Symposium on the Biology of Transuranic Elements, which was largely organized and arranged for by members of the Metabolism Operation, was held May 28-30. Some 70 visiting scientists including 10 from out of the country (England, France, West Germany, Canada) met with an approximately equal number of Hanford Laboratories' scientists. Thirty-five papers were read and discussed, including 13 papers by Hanford Laboratories' personnel. These papers are now being edited for publication as a special issue of Health Physics which is scheduled to appear in October.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

Monitoring of effluent from the 100-KE reactor with fingerling chinook salmon was terminated on June 18. The results are summarized as follow:

Mortality and Growth of Chinooks Reared in
Various Concentrations of Effluent Waters

Treatment, % Effluent	Mortality				Growth	
	Dead	Live	Total	% Dead	Mean Fork Length (mm)	Mean Body Weight (g)
0	18	552	570	3.2	62.9	2.68
3	18	552	570	3.2	62.7	2.57
5	48	522	570	8.4	60.2	2.26
7	106	464	570	18.6	57.8	1.99
	190	2090	2280			

The mortality data clearly show the toxic effect at the 5 and 7 per cent concentration, but not 3 per cent. Statistical difference was found in the average length due to treatment at the 0.01 level of significance. Growth depression in length due to effluent is inferred for the 7 per cent, but no statistical difference in length was found between the 0 per cent, 3 per cent, and 5 per cent treatment levels. Statistical analysis of the growth in body weight has not been completed, but the average weight at end of test also indicates growth depression due to treatment.

Columnaris

The effect of growth temperatures on the virulence of C. columnaris was investigated.

Chinook fingerlings held at 18.5 C were exposed to organisms which had been cultured at 20, 25, and 30 C. Optimum virulence was found at 25 C, organisms cultured at 20 and 30 C were considerably lower in virulence.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Strontium

Three additional female swine, nine months of age, were started on the 125 μc Sr^{90} per day level of feeding. This brings to 6 the number of animals started as young adults on this level and should provide for ~18 F_1 and F_2 generation females to be held for lifetime feeding at this level. The smaller group of animals was originally provided for at this level in anticipation of severe effects in the offspring during early life. Since the F_1 generation animals are showing only minimal effects at ~18 months of age, it is apparent that this level of feeding should be of considerable importance for observing late effects. The additional animals and their offspring should provide adequate numbers for comparison of effects with the large numbers of animals at lower feeding levels. (At the 125 μc Sr^{90} per day feeding level the estimated radiation dose rate to the skeleton is ~20 rads per day for animals 6 months of age and older.)

Studies are in progress to attempt to demonstrate increased titers of "erythropoietin" (a hormone found in plasma which stimulates erythropoiesis) in animals whose bone marrow is being chronically irradiated due to ingestion of Sr^{90}. Studies to date have been made utilizing starved rats which receive injections of plasma from donor experimental swine. Degree of erythrocyte production is based on incorporation of Fe^{59} into circulating red cells. Starved rats are used because their normal erythrocyte production is quite low. Preliminary studies suggest that the plasma of high level Sr^{90} fed animals contains larger amounts of "erythropoietin" than does plasma from low-level Sr^{90} fed or control swine. Additional studies are planned utilizing polycythemic mice which have recently been reported to be a more sensitive test subject than the starved rat.

Preliminary studies are in progress attempting to demonstrate the presence of a factor in plasma of thrombocytopenic Sr^{90} fed swine that will stimulate production and/or release of platelets in test rats when injected parenterally.

Comparative Toxicity

Female miniature swine injected intravenously with Ra^{226} (6.4 μc/kg) 9 months previously are continuing to show increased levels of blood urea nitrogen and creatinine. Of six surviving animals, 5 injected when 1 year of age are showing minimal changes while one injected when ~three years old is showing greatly elevated levels. Two animals injected at the same time, but when ~4½ years old, have succumbed with severe kidney lesions. On the basis of these few animals, there is a strong suggestion of a correlation of degree of kidney damage with age at the time of Ra^{226} administration.

Milk

Plasma and milk concentrations of Ra^{226} were followed in female sheep following a single intravenous dose of Ra^{226} nitrate in a citrate buffer. Milk concentrations of Ra^{226} increased rapidly and were approximately equal to those in plasma by 2 hours post-injection. By 11 hours post-injection, milk concentrations were being maintained at 7 to 10 times those of plasma.

Plutonium

Three swine injected intradermally with 5 μc Pu^{238}, 1 and 5 μc Pu^{239}, and 0.06 μc Np^{237} were sacrificed three months post-injection. Skin biopsies were taken for radioanalysis which have not yet been completed.

Iodine

Thyroidal uptake of I^{131} was determined in two groups of female Palouse swine following a single oral dose of 50 μc to each pig. Both groups of swine are nearly 5 years old, one group of 6 having received 5 μc I^{131} daily for the past 4½ years, the other group of 5 are their controls.

Thyroid I^{131} was determined by frequent monitoring, utilizing a two-probe Na iodide scintillation detector. Correction for shielding of the thyroid by overlaying tissue was made.

Peak I^{131} uptake in the control animals occurred 12 to 24 hours post-administration while animals which had been fed I^{131} chronically appeared to have their highest thyroid I^{131} uptake slightly earlier.

The peak thyroidal uptake for both groups was approximately 18 per cent of the administered dose.

Although the animals have not been followed long enough to calculate an accurate effective half-life, at one week post-administration it appears that the effective half-life in the control animals is somewhat longer than that of the animals chronically exposed to I^{131}.

These animals will be sacrificed next month and tissues taken for histopathological study and for tissue distribution of I^{131} in the animals receiving I^{131} daily. A correction factor for thyroid shielding by overlaying tissue for these particular animals will be established. This factor will be used to recalculate the data, producing more accurate results than those obtained by using the correction factor based on the animal's weight and previous monitoring experience.

Neptunium

Gastrointestinal absorption in rats of the short-lived isotope Np^{239} was found to be significantly lower than that previously measured for Np^{237}. The per cent absorbed was in the range 0.02 to 0.03 for both citrate and nitrate solutions and for solutions of different valence states. This is a factor of approximately 10 lower than the minimum absorption measured for Np^{237} and does not show the wide variation with chemical form observed within Np^{237}. These differences are no doubt associated with the extreme differences in mass of material administered--400 picograms in the case of Np^{239} and 15 milligrams in the case of Np^{237}. This apparent effect of mass on absorption is an interesting and perhaps highly significant phenomena which will be further investigated when an additional supply of Np^{239} becomes available.

Rats were injected with 3 and 6 milligrams of Np^{237} nitrate to determine effects of chemical toxicity. The animals were sacrificed three days after injection for determination of liver fat, blood glucose, serum cholesterol, and liver cholesterol. Preliminary results indicate an approximately 35 per cent reduction in blood glucose 48 hours after injection of 6 milligrams per kilogram of Np^{237}.

Inhalation Studies

Seven dogs were exposed to $Ce^{144}O_2$ aerosols to test the effectiveness of several agents on increasing lung clearance of the radioactive particles. The materials tested were DTPA (via intravenous injection and via aerosol), pyribenzamine + Quadrinal + atropine, and sulfathiazole + prostigmine.

Rats were exposed to I^{131} vapor to test the effectiveness of simultaneous exposure to non-radioactive iodine vapor on accumulation of I^{131} in thyroid. Results are not complete.

Radiation Protective Agents

Experiment was performed to compare the protective effect of Calmagite and AET following 1000 r whole-body irradiation in rats. The superiority of AET was indicated by the fact that all controls in Calmagite-treated rats were dead by the 8th day while 3 out of 5 AET-treated rats survived for at least 10 days. Following 800 r whole-body irradiation the Calmagite protected all animals while half of the controls were dead by 20 days.

Some very interesting results were obtained from experiments performed in attempts to modify the "secondary" disease syndrome which characterizes lethally irradiated animals protected by inoculations of foreign bone marrow. These experiments employed 2 genetically dissimilar hybrid strains, the LAF and C31 mice. Within 1 day of birth LAF mice were given subcutaneous transplants of spleen fragments from C31 mice. Six weeks after this implant the LAF animals were sacrificed and their bone marrow injected into C31 mice which had received 950 r whole-body X-ray. Seven out of the 8 animals so treated continued to live two months after irradiation. Control C31 animals given 950 r whole-body X-ray died within 6 days. Of eight C31 animals receiving 950 r X-ray plus LAF marrow from animals not treated at one day with the spleen implants, seven died between two and six weeks post-radiation, with the secondary disease syndrome. The reverse experiment, in which 3 LAF animals given C31 spleen implants at birth were irradiated with 950 r six weeks later and treated with C31 bone marrow, resulted in all animals dying with secondary disease syndrome within two to four weeks after irradiation. These results suggest that the foreign bone marrow "sensitivity" responsible for the secondary disease can be eliminated by a treatment of the donor animals soon after birth but that this treatment is ineffective when the treated animals are the recipients. Similar experiments involving the course-transplantation of rat and mouse tissues have not as yet been successful in eliminating the secondary disease syndrome.

Cellular Biology

The influence of X-radiation on cichlid egg survival was studied. Doses as low as 50 r can delay hatching. Increased doses of 100 and 250 r delayed hatching further and eggs receiving more than 500 r did not hatch. The lethal dose for eggs lies between 250 and 500 r, while the lethal dose for X-irradiated cichlids is 1000 to 5000 r.

Cichlid eggs were exposed to 0, 10, 20 and 30 per cent D_2O. No differences were found at 0 and 10 per cent levels of D_2O, however, the higher levels delayed hatching. At the 20 per cent level, the cichlids which hatched did not feed and a 7 per cent survival was noted after 17 days. At the highest level, half of the group did not hatch and no survivors were found after 10 days.

Chloramphenicol and X-ray effects on sporulation of B. cereus were compared. In both instances sporulation was carried out under endotrophic conditions using 5-hour lag phase cultures. Neither 70 kr of X-rays nor 2 μg chloramphenicol/ml had any noticeable effect on sporulation compared to controls.

The influence of growth temperature on radiation sensitivity is being studied using a facultative thermophilic bacterium. Thus far only cultures grown at 55 C have been studied. Preliminary data indicate a multihit X-ray survival curve is characteristic of cells grown at this temperature.

The effect of irradiating under O_2 and N_2 was re-examined using a different method of gassing the cell suspensions. Phosphate leakage from control cells was the same with either gas. With irradiated cells (120 kr) leakage of phosphate was 2.5 times greater under N_2 than when O_2 was used. Potassium leakage from control and irradiated cells was greater (about 2 x's) in N_2 than in O_2. Thus, with phosphate a reverse oxygen effect is obtained whereas with potassium no oxygen effect was noted since the relative differences in observed potassium leakage rates were the same under both atmospheres.

Studies on uptake of non-metabolized sugar (l-sorbose) by yeast have been initiated. These studies are preliminary to studying radiation effects on permeability of larger molecules than potassium and phosphate. It has been found that S.C. 17-7-1 takes up sorbose by a diffusion process.

Plant Studies

Beans from long-term plots which had been fertilized with potassium were harvested. Data are preliminary, however, no effect of potassium on Cs^{137} uptake was observed. Uptake of Cs^{137} which was mixed with soil is greater than the uptake from surface contaminated plots.

Short term experiments have demonstrated chloramphenicol inhibits ion accumulation by barley plants. The duration of the inhibitory effect was examined in young pea plants. The accumulation of Rb was reduced 24 hours and 48 hours after exposure to chloramphenicol. Five days post-exposure the inhibitory effect of chloramphenicol had diminished. Comparison of low-salt and high-salt plants indicate chloramphenicol inhibits or partially blocks the metabolically mediated component of ion accumulation.

Population Dynamics

Fifteen hundred juvenile sea gulls were banded from an island colony near Ringold as a part of the continuing study of dispersal of organisms utilizing the Columbia River.

HA Kornberg, Jr.

Manager
BIOLOGY OPERATION

HA Kornberg:es

C. Lectures

a. Papers Presented at Society Meetings and Symposiums

J. J. Davis. Effects of environmental factors upon accumulation of world-wide fallout in natural populations. 29th Annual Meeting, Western Branch, American Public Health Association, Portland, Oregon. June 6, 1962.

H. W. Casey, R. O. McClellan, W. J. Clarke, and L. K. Bustad. Acute toxicity of neptunium^{237} and its relationship to liver function. Health Physics Society Meeting, Chicago, June 11, 1962 (presented by J. W. Cable).

J. W. Cable, V. G. Horstman, W. J. Clarke, and L. K. Bustad. Effects of intradermal injections of plutonium in swine. Health Physics Society Meeting, Chicago, June 11, 1962.

J. E. Ballou. Neptunium in the rat. Health Physics Society Meeting, Chicago, June 11, 1962.

W. J. Bair, J. P. Herring, and L. A. George, Jr. Retention, translocation, and excretion of inhaled plutonium. Health Physics Society Meeting, Chicago, June 11, 1962.

N. L. Dockum. Autoradiographic localization of Zn^{65}. Microscopy Symposium, McCrone Research Institute, Chicago, June 11, 1962.

b. Off-Site and Local Seminars

None

c. Seminars (Biology)

None

d. Miscellaneous

L. K. Bustad. Atomic energy and agriculture - today and tomorrow. Richland Kiwanis, June 20, 1962.

L. K. Bustad. Atomic energy and agriculture - today and tomorrow. State Vocational Agriculture Instructors - Annual Meeting, Port Angeles, Wash. June 13, 1962.

D. Publications

a. Documents (HW)

None

b. Open Literature

Bair, W. J., and D. H. Willard. 1962. Plutonium inhalation studies. IV. Mortality in dogs after inhalation of $Pu^{239}O_2$. Radiation Research 16, 811-821.

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OPERATIONS RESEARCH AND SYNTHESIS OPERATION
MONTHLY REPORT - JUNE, 1962

ORGANIZATION AND PERSONNEL

R. L. Buschbom returned from educational leave for the summer on June 4, 1962.

OPERATIONS RESEARCH ACTIVITIES

Two research areas are being investigated: (1) the economics of HAPO and the Tri-City Area, and (2) the economics of plutonium recycling in power system reactors. The latter is in conjunction with the fuel cycle analysis program of the Programming Operation.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Fuels Preparation Department

A formal document was issued presenting the results of an analysis, performed on Quality Certification data, which relates dimensional instability of fuel elements to reactor variables. A discussion of future program goals is included in the report, which was coauthored with FPD personnel.

A revision was made to the MERCY program to include estimation of an additional component of variation in the situation in which measurements are made on items assigned "standard" values which may be in error. This additional component is the bias between the average for the instruments and the assigned value.

During June, several sets of measurement data were submitted to MERCY. These included data from the UE-1, the UT-2, and the UT-4 nondestructive tester stations. Consulting assistance was provided in interpreting the program output.

Uncharged fuel elements from a multiple-failure lot have been made available to FPD personnel for extensive testing to determine if measurable characteristics can differentiate the difference in performance. Samples from 8 other lots, 4 ingot and 4 dingot, are included in this testing program to function as controls. Assistance is being given in the analysis of the resulting data.

Analyses of data from production test IP-310, designed to determine the effectiveness of the UT-2 tester in sorting out dimensionally unstable fuel elements, continue.

In re-measuring fuel elements in 100 percent inspection, estimates of tester efficiency and product quality can be found by observing the frequencies with which the cores are accepted on neither, one, or both of the measurements. Estimates of these quantitites were found for a given set of data. It was necessary to assume that the tester made only one type error, that of accepting defective pieces. This tends to err on the conservative side.

Some data were examined to investigate the effects of sleeve diameter on the control of finished outer diameters in the canning of normal alsi-product fuels.

An analysis is being made of some pilot plant data in which duplex bath silicon concentration, lead preheat time, and lead temperature were the independent variables. Several dependent variables descriptive of bonding layer quality are under study, one prime purpose of the test being to determine those variables which best depict quality differences.

Reactivity data (dih values) from several lots of enriched fuel elements were analyzed to see if the method of fabrication ("wet" blending vs. "dry" blending) significantly affects reactivity.

Extensive analyses were made of data from thermocouples embedded in fuel cores being preheated in the hot die sizing process. The generalized IBM routine developed for use in analyzing these data permits estimation of the average, linear, and quadratic longitudinal temperature gradients, as well as the axial effects. Estimates can also be made of the measurement variance and other effects of interest. Further work is being done to indicate where thermocouples should be placed in future studies to obtain the most information. Effects of core preparation and furnace set temperature on time to reach a given temperature were also estimated.

A simple means was derived for estimating net volume change of an irradiated fuel element based on delta OD's, delta ID's, and length changes plus nominal fuel dimensions. Knowing the measurement error associated with the input data, the uncertainty of a calculated delta volume was computed. This indicated how large a volume change must be to be considered "real".

Analyses are being made of "as received" and "vector 1" warp data for NPR fuels in order to determine if samples of as received warp can be used as a process control tool indicative of when the process should be stopped for corrective action.

Irradiation Processing Department

Considerable editing of source data was required to correct the input for issuance of accurate exemplary outage reports for IPD reactor control personnel. A small error still exists in the source data file due primarily to erroneous time recordings. The exemplary reports requested by various personnel during meetings held in May have been issued to appropriate personnel. A redefinition of reactor outage cause classification and the corresponding total outage period data for each reactor beginning January, 1962, has been completed by the Production Operation. The collection of outage data for preceding periods will continue to the point where each of the reactor systems was modified from a preceding operational standard. It will then be possible to delineate the outage causes in a more meaningful manner and determine a realistic pattern of outage occurrence identified by the proper causes. As an additional activity in the IPD Operations Simulation, the initial programming required to determine craft-set requirements for outage recovery is now in process.

As requested, an analysis was made to determine the feasibility of measuring partial charges of Quality Certification fuel elements. The analysis developed a formula showing the fraction of a charge to be measured which results in minimum variance for the average result of a group of material, based on a knowledge of within and between tube variation. It was assumed that there is no lack of material to be measured, but that time restrictions do not permit measurement of all pieces.

Quality Certification data are being processed by reactor to investigate more fully the warp and hot spot relationship. Of special interest is a comparison of reactors. This has potential significance from both a fuel performance and tube corrosion standpoint.

Data were analyzed from an experiment designed to separate process error from measurement error in determining reactor orifice constants. This experiment was motivated by an analysis performed in early 1961 in which it was suspected that the measurement error was understated due to the method by which it was estimated. Although these experimental data confirmed this, there was no change in the estimate of the variation between orifices, as had been anticipated. The data are used in establishing safety factor requirements for front header pressures to prevent cavitation.

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

A document entitled "Studies in Reliability I: The Algebra of Four-State Safety Devices" has been submitted for rough draft typing. It is planned to issue this as a research and development report.

Chemical Processing Department

A review was made of data giving the Pu-240 content of fabricated parts as measured by neutron count on a 100 percent basis, and by mass spectograph on a sample basis. Estimates were made of precisions and relative biases of the two methods, and recommendations were given for routinely determining true content of fabricated parts.

Data have been received from the Interstate Commerce Commission giving information on individual train accidents for 1959-61. A study is being made examining the relationships between certain accident variables. The results will be used in helping to establish risks associated with transporting radioactive materials by rail.

An EDPM program has been written and placed in service which converts the design gauge data of manufactured parts from the standard coordinate system to one which is more time saving and convenient for local gauging.

A letter has been sent to interested personnel containing an analysis of medical treatment injury rates.

Relations Operation

Work is proceeding in connection with the forthcoming attitude survey. Methods of analysis were developed and submitted for programming after being approved by other members of the survey task force. The methods are similar in concept, but differ in detail, from those used in the 1957 survey.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HLO

2000 Program

Pulse Column Facility

Statistical analysis was continued of several time series of mid-column aqueous uranium concentration data to estimate the random variability of uranium concentration at a fixed point in the column during equilibrium operation. The variate difference method has been used successfully to remove the time effect from the series. The next step in the analysis is the estimation of the spectral density function from the trend corrected differences. Several standard techniques are currently under consideration for this spectral estimation. It is hoped that the stability of the column can be defined in terms of this spectral density. Of particular interest is whether the loss of stability prior to flooding can be seen quantitatively as a shift of the spectral density.

Fuel Element Swelling Model

The empirical model formulated by Fuels Development Operation was fitted to fuel element swelling data using the NELLY least squares program. Evaluation of the fit showed that the model neglects a burnup-temperature interaction which restricts the effect of burnup on swelling for temperatures below a critical point. Appropriate changes in the mathematical form of the model were made, and the data are currently being reanalyzed with the Mark II version of the model.

General

Work continues to modify and compliment the EDPM program which calculates the longevity and sensitivity of a proposed neutron flux monitor as a function of its isotopic constituents.

Analysis and programming are continuing on a problem of determining the steady-state nonviscous flow pattern of a fluid in a cylindrical tank which has been equipped with an axially-located circulating device. Preliminary computer runs have indicated remarkable agreement between theoretical predictions and actual measurements taken from a small model.

3000 Program

The prototype $\delta - \omega$ lathe control components developed by the Manufacturing Services Laboratories have arrived on plant and are being assembled on the experimental Gorton lathe. A special magnetic tape has been prepared to test these controls for fidelity, accuracy, acceleration and deceleration characteristics, and response to auxilliary commands. Programming is well under way to generate a magnetic tape suitable for milling an actual specimen of a desired manufactured part.

A series of meetings were held to discuss the feasibility of designing metal blanks which can be shear-spun on a Floturn machine into certain pre-selected shapes. Several designs have been submitted which in theory should produce desirable metal characteristics in the final product.

4000 Program

Plutonium Fuels Research

The experiment to measure the within-day and between-day experimental errors in the Azure C method was completed, the data analyzed and the results reported to the customer. As a test to determine the precision with which trace boron content in a nest of four fuel samples can be estimated using the Azure C method, an experiment was designed using sections of the extruded rod immediately adjacent to the four fuel samples. The analysis of the experimental data indicated a time dissolution effect and gave some suggestion of a heterogeneous boron content in the direction of extrusion. An incomplete block design is being used in another experiment to average out the dissolution effect and provide a better estimate of the rod boron content heterogeneity. Also, in this experiment, the slurry and butt scrap material following the rod fabrication will also be analyzed to obtain a material balance check.

PRTR Process Tubes

Several discussions were held concerning the interpretation of PRTR process tube scratch data.

5000 Program

Actinide Element Research

The final version of a program to index cubic crystals was completely debugged, turned over to the customer, and is functioning routinely. A report on its theory and application is being prepared.

A FORTRAN program was written to extrapolate to 90 percent the value of the lattice constant of a cubic crystal. A report is in preparation describing the application. Work continues on the problem of indexing hexagonal crystals. A feasibility study is under way to determine whether or not an attempt should be made to index orthorhombic crystals.

Computation and Statistical Analysis

Work continued on the activation analysis blank study. Further correlations were run to establish the limits and interdependencies among separate blank constituents. A rule of thumb is sought for the routine interpretation of blank data to determine whether or not the activation analysis process is in control. Several programs were written to provide special reports for the existing Mark I IRA file. A discrepancy in the mathematical tables previously calculated in connection with the report, "Fixed Time Estimation of Counting Rates with Background Corrections", was corrected and several debug tables calculated.

Debugging of the GEM FORTRAN language program for the quantitative resolution of a time dependent gamma pulse height spectrum was completed. Two additional input routines and associated output routines for GEM were written. These routines are more general and more flexible than the previous ones used with the SPEC program. A test case of duplicate mixtures of known composition prepared by Chemical Research and counted on a gamma energy analyzer were run using GEM. The results of the estimation were in agreement with the known composition.

Two programs, the active file updating program and the punch tape edit program, of the IRA Mark II data file system are now about 90 percent coded. Routine meetings are being held with EDPO personnel for further definition of the IRA Mark II file as needed. The definition of type and form of data to be stored in the file will be completed in July.

6000 Program

Biology

A program to estimate parameters for a multicompartment model is nearing completion. It will be used first for a Sr-90 retention study on fish.

Work was begun on a curve fitting problem involving liver damage in sheep.

A rebuttal to referee criticism on a paper dealing with Cs-137 content in milk was written and returned with the paper for resubmission for publication.

Further analysis was done on data concerning Ca and Sr-90 in pigs.

Environmental Studies and Evaluation

Statistical analysis of data was initiated to investigate the relationship of certain factors to the uptake of P-32 in Whitefish. The factors under consideration are temperature of the river water, concentration of P-32 in the river water, and the rate of flow of the river water.

Other

Instrumentation

Several discussions were held concerning the modification of the FORTRAN language GRA program to include the listing of 95 percent confidence intervals on the isotope disintegration rate estimates. It was ascertained that a minor revision in the storing of background information and the addition of minor calculation loops will accomplish this revision.

Personnel Monitoring

The present program for neutron measurement using NTA film was reviewed. An experiment was designed to determine whether or not a linear relationship exists between the number of neutron tracks counted and the dose level to which the film is exposed. The experiment was also designed to investigate the effect of different observers and the effect of shielding on the number of neutron tracks counted.

A statistical analysis on data from a study being conducted to evaluate the present pencil program was performed. The analysis determined the precision with which a pencil can be read at different dose levels.

The high sensitivity of detection of Pu in urine samples has led to considering the use of the procedure in which pooled samples are used, with individual samples taken only in the event of a positive result. Data are presently being examined to determine the frequency of positive results. These will be used in determining optimum group sizes using existing results.

General

An analysis was made comparing UO_3 analyses for Hanford and Paducah. The data consisted of mass spectrometer measurements performed on the same samples.

As requested, a critique was made of an expository article dealing with the estimation of measurement variances between laboratories, operators, days, and runs. The analysis was questioned primarily on the ground that "days" was treated as crossed, rather than nested. Other criticisms were made.

Carl A. Bennett

Manager
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CA Bennett:dgl

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REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Code Development

1. Reactor Dimensionality

The two-group diffusion code for linkage with MELEAGER has been debugged and tested with several test problems. The linkage with MELEAGER is being debugged. The code will allow analysis of variously loaded zones in reactors with MELEAGER on an interim basis until more rigorous codes are completed.

The standard three-region test problem supplied with the code F-3 and tested in the HFN code was the most rigorous test completed to date. The calculated reactivity and power were within 0.01 percent, flux 0.05 percent, and spectral index, r, 0.03 percent of values computed by the other codes. Two-region cases with various numbers of mesh points in slab and cylindrical geometry agreed with each other to 0.001 percent.

2. Minimizer

A large number of cases with slightly differing initial enrichments were computed in order to test the MINIMIZER code. It was found that the curve of fuel cost as a function of initial enrichment (all other parameters held constant) was somewhat jagged (.01 to .001 mills/kwhe in height). The accounting formula in QUICK specifies in accordance with AEC regulation that the fuel rental charges are paid each six months throughout the fuel's in-reactor lifetime. Consequently, there will be a discontinuity in the economic interest portion of the total fuel cost when the in-reactor time is an exact multiple of six months. An in-reactor time that is not a multiple of six months will allow the final rental payment to be delayed until the end of the period and, as the QUICK code considers that this money is invested at an interest rate greater than the AEC use charge rate, the operation receives a credit during the balance of the period. The credit will be the greatest when the in-reactor time is just slightly greater than a multiple of six months and will diminish to a minimum when the shutdown time coincides with a payment time.

3. Incremental Costs for Handling Plutonium

The FEFJ code has been applied to compute the additional costs of zirconium jacketing plutonium enriched UO_2 fuel over that of comparable uranium enriched fuel in production sized fuel plants described by H. Hanthorn. Analysis of the factors constituting jacketing costs shows that most of the cost is for zirconium and nonplutonium containing steps, and the incremental charges for plutonium enriched fuel are calculated to be less than \$5 per pound of uranium contained (low exposure plutonium case). Continuing the study further, to simulate high exposure plutonium, the costs of the steps involving plutonium (labor, equipment, rejects, and associated rework) have been multiplied by factors of 2, 4, and 8 over the original design figures. The incremental charges for the factor of 8 are less than \$20 per pound uranium contained. A \$10 per pound uranium increment reduces plutonium values about \$1 to \$2 per gram of fissile plutonium for plutonium enrichment levels in U-238 of 2 to 1 percent.

4. Plutonium and U-233 Values Computed with MELEAGER CHAIN

No more adjustments are anticipated to the MELEAGER CHAIN code, and runs have been started to complete the plutonium and U-233 values analysis with this code using the post-July 1, 1962, uranium price schedule. A reasonably complete study is anticipated involving the reactor concepts simulated in HW-72217, "Fuel Cycle Analysis for Successive Plutonium Recycle." Plutonium recycle in U-238 and U-233 recycle in thorium will be examined for batch and graded fuel cycles. By appropriate replicate experiment design, the effect of variations in interest rates, incremental fuel processing costs, and uranium price schedules will be studied. In addition, plutonium recycle limited to natural uranium feed will be examined. The computations are about 20 percent complete at this time.

5. Correction to Programming - Monthly Report, May 1962, HW-73905 F

Due to a typographical error on page F-3 under the caption Computed Spectral Index, Modified MELEAGER, fifth column of figures, the amount "0.103" should be changed to "1.03."

The related figures in last month's report were part of a comparison of MELEAGER against a special version of the General Atomics SPECTRUM V code. By adjusting the plutonium isotopic chains self-shielding factors in MELEAGER, it was shown that the MELEAGER code could be made to "track" the more rigorous spectrum code so as reasonably to represent plutonium burn-out in reactor circumstances far beyond those possible by use of the classical flux and cross section model originally formulated by Westcott and used in MELEAGER. The self-shielding factors used in MELEAGER have been adjusted so the plutonium isotopes have pessimistic characteristics relative to corresponding values from SPECTRUM V which represents only a homogeneous reactor system and furthermore which accounts only for plutonium resonances up to 4.2 electron volts. (This includes the major resonances in the plutonium chain.) The current MELEAGER code is severely pessimistic with respect to plutonium isotopes in heterogeneous reactors for spectral indexes greater than 0.5 and is quite pessimistic from 0.3 to 0.5. Most of the plutonium recycle fueling systems involving uniform enrichment of U-238 being studied have spectral indexes less than 0.3. Other codes are being applied for circumstances beyond this. The speed with which MELEAGER computations can be made with the modified Westcott system has allowed rapid survey of the many possible fueling schemes of interest. As the more productive systems are identified they can be appropriately analyzed with more rigorous models.

Investigation of Key Factors in Plutonium Value Calculations

This work is an extension of the Error Analysis portion of HW-72217 in which σ and ν of plutonium were varied in order to evaluate their impact on the calculated plutonium value. The results published in HW-72217 suggested to some readers that plutonium value could be correlated with the hybrid parameter $\sigma\,(\eta - 1)$ for plutonium. The present study was initiated to test this contention by varying both σ and ν simultaneously through a wide range of values.

Preliminary results show that the ratio of the calculated value to the standard value plotted as a function of the ratio

$$\frac{\sigma\,(\eta - 1)}{\sigma_o(\eta_o - 1)}$$

(where the subscript zero indicates the standard case) cannot be represented by a single curve. Calculations have been made on three reactor types and similar results have been obtained for each.

Uranium Price Schedule Calculations

Values of the optimum tails composition in the diffusion cascade were calculated as a function of the ratio of feed cost to separative duty cost at the request of the AEC. These data were obtained by programming an iterative solution of the equation:

$$C_f/C_s = V(X_f)-V(X_o) - (X_f-X_o)\,V'(X_o)$$

where:

C_f = cost of feed material, \$/kg uranium

C_s = cost of separative duty, \$/kg uranium

X_f = feed composition, wt fraction

X_o = optimum tails composition, wt fraction.

V(X) represents the "value" function derived by Cohen* while V'(X) is its derivative. Values of the optimum tails composition were obtained for C_f/C_s values from 0 to 5.0 in 0.1 increments considering natural uranium as the feed material.

A similar calculation was made to obtain the optimum tails composition in the AEC price schedule effective July 1, 1962. The parameters in this schedule are C_f = \$23.50 and C_s = \$30 which give an optimum tails composition of 0.002531 wt fraction.

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* Cohen, K., The Theory of Isotope Separation, NNES-III-18, McGraw-Hill, New York, 1951.

Phoenix Fuels

1. Plutonium Isotopic Content

Plutonium fuels of widely varying composition have been studied under varied conditions of geometry and spectrum for conditions yielding Phoenix action. By drawing on previous results, it is possible to eliminate as contenders reactors having well-moderated spectrums for Phoenix fuel burners if the plutonium compositions contain less than 40 percent Pu-240. Few reactors produce plutonium having Pu-240 in greater percentage than this. Consequently, the computer was allowed to adjust the moderating ratio in order to produce initially a medium-hard or a hard spectrum. (Medium-hard is here defined to be $r = 0.3$ and hard is defined to be $r = 0.5$, where r is the epithermal index ratio of Westcott notation.) Cases of low self-shielding (achieved by increasing the fuel surface-to-volume ratio which permits more relative neutron absorption in Pu-240) have been included.

It appears that Phoenix action can be achieved with many plutonium fuel compositions if the fuel is comprised of greater than 20 percent Pu-240. With increased Pu-240 concentration, the surface-to-volume ratio of the fuel element can be smaller. With low Pu-240 concentrations, the surface-to-volume ratio of the fuel element should be large enough that self-shielding of the Pu-240 resonance is decreased. (There are basic limitations of how effective increasing the surface-to-volume ratio can be.)

If the ratio of Pu-241/Pu-240 is too large, there is insufficient fertility for the system and Phoenix action will not result. Although Pu-241 has desirable nuclear characteristics, it cannot be substituted for an equal amount of Pu-239 in a plutonium fuel if Phoenix action with a nearly constant reactivity is desired. The higher eta of Pu-241 tends to raise the reactivity, while the production of some Pu-240 from Pu-239 appears to reduce reactivity appropriately as well as to supply Pu-241 which maintains the fissile inventory. Plutonium fuels with high (Pu-241/Pu-239 + Pu-240) ratios may require added fertility and thereby benefit from the use of small amounts of U-238 or thorium.

2. Four-Integer Identification System

A four-integer identification system has been adopted for identifying the individual cases of the Phoenix study for Phoenix experiments. The first two digits denote the composition as defined in Table I. The first ten compositions shown in Table I were selected so as to indicate the role of Pu-240 which was held constant at three levels. The other compositions are representative of prior studies as well as of plutonium from certain specific reactors.

TABLE I

PLUTONIUM ISOTOPE COMPOSITIONS FOR PHOENIX FUEL STUDIES

Case No. First and Second Identification Digits	Atom Percent Pu-239	Pu-240	Pu-241	Pu-242
01	100	0	0	0
02	90	10	0	0
03	80	10	10	0
04	70	10	20	0
05	70	30	0	0
06	50	30	20	0
07	40	30	30	0
08	50	50	0	0
09	40	50	10	0
10	30	50	20	0
11	70	5	20	5.0
12	40	30	20	10.0
13	30	30	20	20.0
14	58	25	10	6.0
15	37	45	15	3.0
16	52	28	14	6.0
17	62	20	15	3.0
18	85	15	0.0	0.0
19	45	40	10	5.0

The third and fourth digits of the identification system denote the r and SCA as shown in Table II. SCA is the term in MELEAGER that reflects the surface-to-volume ratio. The larger SCA, the less the resonances are shielded. Increasing SCA increases the effective Pu-240 cross section which is usually of benefit to Phoenix fuels.

TABLE II

SCA AND INITIAL r VALUES FOR PHOENIX CASES

Case No. Third and Fourth Identification Digits	r	SCA
30	0.3	0.5
31	0.3	1.0
32	0.3	2.0
35	0.3	5.0
50	0.5	0.5
51	0.5	1.0
52	0.5	2.0
55	0.5	5.0

Some caution must be used in interpreting the cases where SCA is large. In some of the cases studied SCA was numerically greater than SDPV. These cases probably do not have physical meaning since values of SCA are likely restricted by the equation of the following type:

$$SCA = \Sigma_s + \frac{\left(\frac{S_{eff}}{4_V}\right)\left(\frac{SDPV}{\xi}\right)}{\frac{S_{eff}}{4_V} + \frac{SDPV}{\xi}}$$

Σ_s = Macroscopic thermal neutron scattering cross section of the fuel.

S_{eff} = Effective fuel surface.

V = Fuel volume.

ξ = Logarithmic energy decrement.

SDPV = Slowing down power per unit volume.

So far, fuel compositions where Phoenix action occurred with SCA $\simeq$ 5.0, which may not be real, also show Phoenix action with SCA's of about 2.0, which can be physically achieved. In most cases Phoenix fuel action occurred best with SCA values greater than one (large surface-to-volume ratios).

3. Burn-Up Achieved in a Representative Phoenix Fuel

In System 0951, the Pu-241 reached a concentration of twice the initial value about midway through burn-up. The final Pu-241 concentration was about equal to the initial concentration. Ninety-six percent of the Pu-239 and 75 percent of the Pu-240 were destroyed.

$$K_{initial} = 1.10$$

$$K_{max} = 1.16$$

$$K_{final} = 1.045$$

Of an original one gram of fuel (fissile + fertile) approximately 0.75 gram was destroyed. The exposure was 581,444 MWD/T or 0.641 MWD/cc.

4. Specific Power of Phoenix Fuels

After examining the results of operating at several specific powers, it appears that studying two specific power levels, 110 and 1100 watts/cc of fuel, will embrace the extremes. Approximately 60 percent more fuel gave Phoenix action at the higher specific power due to minimization of the effect of Am-241 from the decay of Pu-241 (13-year half life).

5. Economics of Phoenix Fuels

Fuel cost for several Phoenix fuel systems have been determined with the QUICK economics code. The physics of these systems were calculated prior to introduction of the four digit identification system and before standardization of specific power. The fuel cycle cost is principally a function of the interest rate, FEFJ cost, separations cost, plutonium price, and thermal/electrical conversion efficiency. Representative fuel costs for selected combinations of these parameters are reported in Table III. In all cases, the economic interest is 12.5 percent and the plutonium price is taken as $11.64 per gram (fissile). The AEC use charge on fuel was varied as indicated. The thermal-to-electrical conversion efficiency is assumed to be 33 percent in all instances. For comparison purposes an FEFJ of $2.44/cc would correspond to $115/lb for UO_2 at density of 9.25 g/cc; $0.61/cc would correspond to about $29/lb. (The corresponding physics parameters are given in Table IV.)

TABLE III

FUEL COSTS FOR PHOENIX FUELS

Table IV Case No.	Specific Power* MW/T	Specific Power* W/cc	AEC Use Charge %	FEFJ $/cc	Separations Cost, $/cc	Fuel Cost $Mills/kwh_e$
23	1500	3300	4.75	0.61	0.20	0.963
23	1500	3300	12.5	1.22	0.20	1.140
23	1500	3300	12.5	2.44	0.20	1.262
22	450	1000	4.75	0.61	0.20	1.135
22	450	1000	12.5	1.22	0.20	1.492
22	450	1000	12.5	2.44	0.20	1.651

* Note that the impact of specific power on fuel costs is large.

6. Tailored Phoenix Fuels

Preliminary computations indicate that Phoenix fuel action need not be limited to Pu-240 as the primary fertile source. The over-all performance of a Phoenix fuel has been essentially duplicated starting with equal amounts of U-235 and U-238 physically arranged so that the U-238 absorption resonances are unshielded while the U-235 absorptions are heavily shielded. This is achieved by placing the U-238 in rings (0.002-inch thick) interspersed with moderator which surrounds the U-235 which may be concentrated in a thin rod. The quantity of moderator is so adjusted that a classical U-238 resonance escape probability of approximately 0.5 is achieved. This provides nearly as many neutron absorptions in U-238 as in U-235; i.e., a conversion ratio of approximately 0.8. Thus, the ratio of fissile-to-fertile fuel is sufficiently constant with exposure to provide a uniform reactivity for an extended period as in "Phoenix" fuels.

In the same vein, it is currently believed that the use of plutonium that contains some Pu-240 -- but an insufficient amount for Phoenix fuel action in its own right (for example, weapons grade plutonium) -- can use only enough U-238 to hold the initial reactivity down until enough Pu-240 is formed to supply the classical "Phoenix" fuel. For this application, the U-238 and moderator geometry will be less critical than it would be with U-235. These calculations are based upon experimentally determined resonance absorption self-shielding data, published by Hellstrand of Sweden.

TABLE IV

PHYSICS PARAMETERS FOR PHOENIX FUELS

Case	Fuel Density (g/cc)	SCA	SDPV	Specific Power (Thermal) MWD/T	Specific Power (Thermal) w/cc	Exposure MWD/T	Plutonium Isotopic Composition (in atom percent) 239	240	241	242	K_{max}	Exposure at K_{max} MWD/T	SCA and SDPV (normalized to Fuel Density of 1 gram/cc)	
1	0.3	1.0	1.5	1,000	330	661,278	38	44	15	3	1.19	334,204	3.3	5.0
2	0.3	1.0	1.5	3,000	1,000	712,478	38	44	15	3	1.19	547,500	3.3	5.0
3	0.3	1.0	1.5	10,000	3,300	699,248	38	44	15	3	1.18	568,877	3.3	5.0
11	1.0	1.5	3.7	300	330	568,750	33	49	15	3	1.12	383,250	1.5	3.7
12	1.0	1.5	3.7	1,000	1,000	649,763	33	49	15	3	1.15	300,000	1.5	3.7
13	1.0	1.5	3.7	3,000	3,300	661,278	33	49	15	3	1.12	547,500	1.5	3.7
21	2.0	1.6	5.7	150	330	462,945*	32	50	15	3	1.10	0.0*	0.8	2.85
22	2.0	1.6	5.7	450	1,000	561,017	32	50	15	3	1.13	164,250	0.8	2.85
23	2.0	1.6	5.7	1,500	3,300	649,900	32	50	15	3	1.13	387,528	0.8	2.85
31	3.0	1.7	7.3	100	330	310,250*	35	47	15	3	1.10	0.0*	0.567	2.43
32	3.0	1.7	7.3	300	1,000	563,292	35	47	15	3	1.12	517,781	0.567	2.43
33	3.0	1.7	7.3	1,000	3,300	649,900	35	47	15	3	1.17	585,000	0.567	2.43
31b**	3.0	1.7	7.3	100	330	255,000*	35	47	15	3	1.10	0.0*	0.567	2.43
32b**	3.0	1.7	7.3	300	1,000	540,000	35	47	15	3	1.12	109,000 to 380,000	0.567	2.43

* No Phoenix action.

** Constant r (r = 0.5).

Note: At the far right SCA and SDPV have been normalized to the values they would have in order to achieve the same performance if the plutonium density of the fuel elements were 1 gram/cc. In most cases, this brings the values of SCA and SDPV into practical ranges. The compositions of the Phoenix fuels in the above table are so high in Pu-240 as to be scarcely achievable in many power reactor spectrums. The data for lower Pu-240 content plutonium batches were not available for this report.

Computations with more elaborate models are now being made under the direction of the HLO Theoretical Physics group to firm up the application of U-238 as described, and to investigate the possibility of using thorium in the same fashion. By inspection of the cross section data, it appears that U-233 has substantially more resonance absorption than U-235, which may complicate its application with U-238 or thorium.

The application of fuels of the foregoing types is broad since their physical make-up can be matrixes of a durable parent metal with uranium and thorium as relatively fine particles and the fuel geometry can be a variety of shapes. These characteristics should be especially attractive for portable power systems. If completely successful, the fuel costs with these fuels may be low enough for central station power plants because essentially 50 percent of the heat can be generated by fissioning tails uranium.

By-Product Isotopes

In view of the announced prices for the Am-241 and Np-237 isotopes ($1500 and $500 per gram, respectively) preliminary consideration was given to the values of these and the related isotope in spent power reactor fuels, specifically U-236. The anticipated concentrations of U-236 in such fuels encourage consideration for their use as enriching agents in production reactor fuels from which the valuable Np-237 may be eventually recovered. Furthermore, it appears that a significant value or credit may be assumed for U-236 utilized in this way. Although the concentration of Am-241 is extremely low, its high apparent value also encourages consideration of its recovery for credit.

Salt Cycle Economics

Debugging of the Salt Cycle Economics code is nearly completed. A number of runs have been made and the code appears to be working satisfactorily in all major aspects. A small number of hand calculations were continued; further checking of this nature remains to be done.

When this work is completed, it will be necessary to revise the previously prepared conventional Reprocessing code for compatibility with the Salt Cycle Economics logic as it finally evolved. It is expected that this will be completed in the next reporting period.

W. K. Woods

Manager,
Programming

WK Woods:jm

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF JUNE 1962

A. ORGANIZATION AND PERSONNEL

Transfers within the Section during the month included Robert H. Wilson transferring from External Dosimetry to Environmental Studies and Evaluation, and William V. Baumgartner transferring from Radiological Development and Calibrations to External Dosimetry. Temporary summer employees joining the External Dosimetry Operation included Edna D. Britch, Duane R. Naught, Peter O. Anderson, and George M. Stephens. William C. Wann, Jr., joined the Internal Dosimetry Operation. John A. Gile transferred from the Irradiation Processing Department to the Radiation Monitoring Operation. Mary Jo McLean was reactivated into the Composite Dose Studies and Records Operation. Patricia C. Freed and Amelia N. Holland resigned from the Company. James L. Beecroft and Harold E. Ransom, both teachers at Columbia Basin College, joined the Radiation Protection Operation for the summer.

B. ACTIVITIES

Occupational Exposure Experience

On June 3 a pipefitter in the Chemical Processing Department received a plutonium contaminated puncture wound to his left index finger at the 234-5 Building. The injury occurred in a wet chemistry hood at the head end of the purification process line and was probably caused by a piece of corroded wire that was recovered from the floor of the hood. An initial excision was performed by an industrial physician at the 200 West First Aid Station. Subsequent examination with the wound counter showed that about 0.9 μc of Pu was removed with the excised tissue, but that about 0.1 μc remained at the wound site. An additional tissue excision, which required subsequent minor skin grafting, reduced the contamination at the site of the injury to about .006 μc. Rapid analysis of bioassay samples indicated substantial absorption of the plutonium from the injury site. Consequently, it was decided by the industrial physicians to administer DTPA. Analysis of bioassay samples for the four-week period following the injury indicated that the employee excreted about .01 μc Pu. It is not yet possible to provide a reliable estimate of the magnitude of the plutonium deposition because of the effect of DTPA treatment on plutonium excretion rates.

Nine plutonium contamination incidents involving potential inhalation of plutonium for 13 CPD employees at the 234-5 Building occurred during the month. Five plutonium contamination incidents involving nine HLO employees at the 231-Z Building were also reported. All except two of these incidents resulted from ruptured hood gloves or the failure of plastic bags containing contaminated material. Special bioassay sampling was initiated for all employees involved.

Three new cases of plutonium deposition were confirmed by bioassay analyses during June. The total number of plutonium deposition cases that have occurred at Hanford is 291, of which 209 are currently employed.

A maintenance employee in IPD received a whole body dose of 1.9 r while attempting to remove a steel shielding piece from a thermocouple stringer channel at the 105-KW reactor. Movement of the shielding piece apparently exposed a small length of a thermocouple wire. The proximity of the employee's right hand to the irradiated wire resulted in an estimated integrated dose of 7 rads to the hand. Small areas of the fingers may have received a dose of about 45 rads. The employee's accumulated whole body dose for the calendar year including the dose received in this incident is 2.4 r.

In similar work at a different time, a maintenance supervisor in IPD received a gamma dose of 1.2 r while directing preparations for the removal of thermocouple stringers at the 105-KE reactor. During the course of positioning a steel can over the stringer channel opening the radiation dose rate increased from 0.2 r/hour to 40 r/hour. The dose rate at the position of the maintenance supervisor was about 3 r/hour. Analysis of the dosimetry involved indicated that the maximum dose to the body including the gonads was between 1.2 r and 1.9 r. The supervisor's accumulated whole body dose for the calendar year including the exposure received in this incident is estimated to be 3.8 r.

Regular processing and evaluation of the film dosimeter worn by a CPD process supervisor at the Uranium Reduction Plant indicated an apparent beta dose of 4 rads. The film exposure was considered unusual because of the absence of any associated gamma dose. Detailed investigation of the employee's work activities failed to provide any explanation for the apparent exposure, but it was learned that he frequently stored his film dosimeter with his radium-dial wrist watch in a drawer at his home. Experimentally, it was determined that the film exposure was duplicated when the watch dial was placed adjacent to the open window of the film badge dosimeter for about 16 hours. Results of the film study and investigation of the work activities of the supervisor were documented to show that the dose to the film dosimeter was not received occupationally.

The first shipment of the new Hanford beta-gamma film badge dosimeters was received from the vendor early in June. As of the end of the month about 11,000 of the 30,000 dosimeters ordered had been received.

Exceptionally high dose rates were encountered during the handling of radioactive waste from the 327 Building. Dry waste was transported for disposal at the Wye burial ground with a dose rate of 60 mr/hour to the truck driver. On another occasion Gatlin Gun disposal of small pieces of irradiated fuel elements involved momentary dose rates to personnel up to 50 r/hour.

Thorium ingot surface removal operations and other processing at the 306 Building were performed repetitively without detectable surface or airborne contamination. The co-extrusion operation involved slight contamination to the bench of 1000 d/m and to tools of 500 d/m.

Tritium air concentrations at the Plutonium Recycle Test Reactor up to 6.4×10^{-4} μc T/cc resulted from a heavy water spill in the lower access space. The maximum bioassay result was 47 μc T/l. In other activities, a section of an irradiated process tube was inadvertently discharged into the storage basin from a shipping cask. A momentary exposure of one employee at a distance of six feet resulted in a whole body dose of about 20 mr. The process tube showed a dose rate of about 1000 r/hour at 2 inches. During the inspection of 85 process tubes radiation beams up to 50 rads/hour were encountered. Personnel dose rates were limited to 0.5 r/hour during process tube replacement.

Environmental Experience

Fresh fallout (estimated by the I^{133}/I^{131} ratio to be five days old) was detected in the Pacific Northwest in the middle of June. This fallout was evidenced in samples of produce which were found to have fission product concentrations up to 20 times that noted for early June. The I^{131} content of beef thyroids collected in late June were also some 20 times higher. Local milk samples were up to 10 times higher in I^{131} content.

A larger than usual quantity of fission products was released to the Columbia River from a fuel element failure at 100-D. Several special river water samples were taken. Results are incomplete and complicated by the presence of fallout, but the preliminary analyses indicated nothing extraordinary.

A total of 102 fish were taken from Columbia River sampling locations at Priest Rapids, Hanford, Richland, Burbank and McNary Dam. One hundred six tissue samples from these fish were submitted to the laboratory for radiochemical analysis.

The following produce samples were collected: 60 samples of 29 varieties of vegetables and fruits; 4 pounds of Willapa Bay oysters; 40 sets of beef thyroids; 20 samples of pasture grass; 109 gallons of milk; and one chicken. Daily milk samples from the Benton City area are scheduled for July to better follow the recent increase in fallout.

Six aerial monitoring flights were made as part of the background radiation study.

Studies and Improvements

The new personnel dosimeter processing machine was placed into service on June 28. This machine unloads, loads, dates and identifies the film used in the Hanford film badge dosimeter.

Several quality control tests were instituted for use in the pocket ionization chamber program. These tests are designed to identify and exclude the use of dosimeters that have poor insulating characteristics and bad electrical contacts. Periodic checks were adopted to assure that only acceptable dosimeters are in routine service.

The ad hoc dosimetry committee that is preparing the final dosimetry report of the Recuplex incident prepared a second draft of the document.

The 300 Area emergency plan was revised and issued for comment. Detailed evacuation plans in the event of a nuclear excursion in the 300 Area are complicated by the proximity of buildings to each other. Consistent practices regarding evacuation routes, location of area staging sites and personnel accounting are being developed in cooperation with representatives of FPD.

An audit of radiation generating machines at Hanford was completed. Numerous locations were found where the associated administrative controls were not posted at the equipment location. Notification of the custodians of the machines resulted in improvement of controls for these sources of radiation.

The radiological status of the Riverland radioactive waste disposal area was reviewed. The total quantity originally present has been estimated at 250 μc of reasonably short lived fission products and neutron activation products. The residual activity is so small that radiological controls are no longer necessary.

Final calibration of the new portable BF_3 neutron monitoring instruments indicates that full scale readings for thermal neutrons will be about 3, 30, and 300 mrems per hour. Fast neutron calibrations indicate full scale readings of 50, 500, and 5000 mrems per hour.

A purchase requisition was issued for the procurement of a plutonium-238 beryllium neutron source with an emission rate of 8×10^7 neutrons per second. This source will contain about 3 grams of Pu^{238} and will have an emission rate of about ten times that of the plutonium-beryllium source. This new source is required to provide neutron calibration of the new higher range BF_3 neutron monitoring instruments.

The experimental plan for use of the Sandia Corporation Godiva reactor for obtaining actual fast neutron burst data for the Hanford criticality dosimeter and the new Hanford personnel film badge dosimeter was prepared. Discussions with Sandia indicate reactor time will be available about the middle of August for the completion of these studies.

Air samples were collected from four hoods in the 308 Building by connecting the sampler to a quick disconnect fire extinguisher outlet. Three of the four hoods had positive plutonium air concentrations of 3.5×10^{-9} μc/cc, 3.4×10^{-9} μc/cc and 6.5×10^{-10} μc/cc. Small discs were cut out of these filters and were sent to Materials Development Operation for electron microscope analyses. The preliminary electron micrographs of two of these filters revealed a mean particle diameter of 0.046 μ with a particle range on one filter of 0.025 to 0.1 μ and on the other filter 0.025 to 0.15 μ. The alpha track labeling of these particles was not performed; thus, the particle cannot be positively identified as plutonium. However, particles this small would have to be a heavy metal to be opaque to the electron beam. The alpha track emulsion work was initiated but no results are available to date.

The calibration standard Victoreen r-meter set number 1512 was returned from the National Bureau of Standards. Two of the r-meter sets were matched to this recently calibrated r-meter by comparing exposures in a half-gram radium radiation field. The standard r-meter will be used only for calibrations of the other r-meter sets within the Calibrations facility.

A total of eleven emergency monitoring kits were serviced during June.

C. VISITORS

Visitors consulting with members of the Radiation Protection staff during the month included:

D. C. Nichols - Savannah River Plant, E.I. duPont de Nemours, Aiken, South Carolina.
G. Hansen)
C. R. Ogden) - - Pollution Control Commission, State of Washington, Yakima, Washington.
R. Chambers - - AEC, Idaho Operations Office, Idaho Falls, Idaho.
W. D. Carlson - Colorado State University, Fort Collins, Colorado.
J. G. Mehl - - International Atomic Energy Agency, Vienna, Austria.

Visitors who toured the Whole Body Counter and the Bioassay Laboratory during the month included:

14 employees of the Richland Benton-Franklin District Health Department
27 eighth-grade students from the Peninsula Pilgrims Mobile Education Unit, Goodman Junion High, Gig Harbor, Washington

Members of the Radiation Protection Operation visiting off-site during the month included:

B. V. Andersen - Health Physics Society meeting in Chicago to present two papers.

P. E. Bramson - Health Physics Society meeting in Chicago to present two papers; Argonne National Laboratory, Argonne, Illinois, to discuss silicon semiconductor neutron spectrometers.

L. A. Carter - - Health Physics Society meeting in Chicago to present a paper and take Certified Health Physicist exam.

R. F. Foster - - JCAE Hearings in Washington, D.C.; Nuclear Congress in New York to present a paper; Brookhaven National Laboratory to discuss environmental monitoring programs; and Health Physics Society meeting in Chicago to present a paper.

A. R. Keene - - Health Physics Society meeting in Chicago to present a paper, attend Board of Directors meeting, and attend AEC Health Physics Fellowship Advisors meeting.

H. A. Meloeny - Instrument Laboratory, Seattle, Washington, to discuss problems relating to manufacture of instruments on order.

E. C. Watson - - Health Physics Society meeting in Chicago to present a paper; Savannah River Plant, E.I. duPont de Nemours, Aiken, South Carolina, to discuss environmental consequences of accidents.

D. RELATIONS

Five AEC Health Physics Fellowship students arrived at Hanford to begin their ten weeks of summer training. The summer course work started on June 18.

A press tour through the Whole Body Counter was held on June 25. As a result of the tour a news article in one of the local papers described the Whole Body Counter and its function.

Three suggestions were submitted by personnel of the Radiation Protection Operation during June. Two suggestions were rejected; none were adopted. Three suggestions are pending evaluation.

Radiation orientations were presented to 45 Laboratories employees. Two 2-hour orientation talks were presented to Biology Research employees. A second talk and demonstration on personnel surveys were given to the Biology Animal Farm personnel. PRTR-assigned personnel completed formal training on the PRP Gas Loop.

Safety meetings were held throughout the Section. The quarterly safety and housekeeping reports reflect continued safety consciousness at all locations.

E. SIGNIFICANT REPORTS

HW-72691-5 - "Summary of Radiological Data for the Month of May 1962" by R. F. Foster.

HW-73366 - - "Evaluation of Radiological Conditions in the Vicinity of Hanford - January-March 1962" by R. F. Foster and staff.

HW-74056 - - "Radiological Considerations Associated with the Fabrication of High Exposure Al-Pu Fuel Elements" by L. D. Williams.

HW-74156 - - "Monthly Report - June 1962, Radiation Monitoring Operation" by A. J. Stevens.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

External Exposure Above Permissible Limits	June	1962 to Date
Whole Body Penetrating	0	3
Whole Body Skin	0	3
Extremity	0	2
Hanford Pocket Dosimeters		
Dosimeters Processed	2,396	19,238
Paired Results - 100-280 mr	28	58
Paired Results - Over 280 mr	5	8
Lost Results	0	0
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	9,166	57,146
Results - 100-300 mrads	318	1,924
Results - 300-500 mrads	20	180
Results - Over 500 mrads	7	75
Lost Results	13	141
Average Dose Per Film Packet - mrad (ow)	19.71	12.84
- mr (s)	28.85	28.07
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	1,710	8,982
Results - 50-100 mrem	4	9
Results - 100-300 mrem	25	29
Results - Over 300 mrem	0	2
Lost Results	8	18
Fast Neutron		
Film Processed	274	2,271
Results - 50-100 mrem	18	323
Results - 100-300 mrem	6	346
Results - Over 300 mrem	0	11
Lost Results	0	0
Hand Checks		
Checks Taken - Alpha	33,770	183,865
- Beta-Gamma	52,913	315,815
Skin Contamination		
Plutonium	32	122
Fission Products	48	266
Uranium	0	11
Tritium	0	0

Whole Body Counter	Male	Female	June	1962 to Date
GE Employees				
Routine	10	0	10	77
Special	10	0	10	136
Terminal	5	0	5	53
Non-Routine	21	3	24	163
Non-Employees	2	0	2	13
Pre-Employment	0	0	0	3
	48	3	51	445

Bioassay	June	1962 to Date
Confirmed Plutonium Deposition Cases	3	8*
Plutonium - Samples Assayed	272	2,528
- Results Above 2.2×10^{-8} μc/Sample	10	108
Fission Product - Samples Assayed	283	3,214
- Results Above 3.1×10^{-5} μc/Sample	0	15
Uranium - Samples Assayed	105	1,070
Biological - Samples Assayed	0	233
Strontium - Samples Assayed	0	299

Uranium Analyses

	Following Exposure Units of 10^{-9} μc U/cc			Following Period of No Exposure Units of 10^{-9} μc U/cc		
Sample Description	Maximum	Average	Number Samples	Maximum	Average	Number Samples
Fuels Preparation	6.9	2.7	20	4.1	1.7	20
Fuels Preparation**	0	0	0	0	0	0
Hanford Laboratories	9.7	4.3	20	12.9	2.7	28
Hanford Laboratories**	0	0	0	0	0	0
Chemical Processing	0	0	0	0	0	0
Chemical Processing**	0	0	0	0	0	0
Special Incidents	0	0	0	0	0	0
Random	2.7	1.4	17	0	0	0

Tritium Samples	Maximum	Count	Total
Urine Samples			
> 5.0 μc/l	46.5	123	
< 1.0 μc/l		28	
Samples Assayed			230
D_2O Samples			
Moderator	743.3 μc/ml	8	
Primary Coolant	250.2 μc/ml	8	
Reflector	692.5 μc/ml	8	
			24
Other Water Samples			
No. 57-B-4	.345 μc/ml		212
			466

* The total number of plutonium deposition cases which have occurred at Hanford is now 291, of which 209 are currently employed.

** Samples taken prior to and after a specific job during work week.

Calibrations	Number of Units Calibrated June	1962 to Date
Portable Instruments		
CP Meter	1029	6,060
Juno	298	1,657
GM	560	3,348
Other	185	1,199
Audits	99	626
	2171	12,890
Personnel Meters		
Badge Film	768	9,412
Pencils	---	12,670
Other	320	2,557
	1088	24,639
Miscellaneous Special Services	447	5,868
Total Number of Calibrations	3706	43,397

AR Keene
Manager
RADIATION PROTECTION

AR Keene:ljw

FINANCE AND ADMINI TRATION

ACCOUNTING

Cost Accounting

The June control budget was adjusted to reflect the following changes in funding:

	Increase (Decrease)
02 Program R & D	
FPD Sponsored Metallurgy	$(25 000)
04 Program R & D	
Plutonium Recycle Program	18 000
Advance Concept Studies	(18 000)
UO_2 Fuels Research	46 000
Irradiation Damage to Reactor Metals	(46 000)

Anticipated FY 1962 budget underruns on Hanford Laboratories' programs sponsored by the Division of Reactor Development were estimated and the results, shown below, transmitted to HOO-AEC in response to their inquiry concerning funds available to alleviate expected budget overruns at another AEC site.

Hanford Laboratories' Program	Estimated Underrun
GCPR-Loop Project	$ 25 000
DR-1 Loop Operation	25 000
Plutonium Recycle Program	20 000
Other Gas Cooled Reactors	30 000
Environmental R & D	10 000
Subtotal	110 000
Capital Equipment	40 000
Total Estimated Underrun	$ 150 000

Contract and Accounting Operation · s advised that Hanford Laboratories' estimates of fund requirements for Attendance at Meetings of Professional and Trade Societies and for participation in Off-Site Courses and Seminars during FY 1963 are $75,000 and $10,700, respectively.

Research and development proposals submitted to HOO-AEC during June were:

1. Plutonium Physical Metallurgy Research - This proposal was revised to a FY 1963 program level acceptable to the Division of Physical Research and which would involve no over-all increase in manpower in the performing component.

2. Chemistry of Molten Salts - A new proposal designed to supplement the present work in molten chloride salts. Sponsorship by the Division of Physical Research has been requested.

Special request activity during the month consisted of a $2,000 increase in authorization by Argonne National Laboratory for additional work on Pyrophoricity Studies.

Effective July 1, 1962 organization codes 7321 - 5 will be converted to foreman codes and this Test Reactor and Auxiliaries component will code, for cost purposes, to subsection code 7320.

Distribution of a new HAPO work order form was made to all Hanford Laboratories' components for use beginning July 2, 1962. The new form was designed to facilitate keypunching and processing by the data processing system.

In progress is a study of the cost of fabricating UO_2-PuO_2 fuel elements for the Plutonium Recycle Test Reactor.

General Accounting

In response to a letter from Counsel to Department Managers and Memo of Counsel No. 10 A, a comprehensive review of the participation of Hanford Laboratories' people on advisory committees and groups is under way. A formal report to the HAPO General Manager will be prepared and submitted by August 1, 1962.

Following is a summary of the status of letters or agreements covering specific actions requiring AEC concurrence:

AT-244	Participation in Wallowa County Educational Day Camp	Not yet approved
AT-246	Special Science Seminars	Approved 5-23-62
AT-247	Participation in Standardizing Activities	Returned unapproved
AT-6	HLO Participation in Operation Small Boy	Consumated under Agreement AT-6

Following is a comparison of travel activity in FY 1962 and FY 1961:

	Total Trips	Total Expenses Charged to Cost
FY 1961	1 404	$367 000
FY 1962	1 335	396 000

During the month of June equipment valued at $788,962 was transferred to plant accounts from Equipment Work in Progress, $13,624 was transferred from Construction Work in Progress, and $896,619 from the AEC covering projects CAH-870 - Facility for Recovery of Radioactive Materials, CAH-842 - Critical Facility and CAH-902 - Uranium Scrap Burning Facility. The EWIP transfer includes $135,932 for the White Bluffs Tube Shop equipment transferred from Kaiser Engineers to the custody of Physics and Instrument Research and Development Operation.

Preparation was completed for the physical inventory of movable catalogued equipment, property of others and research and development equipment in the custody of Reactor and Fuels Research and Development Operation. The inventory count will start July 5, 1962 and extend through October 1962.

Classification activity for the month included the review of 669 purchase requisitions and 677 work orders for capital or expense determination, compounding, work review, and for reimbursability. To reflect the established trend of Hanford Laboratories classification activities, the following statistics are given:

Documents Processed	FY 58	FY 59	FY 60	FY 61	FY 62
Work Orders (average per month)	500	591	717	717	614 -1)
Requisitions (average per month)	332	523	691	687	748
	832	1 114	1 408	1 404	1 362

(1- A decrease in the number of work orders over last fiscal year resulted from Technical Shops method of allocating work to J. A. Jones by use of Work Authorization instead of individual work orders.

Input data sheets generated jointly by Contract and Accounting and Hanford Laboratories in June, to record Hanford Laboratories plant and equipment changes and additions totaled 2,436 (this was 52% of the total for all of HAPO). The large number of input data sheets and amounts transferred from EWIP during June 1962 represents the largest single month's business in the history of the Laboratories.

Hanford Laboratories' material investment at June 1, 1962 totaled $26.7 million, as detailed:

	(In thousands)
SS Material	$25 304
Reactor and Other Special Materials	976
Spare Parts	380
	$26 660

Total HAPO Nuclear Material Consumed in Research the fiscal year ending June 30, 1962 was $2.4 million - $2.0 million by the Laboratories and $ .4 million by FPD. The following is a detail by program for Hanford Laboratories:

	(In millions)
2000 Program	$ .5
3000 Program	.1
4000 Program	1.4
	$ 2.0

Heavy water losses charged to PRTR operating costs for the month of June amounted to $52,080, which was comprised of loss of $51,321 and scrap valued at $759. Scrap material on hand at June 30, totaled 3,169 pounds valued at $50,529. Total FY 1962 heavy water charges to operating cost were $472,243 made up of $444,412 loss and $27,831 scrap material.

One hundred and thirty-three items valued at $40,489 were received at the Laboratory Equipment and Material Pool facility during the month of June. Fifteen items valued at $5,208 were withdrawn by custodians and 21 items valued at $4,288 were transferred in lieu of placement of requisitions. There are currently 737 items valued at $535,807 located in the Pool, of which 124 items valued at $61,546 are currently on temporary loan. During the month 158 items valued at $42,649 were excessed or retired. In connection with the Laboratory Pool, survey of items held for a period of 24 months with no activity, 466 items valued at $106,926 were disposed of this fiscal year.

During FY 1962, 232 items valued at $107,287 were assigned by the Laboratory Equipment Pool to useful purposes within Laboratories, and operating costs for the same period were $11,330. It is reasonable to assume that the equipment was placed in lieu of requisitions, resulting in a net saving to Hanford Laboratories of $95,957.

Operating costs since the Laboratory Pool was activated (December 1959) have totaled $28,851; total building investment including buildings 3718, 3718-A (acquired January 1962) and 3718-B (acquired June 1962), is $103,770. The total expenditure and facilities investment through June 30, 1962, is $132,621, equipment valued at $197,581 has been redirected to useful purposes, and a total net saving of $64,960 has resulted. In addition the Laboratory Pool has provided intangible benefits such as (1) reactor and other special materials technical personnel have been relieved of the responsibility for reactor and other special materials held for future use, (2) improved housekeeping in the Laboratories, (3) the release of valuable laboratory space for other uses, and (4) the existence of a source of immediately available equipment to fill urgent needs.

Materials on hand in the Laboratory Pool at month's end consisted of the following:

Beryllium	1 035 gms.	$ 652	
Gold	2 352 gms.	3 340	
Palladium	2 224 gms.	2 736	
Platinum	8 831 gms.	21 534	
Silver	6 633 gms.	663	
Hafnium	2 939 gms.	1 499	
Zirconium	9 598 lbs.	122 320	-1)
		152 744	
All other material held for the convenience of others		258 175	
Total Material		$410 919	

(1- The net reduction in zirconium inventory of $190,128 is due to inventory withdrawals of 3,647 pounds valued at $128,604, write-down of Zr-3 ingots (5,862 pounds valued at $41,038) and revaluation of Zr-2 ingots to $3.00 per pound resulting in a write-off to cost of $21,258. Approval was obtained from the Commission for the write-off of Zr-3 ingots and devaluation of Zr-2 ingots.

Action as indicated occurred on the following projects during the month:

New Money Authorized HL

CAH-916 Fuels Recycle Pilot Plant $40 000

Physical Completion Notices Issued

CAH-842 Critical Facility
CAH-870 Facilities for Recovery of Radioactive Materials
CAH-924 200-KW Induction Heating System, 306 Building
CGH-937 Safety Improvements 231-Z Building

Construction Completion Cost Closing Statements Issued

CAH-870 Facilities for Recovery of Radioactive Materials
CAH-902 Uranium Scrap Burning Facility
CAH-924 200-KW Induction Heating System, 306 Building
CGH-937 Safety Improvements 231-Z Building

Two Assistance to Hanford authorizations were issued for FY 1963: ATH-HLO-1-63A, Dr. Poritsky et al which is a renewal continuing on a year-to-year basis, and ATH-HLO-2-63A, a new authorization for Electron Microscope Consultation from the Research Laboratory.

Contracts processed during the month included:

SA-217 The Tool Steel Gear and Pinion Company
CA-336 Amos Lane (Supplement)

Revised OPGs issued are listed below:

OPG No.	Title
2.3.1	IPD General Manager Position Guide
2.3.2	CPD General Manager Position Guide
2.3.3	FPD General Manager Position Guide
7.18	Plant Emergency Plans
8.3	Property Management of Special Equipment
8.21	Contaminated Personal Property

Personnel Accounting

Payroll statistics for the month of June are given below:

Number of HLO Employees Changes During Month	Total	Exempt	Nonexempt
Employees on payroll at beginning of month	1 439	655	784
Additions and Transfers In	107	65	42
Removals and Transfers Out	34	21	13
Employees on payroll at end of month	1 512	699	813

Overtime Payments During Month	June	May
Exempt	$ 7 239	$ 3 564
Nonexempt	33 932	27 416
Total	$41 171	$30 980

Gross Payroll Paid During Month		
Exempt	$ 640 067	$ 613 676
Nonexempt	560 622	442 167
Total	$1 200 689	$1 055 843

Participation in Employee Benefit Plans at Month End	June Number	June Percent	May Number	May Percent
Pension	1 323	99.1	1 299	99.3
Insurance Plan - Personal	378		363	
- Dependent	1 093	99.7	1 069	99.7
U. S. Savings Bonds				
Stock Bonus Plan	90	38.6	91	39.7
Savings Plan	74	4.9	73	5.1
Savings and Security Plan	1 112	89.0	1 092	90.2
Good Neighbor Fund	991	65.5	966	67.1

Insurance Claims	June Number	June Amount	May Number	May Amount
Employee Benefits				
Life Insurance	0	$ 0	0	$ 0
Weekly Sickness & Accident	6	501	7	522
Comprehensive Medical	29	2 139	40	3 276
Dependent Benefits				
Comprehensive Medical	81	6 763	92	7 843
Total	116	$9 403	139	$11 641

TECHNICAL ADMINISTRATION

Employee Relations

Twenty-six non-exempt employment requisitions were filled during June and 18 remain to be filled.

Professional Placement

Advanced Degree - Six Ph.D. applicants visited HAPO for employment interviews. Twelve offers were extended; two acceptances and three rejections were received. Current open offers total 14.

BS/MS - Four program offers and 13 direct placement offers were extended. Program results included one acceptance and eight rejections with three open offers remaining. Direct placement offer response this month included five acceptances, 13 rejections and five remaining under consideration.

Technical Graduate Program - One Technical Graduate was placed on permanent assignment, 34 new members were added to the rolls, nine transferred to other Company sites in accordance with the planned program, and one terminated. Current program members total 56.

Technical Information

A classification guide (HW-74048) was issued to cover work being carried out for UCLRL.

Visitors and Visual Displays

Arrangements were made and nineteen Hanford Laboratories displays supplied for use in the AEC-GE Visitors Center which opened in the Richland Community House on June 13.

ECONOMIC EVALUATIONS

Output results of the Fuel Element Fabrication Cost computer code for a U-235 enrichment case were satisfactorily reconciled with the original calculations of H. E. Hanthorn's Fuel Cycle Economics Process Study. The computer programming will permit rapid calculation and analysis of the effect of varying key cost factors.

The cost elements of the manual study for a Pu enrichment case were recast and reconciled for input into the computer program. A similar series of parameter variation and analysis will be made with this case.

PROCEDURES

A letter entitled, "The Paper Burden," was sent to section managers on June 22 in an effort to (a) stimulate compliance with records management regulations, particularly with respect to use of record flow schedules, and (b) promote more effective utilization of file cabinets.

FACILITIES ENGINEERING

Projects

At month's end Facilities Engineering Operation was responsible for 13 active projects having total authorized funds in the amount of $2,704,600. The total estimated cost of these projects is $8,767,000. Expenditures on these projects through May 31, 1962 were $1,308,000.

The following summarizes project activity in June:

Number of authorized projects at month's end ---------------- 13

Number of new projects authorized ---------------------------- 2
CAH-958, Pu Fuels Testing & Evaluation Labs - 308 Bldg.
CAH-962, Low Level Radiochemistry Building

Projects completed -- 0

New projects submitted to the AEC --------------------------- 1
CAH-958, Pu Fuels Testing & Evaluation Labs - 308 Bldg.

New projects awaiting AEC authorization --------------------- 0

Project proposals complete or nearing completion ------------ 4
CAH-977, Facility for Radioactive Particle Inhalation Studies
High Level Pathogen-Free Laboratory
Neutron Calibration Facility - 3745-A Building
Additions to the 222-U Building

The appended pages provide detailed project status information.

Services

Engineering services were provided during June on the following activities with satisfactory progress shown on each:

Electrical design for decontamination facility - 325 Building
Electrical design for Vacion Installation - 3745 Building

Fourteen equipment requisitions totaling $7,000 were issued during the month, bringing the total value of materials and equipment being processed to $400,000.

Plant Engineering effort was expended on:

Pressure Systems: Irradiation Studies Loop (C-1)
Hydraulic Compaction Unit (231-Z)
Tube Burst Facility
328 Building Compressed Gas System
Visual System for Fuel Rupture Autoclave
Third Party inspection was arranged on 28 vessels and items #2 and 5, above.

325 Analytical Laboratory glove box
209-E compressor and motor-control center
325 Ceramic Fuels area airconditioner
325 "B" system duct cleaning
309 M & M Building office addition
328 First floor office modifications
308 Ventilation control review
308 Emergency power tie-in
325 Ceramic fuels lighting
325 Switchgear circuit breaker tests
325-B Switchgear modifications

Maintenance and Operation

Costs for May were $109,834. Costs fiscal year to date, $1,600,399, are 98.1 percent of forecast. Improvement maintenance costs for May were $5,603.

The following tabulation summarizes waste disposal operations:

	May	April
Concrete Barrels	3	21
Loadluggers-Hot Waste	4	4
Crib Waste	240,000 gal.	240,000 gal.

Leaks in the retention waste lines near the basin were a problem during the month. After repairs were made to two joints and new leaks continued to erupt, work was started on recaulking all joints to the first manhole in the system. About three fourths of the work was accomplished ahead of the construction work stoppage and no additional leaks have developed.

Drafting

The equivalent of 184 drawings was produced during the month for an average of 17.5 man-hours per drawing.

Major jobs in progress are: 280 ton extrusion press installation, PRTR "as-builts," new 7-rod fuel element for PRTR, shim rod control, electrical resistivity sample holder, scintillation scanner housing, 300 Area retention and crib waste piping, cladding cutter assembly for PRTR, Zr process tube 3 x 3 loop, remote welding chamber, and mark I-H end bracket.

There were 68 existing J. A. Jones Company orders at the beginning of the month with a total unexpended balance of $216,276. One hundred and forty-eight new orders, four supplements and adjustments for underruns amounted to $99,649. Expenditures during the month on Hanford Laboratories' work were $140,206. Total J. A. Jones backlog at month's end was $175,719.

	HL Unexpended Balance
Orders outstanding beginning of month	$216 276
Issued during the month (inc. Supp. and Adj.)	99 649
J. A. Jones Expenditures during month (inc. C.O. costs)	140 206
Balance at month's end	175 719
Orders closed during month	105 958

On June 13, 1962 carpenter pickets were removed and construction forces started back to work. On June 20, 1962 the ironworkers walked off the jobs but did not place pickets. Construction work has proceeded, where possible, on a curtailed basis. On Hanford Laboratories work a total of six jobs have stopped or could not be started and three more will stop early in July unless the ironworkers come back to work.

Because of the strikes, approximately $23,000 of the Hanford Laboratories FY 1962 allocation for miscellaneous capital work orders was not spent.

Construction and maintenance activities completed during June included:

141-C - Install grass pasture, trees and sprinkling system
141-C/141-F - Install desert coolers and paint
144-FR/146-FR - Move and reinstall desert coolers
144-F - Install air, vacuum and electrical service
108-F - Move incubators
241-C - Construct storage area and fencing
306 - Modify condensate system. Procedure filters
307 Basin - Repair 8" VCP waste line
321-A - Construct engineering test shop
325 - Install 440 volt bus duct

W. Sale
Manager
Finance and Administration

W Sale:whm

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74153

DATE 6-30-62

PROJ. NO.	TITLE	FUNDING
CAH-962	Low Level Radiochemistry Building	Funds Available to AEC

AUTHORIZED FUNDS			COST & COMM. TO	
$ 113,000	DESIGN $ 113,000	AEC $ *		$
	CONST. $	GE $ *	ESTIMATED TOTAL COST	$ 1,200,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN	7-23-62*	6-28-62	DESIGN	5-15-63*
CONST.	8-1-63*	DIR. COMP. DATE	CONST.	8-1-64*

ENGINEER

FEO - DS Jackson

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides a building in which extremely sensitive radioanalyses and methods development can be performed in an atmosphere protected from the environs. It consists of designing and constructing a building housing approximately 22,000 square feet of floor area including the basement.

The project proposal requesting $113,000 total design funds was submitted to the AEC, for authorization on April 30, 1962.

Directive No. AEC-2079 dated June 28, 1962 authorizing HOO-AEC total design funds in the amount of $113,000 has been issued.

* The AEC Work Authority has not been issued.

PROJ. NO.	TITLE	FUNDING
CAH-963	Geological & Hydrological Wells - FY-1962	62-k

AUTHORIZED FUNDS			COST & COMM. TO 6-10-62	
$ 80,000	DESIGN $ 1,400	AEC $ 68,500		$ 10,318 (GE)
	CONST. $ 78,600	GE $ 11,500	ESTIMATED TOTAL COST	$ 80,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN	5-18-62	5-9-62	DESIGN	6-1-62
CONST.	7-15-62	DIR. COMP. DATE 4-1-63	CONST.	4-1-63

ENGINEER

FEO -HE Ralph

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100		
PF			
CPFF	2	0	0
FP	98	0	0

SCOPE, PURPOSE, STATUS & PROGRESS

This project involves the continued drilling of special hydrological research, test and monitoring wells.

The construction contract was awarded to Haden Drilling Company for $56,207.50. Notice to proceed was mailed to contractor on June 22, 1962. Construction work will start concurrent with arrival of well casing sometime between July 1 and 15, 1962.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74153

DATE 6-30-62

PROJ. NO.	TITLE	FUNDING
CAH-958	Plutonium Fuels Testing and Evaluation Laboratory-308 Bldg.	62-k

AUTHORIZED FUNDS			COST & COMM TO	
$ 150,000	DESIGN $ 15,500	AEC $ *	COST & COMM TO	$ -0-
	CONST. $134,500	GE $ *	ESTIMATED TOTAL COST	$ 150,000

STARTING DATES			EST'D. COMPL. DATES	
DESIGN	8-1-62	DATE AUTHORIZED 6-22-62	DESIGN	11-1-62
CONST	11-1-62	DIR. COMP. DATE 5-15-63	CONST.	3-15-63

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides for the extension of plutonium research laboratories on the second floor of 308 Building by erection of plastered ceilings and walls to provide contamination control barriers. It also includes laboratory service extension and fabrication of a metallography hood.

The project was authorized by HOO-AEC June 22, 1962. [illegible] The project [illegible]

*AEC Work Authority has not been issued.

PROJ. NO.	TITLE	FUNDING
CAH-959	Graphite Machining Shop - 300 Area	62-k

AUTHORIZED FUNDS			COST & COMM. TO	
$	DESIGN $	AEC $	COST & COMM. TO	$
	CONST. $	GE $	ESTIMATED TOTAL COST	$ 105,000

STARTING DATES			EST'D. COMPL. DATES	
DESIGN		DATE AUTHORIZED	DESIGN	
CONST.		DIR. COMP. DATE	CONST.	

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides for a new graphite machining facility near the graphite storage building. The facility will permit greater flexibility in the handling and machining of graphite shapes as well as providing additional space for non-metallic materials testing in the area presently used for graphite machining.

The project proposal was submitted to the Commission for approval March 2, 1962.

The proposal was returned unapproved June 8, 1962.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74253

DATE 6-30-62

PROJ. NO.	TITLE	FUNDING
CGH-955	Reactivation of the H-1 Loop - 105-H Building	0490

AUTHORIZED FUNDS			COST & COMM. TO 6-10-62	$ 10,000
$ 10,000	DESIGN $ 10,000	AEC $	ESTIMATED TOTAL COST	$ 105,000
	CONST. $	GE $ 10,000		

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 4-15-62	DATE AUTHORIZED 3-29-62		DESIGN 8-30-62	
CONST. 7-15-62	DIR. COMP. DATE		CONST 12-15-62	

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	38*	11
TITLE I			
GE-TIT. II	100	38*	11
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide the primary test facility for determination of the feasibility of using aluminum-clad fuel elements in high temperature water by studying improved alloys and corrosion inhibitors.

AEC Directive No. HW-536, dated March 29, 1962 authorized $10,000 to initiate design and provide new cost estimate for review by the Commission. Design work is in progress.

*Taken from the Project Planning Schedule.

PROJ. NO.	TITLE	FUNDING
CGH-957	Small Particle Technology Laboratory - 325 Building	62-k

AUTHORIZED FUNDS			COST & COMM. TO 6-10-62	$ 4,050
$ 40,000	DESIGN $ 2,000	AEC $ --	ESTIMATED TOTAL COST	$ 40,000
	CONST. $ 38,000	GE $ 40,000		

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 4-23-62	DATE AUTHORIZED 3-21-62		DESIGN 5-31-62	
CONST. 7-15-62	DIR. COMP. DATE 11-1-62		CONST. 11-1-62	

ENGINEER

FEO - DS Jackson

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION (HLO)	0.5	25
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides laboratory space for research and development in small particle technology related to the generation, control, and disposal of radioactive wastes.

Directive HW-535, dated March 21, 1962, authorized total project funds in the amount of $40,000.

Detailed design is complete. A work order has been issued to J. A. Jones Construction Company for performance of construction.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- [illegible]

DATE 6-30-62

PROJ. NO.	TITLE	FUNDING
CAH-936	Coolant Systems Development Laboratory 1706-KE Building Addition	62-k

AUTHORIZED FUNDS			
$ 130,000	DESIGN $ 9,000	AEC $ 115,000	COST & COMM. TO 6-20-62 $ 24,906 (GE)
	CONST. $ 121,000	GE $ 15,000	ESTIMATED TOTAL COST $ 130,000

STARTING DATES			EST'D. COMPL. DATES
DESIGN 9-8-61	DATE AUTHORIZED 4-5-62*		DESIGN 1-1-62
CONST. 5-1-62	DIR. COMP. DATE 10-31-62		CONST. 11-30-62

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	3.5	48
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		166
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING	.4	4

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	22	7
PF			
CPFF			
FP	100	22	7

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides facilities for the conduct of corrosion and decontamination studies for nuclear reactor coolant systems, by the addition of 2,700 sq. ft. laboratory facility on the west side of the 1706-KE Building. Design was accomplished by the Bovay Engineers. Current estimate of Title I and II costs - $11,000.

*Original authorization for design was 8-9-61.

Continuation of concrete form placement started, after settlement of the carpenters' strike, on 6-14-62.

PROJ. NO.	TITLE	FUNDING
CHF-951	A-C Column Facility - 321 Building	0290

AUTHORIZED FUNDS			
$ 55,000	DESIGN $ 5,000	AEC $ -0-	COST & COMM. TO 6-20-62 $ 15,380
	CONST. $ 50,000	GE $ 55,000	ESTIMATED TOTAL COST $ 55,000

STARTING DATES		EST'D. COMPL. DATES
DESIGN 1-30-62	DATE AUTHORIZED 1-12-62	DESIGN 4-1-62
CONST. 3-25-62	DIR. COMP. DATE 10-31-62	CONST. 10-31-62

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II	0		
CONST.	100	31	31
PF	100	31	31
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide a closely integrated "A" Column in series with the relocated "C" Column to permit the development of a mathematical model for the mass transfer of uranium, as well as the exploration of the possibilities of computer optimization of a combined "A-C" extraction battery.

Relocation of "C" column is complete. Instrument line gutters are installed. Miscellaneous interconnecting piping work is continuing. "A" Column fabrication is in progress.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74.5?

DATE 6-30-62

PROJ. NO.	TITLE	FUNDING
CAH-922	Burst Test Facility for Irradiated Zirconium Tubes	62-k

AUTHORIZED FUNDS			COST & COMM. TO 6-10-62	$ 29,600
$ 29,600	DESIGN $ 29,600	AEC $		
	CONST. $	GE $ 29,600	ESTIMATED TOTAL COST	$ 289,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN 11-7-61		10-23-61	DESIGN 5-31-62	
CONST. 10-1-62		DIR. COMP. DATE To be established at a later date	CONST. 9-1-63	

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER - Bovay Engineers		260
DESIGN ENGINEERING OPERATION		200
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	57	100	100
AE-TIT. II	43	100	100
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facilities to permit deliberate destructive testing of irradiated zirconium tubing. This will provide operating and tube life data not available because of the limited operating history of Zircaloy-2 pressure tubing in reactors.

A project proposal revision requesting construction funds is being submitted for approval.

PROJ. NO.	TITLE	FUNDING
CAH-927	Additions to the 271-CR Building-Waste Treatment Demonstration Facility	61-j

AUTHORIZED FUNDS			COST & COMM. TO 6-10-62	$ 14,914 (GE)
$ 92,000	DESIGN $ 11,000	AEC $ 76,300		
	CONST. $ 81,000	GE $ 15,700	ESTIMATED TOTAL COST	$ 92,000

STARTING DATES	DATE AUTHORIZED	EST'D. COMPL. DATES
DESIGN 6-15-61	5-15-61	DESIGN 2-5-62
CONST. 2-15-62	DIR. COMP. DATE 7-31-62	CONST. 9-15-62

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		80
COST PLUS FIXED FEE		110
PLANT FORCES		
ARCHITECT - ENGINEER		150
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II			
AE-TIT. II	100	100	100
CONST.	100	73	50
PF			
CPFF	33	100	90
FP	67	60	30

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides facilities for pilot plant development of decontamination processes for intermediate level chemical processing plant waste for safe discharge to the plant environs. Design was accomplished by the Bovay Engineers.

Continued unsettled labor contracts have resulted in lack of progress on this construction work. Some concrete clean-up was accomplished but not significant enough to show for progress calculations.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74[illegible]

DATE 6-30-62

PROJ. NO.	TITLE	FUNDING
CAH-916	Fuels Recycle Pilot Plant	4-62-d-3

AUTHORIZED FUNDS			COST & COMM. TO	
$ 425,000	DESIGN $ 425,000 / CONST. $ -0-	AEC $ -- / GE $ 425,000	COST & COMM. TO 6-10-62	$ 384,900
			ESTIMATED TOTAL COST	$ 5,100,000***

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 3-15-61	DATE AUTHORIZED 6-22-62**		DESIGN 8-15-62	
CONST. 9-15-62*	DIR. COMP. DATE		CONST. 11-1-64	

ENGINEER

FEO - RW Dascenzo

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	94	94
TITLE I	11	100	100
GE-TIT. II	89	90	90
AE-TIT. II	0		
CONST.	100	0	0
PF			
CPFF			
FP			

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION	30	
GE FIELD ENGINEERING		

SCOPE, PURPOSE, STATUS & PROGRESS

This project is to provide a facility to perform a full scope of engineering tests and pilot plant studies associated with fuel reprocessing concepts. HOO-AEC approved the revised project proposal requesting total project funds of $5,100,000 and sent it on to Washington AEC for final approval.

Design is continuing on all phases as scheduled.

No additional information has been received from Washington AEC, concerning possible modifications to FRPP design to accommodate an AEC Waste Calcination Program.

Directive No. AEC-187, Mod. No. 3, dated June 22, 1962, was issued increasing design expenditures to $425,000 from $385,000 as an interim authorization for continuation of design. Work Authority No. CAH-916 (4) dated June 22, 1962 increased the General Electric authorization to $425,000 for design.

A total of 306 drawings have been issued for comment and 87 for approval. The specifications are 70% complete.

*Estimated construction starting date for removal of burial ground fill.

** Original authorization for initiation of design was February 9, 1961. June 22, 1962 is the authorization date for the last design supplement.

*** Including transferred capital property valued at $100,000.

Project Proposal, Revision No. 4, dated June 25, 1962 has been written by AEC requesting an additional $40,000 for redesign of FRPP to include modifications to accommodate AEC's Waste Calcination Program.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74153

DATE 6-30-62

PROJ. NO.	TITLE	FUNDING
CAF-888	Biology Laboratory Improvements	60-h-1

AUTHORIZED FUNDS			COST & COMM. TO 6-10-62	
$ 420,000	DESIGN $ 44,000	AEC $ 359,500	$ 54,520 (GE)	
	CONST. $ 376,000	GE $ 60,500	ESTIMATED TOTAL COST $ 420,000	

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 8-8-60	DATE AUTHORIZED	4-18-61*	DESIGN	3-31-61
CONST. 7-10-61	DIR. COMP. DATE	3-31-62	CONST.	8-1-62**

ENGINEER

FEO - JT Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		2650
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	NS	100
TITLE I			
GE-TIT. II	17	NS	100
AE-TIT. II	83	NS	100
CONST.	100	100	88
PF	1	100	100
CPFF	10	NS	0
FP	89	100	97

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides additional space for biological research supporting services, and involves an addition to the 108-F Building.

The Radiation Source Handling Facilities were received at Hanford on 6-4-62. The console has been delivered to the 108-F Bldg. All other crates cannot be moved due to the steel workers' strike.

Laying of vinyl floor covering was completed after the carpenters' strike was settled. Air balancing has been resumed. A preliminary inspection was made on June 14, 1962 and the contractor was supplied with a list of uncompleted items. The radiation and control rooms were accepted with exceptions to permit J. A. Jones forces to start installation.

**Estimated project completion date provided the strike has been settled by 7-1-62.

*Original authorization for design was May 3, 1960.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74153

DATE 6-30-62

PROJ. NO.	TITLE	FUNDING
CAH-867	Fuel Element Rupture Test Loop	58-e-15

AUTHORIZED FUNDS			COST & COMM. TO 6-10-62	$ 550,536 (GE)
$ 1,500,000	DESIGN $ 130,000	AEC $ 820,000	ESTIMATED TOTAL COST	$ 1,500,000
	CONST. $ 1,370,000	GE $ 680,000		

STARTING DATES				EST'D. COMPL. DATES	
DESIGN	8-1-60	DATE AUTHORIZED	6-24-60*	DESIGN	3-15-61
CONST.	11-2-60	DIR. COMP. DATE	6-30-62	CONST.	10-31-62

ENGINEER

IR&AO-MEEO - PC Walkup

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	5	2385
COST PLUS FIXED FEE		
PLANT FORCES	0	1940
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	91	100	100
AE-TIT. II	9	100	100
CONST.	100	100	95
PF	2	100	50
CPFF	57	100	97
FP (1)	10	100	100
(2)	31	100	95

SCOPE, PURPOSE, STATUS & PROGRESS

(1) G. A. Grant Company
(2) Lewis Hopkins Construction Company

This facility is to be used for fuel rupture behavior studies with respect to physical distortion and rate of fission product release.

Project is behind official schedule because of delays in delivery of material and due to CPFF labor strike. A revised project proposal has been submitted which extends the project completion date to October 31, 1962.

*Initial authorization was on 10-1-59.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS			COST & COMM. TO	$
$	DESIGN $	AEC $	ESTIMATED TOTAL COST	$
	CONST. $	GE $		

STARTING DATES				EST'D. COMPL. DATES	
DESIGN		DATE AUTHORIZED		DESIGN	
CONST.		DIR. COMP. DATE		CONST.	

ENGINEER-

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74153

DATE 6-30-62

PROJ. NO.	TITLE	FUNDING
CAH-866	Shielded Analytical Laboratory - 325-B Building	61-a-1

AUTHORIZED FUNDS			COST & COMM. TO 6-10-62	$ 131,643 (GE)
$ 700,000	DESIGN $ 60,000	AEC $ 546,500	ESTIMATED TOTAL COST	$ 655,000
	CONST. $ 640,000	GE $ 153,500		

STARTING DATES			EST'D. COMPL. DATES
DESIGN 9-5-59	DATE AUTHORIZED 5-31-60*		DESIGN 11-14-60
CONST. 6-15-61	DIR. COMP. DATE 6-30-62		CONST. 10-30-62

ENGINEER

FEO - RW Dascenzo

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	0	1760
COST PLUS FIXED FEE		
PLANT FORCES	0	
ARCHITECT-ENGINEER		3
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING	1	

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	10	100	100
AE-TIT. II	90	100	100
CONST.	100	100	60
PF	3	1	1
CPFF	2	0	0
FP	95	100	60

SCOPE, PURPOSE, STATUS & PROGRESS

This project will allow greater capacity for analytical work involving today's more highly radioactive solutions and consists of adding a shielded laboratory to the 325 Building.

*Original authorization for preliminary design was August 12, 1959.

The contractor's completion date has been extended from April 15, 1962 to June 24, 1962. This does not reflect lost time due to the carpenters' strike which lasted from 5-16 to 6-13.

The 3 rejected lead glass viewing windows that were reviewed have been accepted by AEC with a contract modification.

A revised project proposal has been prepared, but not approved by AEC, extending the completion date, reducing the total authorized project funds and increasing G. E.'s field engineering authorization.

The building has been furred for plastering; a new window and door have been cut into the common wall with the 325 Bldg; 95% of the roof has been put on; cell wall concrete partially finished; first floor electrical runs partially installed and preparations initiated for the in-cell conveyor.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 74253

DATE 6-30-62

PROJ. NO.	TITLE	FUNDING
CGH-857	Physical & Mechanical Properties Testing Cell - 327 Bldg.	0290

AUTHORIZED FUNDS			COST & COMM. TO 6-10-62	$ 284,220
$ 460,000	DESIGN $ 45,000	AEC $ --		
	CONST. $ 415,000	GE $ 460,000	ESTIMATED TOTAL COST	$ 460,000

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 11-2-59	DATE AUTHORIZED 9-22-61*		DESIGN 3-15-61	
CONST. 2-12-62	DIR. COMP. DATE 12-15-62		CONST. 12-15-62	

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		34
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		833
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	2	2
PF			
CPFF	18	9	9
FP			
Equip	82	0	0

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facities for determining physical and mechanical properties of irradiated materials, and involves the installation of a cell in the 327 Building.

Current estimate of Title I and II costs - $55,000. Detailed design started 4-1-60. Procurement and construction authorized 9-22-61.

Basement floor and foundation concrete work is completed. Construction has stopped until cell assembly is delivered.

Number of purchase orders required	19 Value (Est.)	$253,000**
Number of purchase orders placed	19 Value	203,000

*Original authorization for design was October 1, 1959.
**Includes delivery charges, inspection and contingency.

Vendor mistakes on cell castings have included boring holes over tolerance, boring holes off-center, machining castings out-of square and poor workmanship on the cell liner.

The completion of the cell assembly remains the most critical item in the completion of this project.

It appears that considerable time and effort will be required to correct the deficiencies and expedite the completion of this order.

SEMI-MONTHLY PROJECT STATUS REPORT

HW-	74153
DATE	6-30-62

GENERAL ELECTRIC CO. — Hanford Laboratories

PROJ. NO.	TITLE	FUNDING
CAH-822	Pressurized Gas Cooled Facility	4141 Operating

AUTHORIZED FUNDS			COST & COMM TO 6-17-62	$ 1,129,996
$1,170,000	DESIGN $ 43,000 / CONST. $ 1,127,000	AEC $ 15,000 / GE $ 1,155,000	ESTIMATED TOTAL COST	$ 1,170,000

STARTING DATES		DATE AUTHORIZED	2-2-62*	EST'D. COMPL. DATES	
DESIGN	8-19-59			DESIGN	4-29-60
CONST.	10-17-60	DIR. COMP. DATE	6-30-62	CONST.	10-31-62

ENGINEER

TR&AO-MEEO - DP Schively

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100	100	91
PF	1.4	100	0
CPFF	22.1	100	99
FP	6.6	100	100
Govt.Eq	69.9	100	90

SCOPE, PURPOSE, STATUS & PROGRESS

Struthers Wells has all components but is unable to complete assembly and test of heater before August 15, 1962.

The Supervisor, Maintenance & Equipment Engineering, witnessed preliminary gas-bearing blower tests at the Bristol Siddeley Plant. Preliminary tests indicate completion of two blowers, including testing, by August 15, 1962.

Project Proposal, Revision V is being circulated for approval. This revision requests extension of completion date to 10-31-62 with no increase in project funds.

*Initial authorization date was December 18, 1958.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS			COST & COMM. TO	$
$	DESIGN $ / CONST. $	AEC $ / GE $	ESTIMATED TOTAL COST	$

STARTING DATES		DATE AUTHORIZED		EST'D. COMPL. DATES	
DESIGN				DESIGN	
CONST.		DIR. COMP. DATE		CONST.	

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

TEST REACTOR AND AUXILIARIES OPERATION

JUNE 1962

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output was limited to 185 MWD for a plant efficiency of 8.8%. Because of the low operating efficiency, exposure data normally included in this report did not change significantly during the month and will not be presented.

The reactor was shut down on June 3, 1962, for refueling and process tube examination. Seventy-six process tubes, four UO_2 elements, three LX Pu-Al elements and one mixed oxide element were inspected during the outage. All process tubes evidenced wear corrosion. Two tubes were more severely corroded than the remainder (22 and 26 mil deep marks, respectively). These were removed and burst tested. Tests demonstrated greater hoop stress strength than for unirradiated tubes. One Pu-Al element was found to have lost a band, and was removed to the storage basin. Another Pu-Al was removed when the results of stagnant water tests indicated that it probably had defective cladding. A UO_2 element showed severe wear on the spiral bundle wrapping wire and was removed to storage.

Nine fresh mixed oxide elements and one special MgO-PuO_2 spike element were charged. The June 28th core loading was composed of 37 Pu-Al elements, one MgO-PuO_2 element, 27 UO_2 elements and 20 mixed-oxide elements.

A "shim free" approach to critical, made on June 28, predicted a cold, clean critical level of 63 inches.

A piece of 304 SS found in the inlet jumper of a process tube led to radiographic examination of the flow straightening vane upstream of the bulk flow venturi, which confirmed damage to the straightening vane. At month end the reactor was discharged of all fuel elements in preparation to draining the primary system to further investigate and repair the vanes and evaluate possible corrosion problems throughout the system.

D_2O and helium losses were 3694 lbs and 66,600 scf, respectively. High D_2O losses are attributed to extensive charge-discharge and tube inspection work which required the system to be open. In addition, a D_2O spill from which 7440 lbs of D_2O were recovered resulted in an additional loss estimated at 150 lbs from stack gas condensate sampling. The spill came about during preparations to remove a process tube while another process tube was partially disconnected; a combination pressure and syphoning action caused D_2O to drain from the primary system into the upper and lower access spaces. The water loss was stopped after about 5 or 6 minutes.

Equipment Experience

A weld failure was found on the 26-inch expansion spool bellows providing containment between the vessel and the export steam line. About a 120^{o} of seam weld on one of the folds had failed. Repairs were made and the rest of the bellows including the one on the relief line was thoroughly inspected and leak tested. No leaks or other defects were found.

No serious difficulties were encountered during removal of three process tubes.

The inlet jumper between the angle valve and venturi on tube #1542 was replaced when the nut galled during removal.

About two tablespoons of rust, metal shavings, etc., were removed from a high pressure helium valve and adjacent piping. During the course of repairing the above valve, an isolation valve galled between the stem and bonnet. It was removed and modified parts fabricated to effect repairs. Additional modifications are planned for it and other valves of the same type to prevent this problem in the future.

A number of problems were encountered during the annual test and overhaul of a high pressure helium safety relief valve. Although the valve was set and relieved at the proper pressure, it would not reseat. It was found after several trials that the bevel on the nozzle retainer was too large and would not allow the "O" ring to seat properly. A new retainer was made.

Both boiler feed valves were repaired this month. A large piece of welding stag was found in one.

Bearings in Moderator Pump Motors #2 and #3 were replaced. Bearings in Motor #2 had been in service less than three months and those in Motor #3 less than a month. Precision bearings were installed in these motors and their performance will be watched closely.

The FEEF manipulator was installed this month but gear noises required removal for additional repair.

Two repaired shim rods were installed in place of defective rods. The replacement rods had received limited previous exposure. No significant difficulties were encountered in the repair or replacement.

Programmed maintenance required 571 manhours or 11% of the total available manhours.

Improvement Work Status (significant items)

Work Completed:

Permanent mounting of second startup channel
Relocation of Pressurizer Pressure Transmitter
Leak collection tank modifications
Instrumentation for Helium System Oil Absorbers
Installation of Experimental Low Level Effluent Monitor
Replacement of Generator on shipping cask

Work Partially Completed:

Safety circuit ground and low voltage detector
Outlet nozzle cap modification (now 90% complete)
Fueling vehicle hoist modification
Reactor core liquid level instrumentation
Primary Oxygen Analyzer Installation
Shim rod readout modification
Prototype replacement inlet gas bellows
Core blanket system piping modifications
Flanges for safety relief valves in helium system
Rupture Monitor sample line changes
Chain barricade for rotating shield

Design Work Completed:

Enlarge Chemical Feed system
Decontamination Facility
Modification to storage basin crane

Primary pump recording ammeters
High pressure helium compressor inter-after cooler relief
Outlet nozzle bracing
Fuel transfer pit hoist drain
Modification to helium valve stems
Interlock between charge-discharge machine, shroud seat and discharge hoist

Design Work Partially Completed

Control room ventilation
Additional fuel storage and examination layout
Oil storage building
Boiler feed pump seals
Third exhaust air activity channel
Compressed air supply revision
Fuel transfer system modifications

Process Engineering and Reactor Physics

Calculations of heat generation in Moxtyl fuel elements which take into account the neutron flux depression due to the higher cross sections indicate a net generation of 1.35 times that for a natural UO_2 element. This calculated heat generation rate is within 7% of the observed rate.

The data obtained from PRTR Test Number 15 (HX-1 Enclosure Temperatures) showed a gradual buildup of the air temperatures in the enclosure to about 155 F. An inspection of the concrete surfaces in the enclosure showed no signs of excessive dehydration due to the high temperatures.

Procedures

Revised Operating Procedures issued			5
Revised Operating Standards issued			10
Temporary Deviations to Operating Standards issued			4
Revised Process Specifications accepted for use			0
Maintenance Manuals issued			2
Maintenance Procedures issued			0
Drawing As-built Status	June	Total	
Approved for as-built	94	656	
Ready for approval		75	
In drafting		50	
Voided		46	
No change required		85	
		912 - 82% of total	

Personnel Training

Qualification subjects	120 manhours
Specifications, Standards, Procedures	31
Fueling Vehicle	10
	161 manhours

Status of Qualified Personnel at Month End

Qualified Reactor Engineers	10
Qualified Technicians	6
Qualified Technologists	19

Plutonium Recycle Critical Facility

Design Test Status:

Electrical	100%
Instrumentation	96
Fuel Handling	25
Moderator System	100
Thimble Coolant	75

Improvement Work Status:

Work Completed:
- Installation of D_2O purification system
- Installation of flange in vent line from weir
- Removal of PRCF cell radiation monitor from PRTR annunciator
- Rerouting of weir overflow-reactor drain and moderator addition lines to the moderator storage tank

Design Work:
- Installation of interlock to prevent loss of transfer lock seal air or loss of process water pressure

The safety rods were inspected and found satisfactory. A new cable was installed on a telephone line which had been causing spurious trips in the startup channels.

Operation

All Critical Facility and two PRTR assigned personnel satisfactorily completed written examinations. A total of 62 CFO and PRTRO personnel manhours was devoted to Critical Facility training.

Fuel Element Rupture Test Facility

Project Status (Project CAH-862)

A revised project proposal requesting a completion date extension to October 31, 1962, is circulating for approval. Design of B cell shielding is complete. Several design changes are in process as a result of detailed hazards analysis. Over-all construction is estimated at 96% complete.

Operation

Training sessions for operation of the filter plant were initiated. Considerable work on training aids, manuals and procedures was accomplished.

GAS COOLED POWER REACTOR PROGRAM

Project Status (Project CAH-822)

The project is 91% complete. A project proposal revision requesting a completion date extension to October 31, 1962, is being circulated for approval. Preliminary testing of gas blowers was witnessed at the vendor's plant. Bearing instability problems appear to have been solved and final assembly of blowers is under way. Heater delivery has been delayed and is now scheduled for August.

Operation

Classroom training was completed. A total of 314 manhours was devoted to training. Written qualification tests were taken by the majority of personnel. Work on manuals, procedures, data sheets, and startup tests continued.

TECHNICAL SHOPS

Total productive time for the period was 22,499 hours. This includes 14,677 hours performed in the Technical Shops, 7332 hours assigned to Minor Construction, 375 hours assigned to off-site vendors, and 115 hours to other project shops. Total shop backlog is 19,474 hours, of which 70 percent is required in the current month with the remainder distributed over a three-month period. Overtime hours work during the month was 8.9 percent (1702.1) of the total available hours.

Distribution of time was as follows:

	Man-hours	% of Total
Fuels Preparation Department	5,270	23.42%
Irradiation Processing Department	2,052	9.12%
Chemical Processing Department	978	4.35%
Hanford Laboratories Operation	14,199	63.11%

Requests for emergency service required an overtime ratio which is in excess of that normally experienced by this operation. Factors influencing the overtime rate included the urgency associated with several large research and development projects within HLO, plus the shop support required by CPD in the Weapons Program and FPD in the program to develop rail supports for NPR fuel elements.

WD Richmond

Manager
Test Reactor and Auxiliaries

WD Richmond:dde

INVENTIONS OR DISCOVERIES

All persons engaged in work that might reasonably be expected to result in inventions or discoveries advise that, to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during the period covered by this report except as listed below. Such persons further advise that, for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

INVENTOR	TITLE OF INVENTION OR DISCOVERY
C. H. Bloomster	Nuclear Fuel Materials - A Plutonium-Zirconium Alloy HW-73910. June 4, 1962.
D. L. Ballard and C. W. Harrison	Design of Highly Flexible Metal Gasket

H M Parker

Manager, Hanford Laboratories

END

DATE FILMED

4/8/93