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HANFORD LABORATORIES OPERATION MONTHLY ACTIVITIES REPORT

FEBRUARY, 1962

MARCH 15, 1962

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

GENERAL ELECTRIC

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HANFORD LABORATORIES OPERATION
MONTHLY ACTIVITIES REPORT

FEBRUARY, 1962

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Compiled by
Operation Managers

March 15, 1962

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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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2	H. M. Parker
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5	F. E. Creter, GEAPD
6	I. H. Dearney
7	W. E. Foust
8	W. S. Frank
9	P. F. Gast
10	A. R. Keene
11	H. A. Kornberg
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15	H. P. Shaw - W. P. McCue
16	W. K. Woods
17 - 20	Atomic Energy Commission, Hanford Operations Office Attn: L. E. Travis
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TABLE I. HIO FORCE REPORT

	DATE: February 28, 1962			
	At beginning of month		At close of month	
	Exempt	Salaried	Exempt	Salaried
Chemical R & D	129	113	132	115
Reactor & Fuels R & D	170	155	174	152
Physics & Instrument R & D	90	59	88	61
Biology	36	52	36	51
Operations Res. & Syn.	18	4	17	4
Radiation Protection	41	92	41	90
Finance and Administration	134	210	116	102
Programming	18	3	17	3
General	3	4	3	4
Test Reactor & Auxiliaries	35	74	46	182
TOTAL	674	766	670	764
				1,434

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BUDGETS AND COSTS

February operating costs totaled \$2,197,000; fiscal year-to-date costs are \$17,590,000 or 63% of the \$27,764,000 control budget. Hanford Laboratories' research and development costs for February, compared with last month and the control budget are as follows:

(Dollars in Thousands)	C O S T			Budget	% Spent
	Current Month	Previous Month	FY To Date		
HLO Programs					
02 Program	\$ 39	\$ 43	\$ 348	\$ 605	58
03 Program	4	-	4	175	2
04 Program	894	727	7 239	10 925	66
05 Program	98	66	595	1 088	55
06 Program	194	163	1 567	2 720	58
	<u>1 229</u>	<u>999</u>	<u>9 753</u>	<u>15 513</u>	<u>63</u>
FPD Sponsored	96	74	890	1 400	64
IPD Sponsored	124	98	879	1 325	66
CPD Sponsored	<u>125</u>	<u>122</u>	<u>1 067</u>	<u>1 609</u>	<u>66</u>
Total	<u>\$ 1 574</u>	<u>\$ 1 293</u>	<u>\$12 589</u>	<u>\$19 847</u>	<u>63</u>

RESEARCH AND DEVELOPMENT1. Reactor and Fuels

Examination of inner components of N fuel elements after irradiation to 2300 MWD/T revealed adequate behavior of the braze and heat-affected zone and no indication of collapse of the inner bore. The first irradiation of a fully prototypical N fuel element has reached 700 MWD/T in the ETR without incident.

Beta heat treatment of Zr/U coextrusion billets has resulted in a pronounced improvement in inner clad thickness control in coextrusion of NPR fuel tubes.

A nondestructive test has been developed which permits the detection of cylindrical asymmetry in the grain structure of uranium billets.

Ex-reactor rupture tests on irradiated prototypical N fuel elements again confirmed that the severity of rupture behavior is much less using the NPR scram cool-down rate than when using the normal cool-down rate.

A coextruded fuel element containing thermocouples to measure the core and water temperatures, and thereby monitor "crud" deposition effects, was successfully charged into a KER loop.

Cross travel life testing of the N Reactor charging machine was completed. N Reactor fuel element footwork tests were continued with indications of galling between autoclave feet and autoclaved nozzle.

Preparations were completed for laboratory investigation of heat transfer conditions and fuel temperatures which would result from coolant flow loss following a failure of a hydraulic fitting at the inlet of a process tube at C Reactor.

Monitoring of graphite burnout samples from D Reactor continue to indicate a peak oxidation rate about nine feet in from the front of the graphite stack.

Prolonged irradiation, 3.2×10^{21} nvt ($E > 0.18$ Mev), at 1150 C apparently increased the oxidation rate of graphite by a factor of two. Exposures of 10^{20} nvt at 550 C did not alter the oxidation rate of similar graphites.

The irradiation at 500 to 600 C of a series of graphites processed at final graphitization temperatures from 1370 to 2700 C has been extended to 8000 MWD/AT. The higher the initial graphitization temperature, the lower the initial contraction rate. However, at high exposures the contraction rates tend to equalize for all graphitizing temperatures.

Removal of the oxidants (O_2 and H_2O) from a gas mixture of 99% helium - 1% hydrogen resulted in accelerated hydriding of preautoclaved Zircaloy-2 samples exposed to this mixture at temperatures as low as 200 C.

In the temperature range 300 to 400 C, Zircaloy-2 was found to have an activation energy for creep in-reactor which varied with the mode of measurement; during heating, the value was 85,000 cal/mol but during a cooling cycle it was 60,000 cal/mol. These measurements are the first of their kind ever attempted in-reactor and promise to furnish an understanding of the dynamics of damage processes.

Creep rupture tests at reactor operating temperatures are continuing on sections of 18% cold worked NPR tubes and annealed PRTR tubes.

Work was initiated on the new Division of Reactor Development sponsored program for the study of irradiation damage to reactor metals. At this time, it is proposed to include austenitic stainless steels, nickel base alloys an iron aluminum alloy, and a number of low alloy ferritic steels in this program.

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In comparing the ultrasonic response from a standard groove in a Zircaloy-2 sheath tube using flat unfocused and focused transducers, it was found that a much stronger response was obtained over a wide range of entry angles and lengths of metal path for the unfocused transducer.

Six PRTR pressure tubes monitored in-reactor during February appear satisfactory though gas gaps for two tube channels were near minimum.

The analysis of the heat transfer conditions that would exist in the PRTR following a loss of power to primary pumps was continued. It was concluded that cooling without boiling would be desirable in such an event but that if flow by free convection were lost, the fuel elements could be cooled adequately by boiling provided the fuel was covered with water at all times.

The final report on PRTR critical test results, HW-61900 BA, was completed.

Testing of the Critical Facility fuel element transfer thimble has shown that it cannot be manipulated satisfactorily through the transfer lock. It may be necessary to reduce the size of the cooling hoses attached to the thimble.

A total of 67 Mark I-H, 19-rod cluster aluminum low-exposure plutonium alloy fuel elements have been fabricated for PRTR irradiation. This completes this fabrication campaign. Components for fabrication of 60 swaged UO_2 - PuO_2 PRTR fuel rods are on hand and loading began during the week of February 26. Fuel materials and cladding components for an additional 60 swaged rods are being prepared.

The first cycle of irradiation for the aluminum-plutonium injection cast seven-rod cluster (GEH-11-6) has been successfully completed in the ETR 3 x 3 loop. The UO_2 - PuO_2 , cosine-enriched, seven-rod cluster element (GEH-11-8) is operating successfully in the ETR 3 x 3 high temperature loop facility.

Two zirconium-clad, plutonium-Zircaloy core fuel plates were made by roll cladding.

Synthesis of PuN was accomplished by arc-melting a small plutonium metal rod under one atmosphere of nitrogen. The melting point of PuN in argon is 2450 ± 50 . Plutonium mononitride is not compatible with boiling water.

Vibratory compaction techniques adaptable to remote operation have been furthered by the technique of suspending the fuel element from a vibrating beam. This development permits location of the vibrator outside of the potentially contaminated work area.

Post-irradiation examination of a UO_2 single crystal, irradiated to 7400 MWD/T, revealed significant increases in hardness, a significant increase in melting point, and small change in surface texture. The fission gas release from the sample during irradiation was negligible.

Post-irradiation examination of swaged UO_2 , irradiated to 37,000 MWD/T at very high temperatures, revealed extremely long, thin, radially oriented columnar grains quite different from those observed in fuel irradiated to lower exposure.

Simulation of in-reactor fuel heating has been demonstrated by relatively simple direct resistance heating of UO_2 in a closed capsule. Heat fluxes and temperature gradients very similar to those achieved in-reactor have been accomplished.

2. Chemical Research and Development

Laboratory tests indicate that the removal of arsenic-76 parent material from reactor process water at pH 6.7 is optimized by the addition of 8 ppm alum, vice 18 ppm.

Effluent water from two reactor tubes charged with nickel clad slugs is found to contain a phosphorus-32 concentration twenty-fold greater than that of the total effluent.

A core drilling program is being undertaken on the 216-Z-9 Recuplex crib to determine the extent of plutonium migration.

Preliminary experiments indicate that over 99.5% of the neptunium present in a synthetic formaldehyde treated Purex waste can be extracted in a single batch contact with an equal volume of 0.37 M D2EHPA - 0.2 M TBP-Shell Spray base.

Using a redesigned steel core bit, the sludge in Purex underground waste storage tank, 241-A-103, was completely penetrated. The sludge appeared to have a hard crust, with a soft underlayer.

The mineral clinoptilolite was found capable of removing 98% of the fission product cesium present in 17 column volumes of actual alkaline Purex waste supernate. Over 99% of the cesium can be eluted from the mineral with three column volumes of 5 M NH_4NO_3 at 50 - 60 C. The removal of strontium from formaldehyde treated acidic Purex waste with clinoptilolite is not practical for process application. Additional laboratory results continue to support the feasibility of removing fission product cesium from Purex wastes by extraction with a dipicrylaminenitrobenzene solvent.

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Technetium-99 recovered from hot cell experiments with Purex 103A waste supernate has been converted to the sulfide (Tc_2S_7). The sulfide is used for the preparation of additional technetium compounds.

Treatment of strontium peroxide and strontium carbonate powders with two weight per cent lithium fluoride at elevated temperatures reduces the bulk material to a dense fused product.

The feasibility of using a X-ray spectrometer to determine lattice parameters of iron crystals as a measure of strain near pipe welds was demonstrated.

Electrodeposition studies of molten chloride solutions show that crystalline UO_2 deposits can be prepared containing, in solid solution, at least 14 weight per cent ThO_2 .

Assembly of the radiant heat spray calciner in a cold mock-up area prior to final installation in a high level cell is nearly complete. Initial component shakedown tests were quite satisfactory.

A Micro Pilot Plant run was terminated when cesium-137 in the effluent from the Amberlite IR-120 column exceeded its MPC_w after the treatment of 10,800 column volumes of Purex Tank Farm condensate waste.

3. Physics and Instrumentation Research and Development

The instrumentation for testing the nuclear purity of the NPR graphite stack was completed and moved to the reactor for installation for the March 3 tests.

NPR simulator technical scoping continued with major emphasis on developing more detailed simulation circuits and providing IPD additional information needed for their preparation of a project proposal.

An eleven-region reactor simulation is being developed to study both the space and time effects on the transient behavior of reactor safety instrumentation. Simulation of 11 nodes will require all the available EASE and GEDA analog equipment. A new circuit was developed to simulate the delayed neutron effect in each region with much less equipment than required for previously used techniques.

Technical analyses and consultations were provided IPD on the NPR Fuel Failure Monitor, the Integrated Data and Temperature Monitor System, the Rod Position Display System, and the Primary Loop Control System.

The properties of concrete reflectors continued under study at the Critical Mass Laboratory. Results indicate that such material reaches full effectiveness at smaller thicknesses (about four inches) than does water and than had been previously expected for concrete.

Improvements in calculational methods for the Plutonium Recycle Program were promised by the discovery of two quite different methods of attacking the problem of computing isotopic changes resulting from reactor operation. The two methods may be arranged to check each other, thereby increasing the reliability of the results. In other code development work, some progress was made in the search for difficulties in the RBU, and methods for improving the physics sophistication of MELEAGER were formulated.

More basic theoretical developments included a promising method for predicting the neutron spectrum in the thermal region. This method avoids time-consuming complexities but appears to reproduce experimental results through the use of a fictitious variable mass for the moderator nuclei in order to account for crystalline and molecular binding effects.

Phoenix fuel burnup experiments may be feasible in the PRTR to supplement data now being obtained at the MTR. Preliminary analyses indicate the hard spectrum required can be obtained by proper choice of fuel and poison compositions.

Light water lattices containing Pu-Al fuel rods of 1.8% plutonium concentration were investigated in a continuing experimental program to obtain data on the critical size of such lattices.

Studies continued on combined fast-thermal reactor complexes for plutonium fuel cycling, and on other uses of plutonium fuels such as fast-ceramic compact reactors for space applications and oxide breeders for special materials production. The feasibility of cycling fuel elements between fast and thermal reactors without reprocessing is also being studied. Preliminary results on the compact reactors indicate that plutonium may offer considerable weight savings compared to U^{235} fueled systems.

One transistorized liquid effluent activity monitor for the PRTR containment trip circuit is being fabricated for immediate application. On the basis of recent development work, this new system is designed to monitor for fission product activities from 10^{-5} $\mu\text{c/cc}$ to 10^{-1} $\mu\text{c/cc}$ with much more reliability than obtained with the originally installed equipment.

All the remaining (34) spare PRTR pressure tubes were nondestructively tested and prepared for autoclaving.

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In the Neutron Flux Monitor Program, a dual chamber fission counter was completed for use in experiments to measure the performance of breeder type detectors. Calculations were made also regarding the possible uses of Nuclear Magnetic Resonance, Electron Paramagnetic Resonance, and Microwave Spectroscopy for in-core flux monitoring at high temperature and flux.

A new method is being developed for nondestructively determining whether or not welds in steel pipes have been stress relieved. It is based on the difference in broadness of X-ray diffraction lines from stress-relieved and nonstress-relieved cold-worked metals recently observed in laboratory experiments.

Recent California weather has produced a major setback to the Atmospheric Diffusion Program conducted under our direction at Vandenberg Air Force Base. The heavy rains washed out roads and flooded sampler locations. Extensive repairs will be required before resumption of a full-scale program.

Sensitivity of the developmental continuous ZnS particle monitor for air diffusion experiments was improved a factor of four, to 10^{-8} gm/ft³, by changing from a dc to a pulse counting method. This is just a factor of five less than the sensitivity of the filter collection method now used.

Good progress was made in the development of a new data logging system for the Atmospheric Physics experimental portable meteorology mast.

In the whole body counting work the techniques for measuring the burden of non-gamma emitting materials, such as P-32, were improved through studies of counter and subject arrangements.

Coincidence-counting techniques previously developed for continuous monitoring of airborne Pu activity are being successfully applied to provide an immediate measurement of alpha activity collected on filters without the usual four-to-five hour wait for decay of radon-thoron products.

4. Biology

Attempts again failed to increase the virulence of columnaris by producing a mutant strain by radiation.

Some interesting dose-distribution data were obtained from a pig that had been fed 3.1 mc Sr-90/day for 118 days. (Results are listed on Pages D-1 and D-2 of this report.) Of the dose rates measured in soft tissue with a Victoreen pocket dosimeter, those for the ovaries were highest, due to their close proximity to the colon. Radioanalysis of the skeleton is in progress.

[REDACTED]

A sow that had received 625 μc Sr-90 per day for about six months gave birth to a litter of four. Two of the offspring were normal in size and weight, and two were slightly subnormal. Estimated skeletal dose rates are five to eight rads at birth and 30 to 40 rads per day at weaning.

Apparently Np-237 exerts its toxicity as a heavy element rather than as a radioactive substance. A relatively high dose administered to a sheep caused fatty degeneration of the liver, hemorrhages in the ileum and mesentery, and impaired liver function.

Small fish have been maintained successfully for one week in water containing 40% D_2O . A content of 50% was found to be lethal.

Work with Cs-137 in the root environment of coleus plants suggests that the polar hypothesis, which states that ions travel in a vascular bundle without crossover, should be revised.

5. Programming

Preliminary consideration has been given to the possibility of producing milligram quantities of extremely pure mercury-204 for production of a few small mercury vapor lamps. The technique would involve the irradiation of thallium to produce some thallium-204, which in part of its decay process produces mercury-204. It was also found feasible to conduct such irradiation in PRTR and at the same time produce sufficient thallium-204 for study as a radionuclide heat source (four year half life, pure beta emitter, about 0.7 watt per gram of pure thallium-204). It is indicated that such a small program can be accommodated in PRTR without adversely affecting the PRP.

The second in a series of formal reports describing work on fuel cycle analysis was issued. This report, "QUICK - A Simplified Fuel Cost Code," HW-71812, describes the economic model and IBM-7090 computer code used to calculate fuel cost.

TECHNICAL AND OTHER SERVICES

In connection with the characterization and evaluation of the effects of coextrusion process variables on warp, several test designs have been submitted in an attempt to isolate the effects of several variables which are almost completely confounded in existing process data.

In contrast to the success of the linear programming forecasting model, which has now become an accepted part of the Fuels Preparation Department forecasting procedure, results from the first trial run of a similar model for manufacturing

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scheduling indicate conclusively that in this case discrete and continuous variable optima do not coincide. This precludes the use of the standard linear programming techniques. Although a well-known algorithm does exist for obtaining the desired discrete variable optimum, the additional programming and computation involved virtually prohibit its use.

The first experimental magnetic tape to control the Gorton lathe have been prepared using the computer program developed for this purpose.

A presentation was made to area analysts in connection with the objective appraisal of the effectiveness of down time utilization, and a report was written presenting the results of a preliminary analysis of 44 weeks of Irradiation Processing Department overtime data.

A numerical method was devised for measuring the thermal conductivity of materials at extremely high temperatures. The method is based on the use of measured temperatures at selected points on the sample in formulas obtained from approximate solutions of the diffusion equation with black-body radiation boundary conditions.

Calibration work on pulse column experiments is complete. An analysis of the experiments themselves has been started.

The average concentrations of radionuclides of HAPO origin in the Pasco sanitary water was equivalent to $< 2\%$ of the MPC_W-GI . The trend continues downward.

The concentrations of fallout materials at various locations in the Pacific Northwest averaged $5.2 \mu\text{mc}/\text{m}^3$ during February. This is about 20% under that reported last month.

No new cases of plutonium deposition were confirmed by bioassay analyses during the month. The total number of plutonium deposition cases that have occurred at Hanford remains at 283, of which 204 are currently employed.

There are ten currently active projects having combined authorized funds amounting to \$2,653,600. The total estimated cost of the projects is \$7,467,000. Total expenditures on them through January 31, 1962 were \$789,000. One new project proposal for Reactivation of the H-1 Loop (H Reactor) has been submitted to the Commission.

SUPPORTING FUNCTIONS

PRTR output was 1013 MWD for a plant efficiency of 51.7%. Accumulated exposure to date is 5612 MWD. Additional exposure to date information is as follows:

Maximum UO ₂ fuel element	86.8 MWD or 1710 MWD/T
Average UO ₂ fuel element	57.8 MWD or 1130 MWD/T
Maximum Pu-Al fuel element	75.3 MWD or 45% burnup
Average Pu-Al fuel element	37.6 MWD or 25% burnup

The seventh refueling was accomplished on February 7, 1962. One Pu-Al test element was discharged at the planned exposure and retained in the basin for future PRP-CF measurements. Three new Pu-Al elements were charged on the outside of the Pu-Al region and the shim-free critical level was 68 inches. There were two minimum outages during the month in which three and two new Pu-Al elements were charged.

The following table shows the status of the test fuel elements at month's end:

PRTR				Date		
Test	Tube	Element		Date	Dis-	Accumulated
No.	Location	Number	Description	Charged	Charged	MWD
5	1653	1501	UO ₂ -Tubular	11-3-61		36.8
10	1051	1082	UO ₂ -Hot Swaged	11-3-61		20.5
10	1857	1067	UO ₂ -Vipac	11-3-61		22.9
13	1253	5092	Pu-Al Instrumented	12-3-61		32.3
13	1544	5093	Pu-Al Instrumented	12-3-61		28.5
13	1556	5095	Pu-Al Instrumented	1-17-62		24.1
13	1247	5051	Pu-Al Instrumented	1-17-62	2-7-62	13.1
13	1847	5052	Pu-Al Instrumented	1-17-62		21.9

The Plutonium Recycle Program Critical Facility is an estimated 98% complete. Control rods were installed and the drives tested satisfactorily. One safety rod assembly was installed. The vendor was unsuccessful in attempts to clad two additional rods. Performance of design tests was halted pending completion of additional work.

The Fuel Element Rupture Test Facility is an estimated 86% complete compared to 88% scheduled. Design and development started on the cooling assembly for discharge operations. Design was started for shielding of ex-reactor components. Design of the basin clean-up system is essentially complete. Water Plant work is approximately 67% complete. Concrete placement is approximately 95% complete.

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The Gas Cooled Loop Project is 91% complete based on a revised authorization received during the month. Progress on the primary blowers was negligible.

Total productive time of the Technical Shops Operation for the period was 21,094 hours. This includes 14,173 hours performed in the Technical Shops, 4,315 hours assigned to Minor Construction, and 2,606 hours assigned to off-site vendors. Total shop backlog is 18,930 hours, of which 60% is required in the current month with the remainder distributed over a three-month period. Requests for emergency service increased requiring a 4.2% overtime ratio compared to a 2.8% ratio for the previous period.

Holdings of SS material increased \$4.3 million during February bringing the total held to \$24.8 million. The increase consists primarily of plutonium for use on programs sponsored by the Division of Reactor Development.

Heavy water losses chargeable to PRTR operating cost during February amounted to \$38,000. Accumulated scrap for return to Savannah River is valued at \$218,000.

Recruitment - Six PhD applicants visited HAPO for employment interviews. Seven offers were extended; no acceptances and three rejections were received. Current open offers total four.

Ninety-six program offers and one direct placement offer were extended to B. S. /M. S. applicants; one program offer was rejected. Offers open total 102.

Technical Graduate Program - Ten Technical Graduates were placed on permanent assignment and one new member added to the rolls. Current program members total 54.

The annual inventory of Secret and Weapons Data research and development reports was completed. For 1961, all copies were accounted for.

for Manager
Hanford Laboratories

HM Parker:CAB:st

REACTOR AND FUELS RESEARCH AND DEVELOPMENT OPERATIONTECHNICAL ACTIVITIESA. FISSIONABLE MATERIALS - 2000 PROGRAM1. METALLURGY PROGRAMCorrosion Studies

Corrosion of Copper Bonded End Closures. Investigation of the corrosion characteristics of copper bonded end closures is continuing. To date the samples exposed to 360 C water are showing small gray lines along the edges of the welded areas which contain copper contamination. The samples exposed to 400 C steam appear to be normal and show no gray areas. Following further exposure, the samples will be sectioned to determine the oxide distribution on various areas of the ring closures.

Dual Autoclave Cycle Modification. A statistical test was devised to evaluate four variables of the chromate-dichromate dual autoclave cycle and thus maximize the corrosion resistance with regard to these variables. The variables under consideration are water autoclave time and temperature, and chromate-dichromate autoclave time and temperature (temperatures 170-290 C, times 3 to 27 hours). The autoclaving of the coupons is complete and the accelerated corrosion test is presently being performed.

Zirconium Oxide Removal by Cathodic Etching. The use of cathodic etching instead of vapor blasting to remove zirconium oxide films from autoclaved fuel elements appears promising. During etching some of the oxide dissolves into the metal in an amount proportional to the thickness of the original oxide. Subsequent chemical etching in an HNO_3 -HF solution is required to produce acceptable surfaces for re-autoclaving.

Erosion-Corrosion of Aluminum Alloys. Samples of aluminum alloy X-8001 have been exposed to 300 Area tap water containing various amounts of sodium dichromate at a flow velocity of 75 ft/sec and a temperature of 102 C. Dichromate concentration varied from 0 to 2 ppm. Exposure times were 24, 48, and 72 hours. Samples exposed for 24 hours at dichromate concentrations from 0.1 to 0.9 ppm showed a definite decrease in corrosion as dichromate increased. No corrosion was seen in any 24-hour tests at dichromate concentrations of 0.9 ppm and above. Samples exposed for 48 and 72 hours showed pitting at dichromate concentrations from 0.9 to 2.0 ppm. The pits occurred in the area where the sample covered the water intake hole, but not along the exposed groove where the water flowed. Tests are planned with alloy 1245 to compare the pitting tendency of the two alloys.

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Radiometallurgy Laboratory Studies

Two defected KER single tube fuel elements with 3700 MWD/T exposure were examined. One of the elements was severely broken while the other had sustained only a small rupture blister (RM 346). One of three NIE elements with enriched cores alloyed with iron and silicon and irradiated 2300 MWD/T was sectioned and end cap closure and braze were found to be in good condition (RM 593).

Metallic Fuel Development

Charging Force Studies for Present Reactors. Charging force testing is complete. In all tests involving self-supported fuel elements in ribless process tubes the average coefficient of friction was reduced from 16 to 40 percent by orienting the supports in the 45 degree planes rather than in the vertical plane. Tests to evaluate the influence of charging oil indicate that in all cases of similar metals (aluminum-aluminum) a ten percent solution of A-60 soluble oil in water decreases the coefficient of friction 13 to 50 percent. In the cases of water lubrication in the aluminum-aluminum combination galling was noted in all cases. Addition of soluble oil reduced the galling to incipient scratches. Tests involving dissimilar metals, aluminum-Zircaloy-2, nickel plate-aluminum, nickel plate-Zircaloy-2 indicate that water lubrication is superior. Coefficients of friction were 10 to 30 percent lower using water lubrication rather than a soluble oil solution.

Fuel Irradiations. Five NIE fuel elements (N-fuel inner component) have been discharged from KER Loop 1 after attaining an exposure of about 1100 MWD/T. These elements, charged as a warp test, displayed no evidence of serious post-irradiation warp. The initial 60 percent of the irradiation was in hot water and the final 40 percent was in once-through cold water. A very light reddish deposit had developed on the upstream elements, probably as a consequence of having operated in the once-through water. Fuel swelling determinations gave an average of 0.2 ± 0.1 volume percent increase for the five elements.

Post-irradiation evaluation of three NIE elements irradiated to 2300 MWD/T is in progress in the radiometallurgy facility. Post-irradiation warp of the 17-inch elements was slight, averaging only 0.003-inch single throw, with the plane of warp having shifted in a random fashion. No inadequacies have been observed in the Zr-Be brazed closures or the heat-affected zones of the fuel. There have been no indications of inner bore collapse from fuel swelling or of any unusual behavior of the fuel.

A thermocoupled fuel rod has been charged into KER Loop 1. This is of an improved design developed after previous thermocoupled fuel rods failed (cladding failures associated with two thermocouples located just under the cladding). The present probe has just one thermocouple located in the center of the fuel rod. Thus, the stress raisers which caused the cladding to split in the previous design have been eliminated. The specimen is operating in 252 C water with a core temperature of 615 C and a surface

heat flux slightly in excess of 600,000 Btu/sq-ft-hr. The core temperature and surface heat flux are close to the design values.

Irradiation of the first full-sized tube-in-tube N fuel element in the ETR continued until February 5, 1962, completing the third full cycle of irradiation. The average total exposure at the time of discharge was 700 MWD/T. This N fuel assembly will be visually examined and diameter and warp measurements will be made before it is recharged into the reactor for additional irradiation. The inner tube which was replaced because of broken supports at the end of the second cycle, has been returned to HAPO for examination.

The irradiation testing of a single tube, dual enriched fuel element, having the same fuel cross section as an N fuel assembly, was started in the M3 loop of the ETR. Initial temperature and flow measurements indicate that at full reactor power the element is operating at a peak specific power of approximately 180 kw/ft. With the loop controlling at an effluent temperature of 260 C, the maximum uranium temperature is calculated to be 570 C.

Periscope measurements and pictures have been completed on the production brazed irradiation test at the MTR, GEH-4-63 and 64, and the samples have been loaded in a cask for shipment to Hanford. No visual evidence of damage to the fuel was noted. Radiometallurgical examination should begin on these elements early in March, 1962.

The variable braze thickness test, GEH-4-68, 69, and 70, is now in its fourth cycle of a requested five cycles of irradiation at the MTR.

Heat Treatment Studies. Six NOE uranium coextrusion billets were beta heat treated as a final fabrication step before coextrusion. Pronounced improvement in inner clad thickness control was observed on subsequent coextrusion. An additional thirty NOE billets are being processed and will be completed by March 1. The treatment consists of heating to 730 C in chloride salt, holding 30 minutes, and quenching with 25 $\pm$ 2 seconds transfer time to a circulated one percent salt solution at 25 $\pm$ 6 C. Six NIE billets have been processed with the same treatment. These coextrusions are currently being evaluated by FPD.

Two 11-inch diameter, 19-inch long NOE castings of Fe-Si additive material were received for heat treatment, primary fabrication, and coextrusion. One ingot was given a 15-hour alpha soak at 540 C (1000 F) followed by a single beta heat treatment, consisting of one-hour, 15-minute immersion in 730 C chloride salt, 50-second transfer to 20 C circulated water quench. The primary extrusion was completed by Coextruded Products Engineering. The second ingot is being held, pending evaluation of the primary extrusion and coextrusion.

Fluted Fuel. Some measure of the ability of a fluted tubular fuel element to accommodate uranium swelling was obtained by expanding a fluted fuel element jacket hydraulically. Uranium was chemically removed from a six-inch section of coextruded fuel tube and Zircaloy-2 end caps were brazed

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into the two ends. The space between the inner and outer cladding was then pressurized with water. At 150, 350 and 650 psi pressure the core space volume increased 17, 31 and 37 percent, respectively. The cladding failed at the end cap at a volume expansion of 37 percent.

N Fuel Supports. Fatigue tests on NPR inner fuel tube supports were continued at stress levels sufficient to cause failure at several million cycles or less. The calculated point of maximum stress corresponds with the observed point of failure. On the basis of published data and the calculated stress conditions in the support fatigue tests, the supports should last 10^5 cycles. Some of the supports lasted 10^7 cycles whereas others failed at 10^4 cycles. Those that failed at 10^4 values probably had cracks in the bends that concentrated stresses. Thermal Hydraulics Operation is initiating a study to determine the extent to which an NPR inner fuel tube vibrates in a horizontal tube under NPR flow conditions. These tests will help determine if supports are likely to be subjected to fatigue conditions in the NPR.

Placement of support welding processes in the N fuel production plant by Fuels Fabrication Development is complete. The final report on this work is now in draft form. An FPD engineer has been assigned the future responsibilities for this project. It is estimated that one more week will be required to acquaint this man with the background and present state of the project. Of the two magnetic force welders installed in 333 Building, one is equipped for manual welding of supports while the other has tooling designed to attach outer or inner supports on a semi-automatic basis. There are some deficiencies in the tooling primarily from the aspect of durability. It is expected that the difficulties will be overcome by the second generation tooling. A "spike power" welder is installed in 333 Building with tooling for welding locking clips. Weld schedules have been furnished for various supports and clips on these respective machines. Coextruded Product Engineering has a Sciaky welder in 306 Building. This machine has tooling for inner and outer supports and can be adapted to weld locking clips. Operating instructions and machine settings have been furnished to enable CPE personnel to use this machine for pilot plant support attachment.

Extrusion Development. Four Zircaloy-2 blanks were successfully back-extruded into four-inch OD by 3-9/16-inch ID sleeves for use in special coextrusion billet components. These sleeves were extruded at different extrusion speeds using a modified core punch. Previous extrusions were made using a flat punch and a 120 degree included angle pointed punch. The two 120 degree angle punches which were tried cracked on the first extrusions; however, the extrusions were more concentric than those produced by the flat punch. A redesigned core punch did not crack during four extrusions, and produced even better concentricities in the resulting components. The extrusion speed was varied from 38 inches/min to 258 inches/min. No appreciable difference in concentricity or quality of the extrusion was evident over this speed range.

Closure and Joining. Development work on Projection Welded-Brazed Closures essentially stopped for the first two weeks in February due to mechanical failures in tooling, fabrication of a new wafer stack to accommodate standard production size fuel, and fabrication of form tools for end caps and fuel. The tightening nut for the wafer stack takes high thermal shock, moderate mechanical stress, and high thermal expansion stress during each weld. After three months of continuous use the original nut split longitudinally. The cause of the split was a combination of the above forces and a slight misfit between the nut and wafer stack.

Mating form tools (male and female) were developed for machining end caps and fuel. The tools provide accurate alignment of the parts during welding and reduces uranium movement in the joint. Current distribution within the joint is controlled by electrode configuration and several electrode shapes are being evaluated. A concave electrode with a four- to five-foot radius produces uniform heating rates between the ID and OD cladding and has significantly improved joint integrity. A torus shaped electrode has also improved current distribution, not as effectively, however, as the concave electrode.

Two promising approaches to the problem of producing consistently sound bonds at the closures on Zircaloy clad NPR fuel elements were further exploited: (1) A ceramic-lined steel die to enclose and support the test element between the electrodes of the heavy duty Sciaky spot welder was used to process two experimental fuel elements. With the adjustments used in the first trial, the heat generated in the end zones was not quite sufficient to effect complete bonding. In the second trial, the ceramic lining became damaged, but there was evidence that complete bonding could be obtained. (2) Two already-pressed fuel elements having defective closures were processed by placing them in the all-ceramic die with imbedded induction coil and without axial load, raising the closure zone to about 800 C, holding at this temperature for two minutes. There was sufficient differential expansion of the uranium core, versus that of the Zircaloy in both cases, that the end cap was slightly bulged. The contact pressure at the core/cap interface was enough at that temperature to cause complete bonding. However, the high strength alumina die liner proved incapable of withstanding the tremendous thermal gradient and a number of circumferential cracks developed in the liner. The results support the principle that the components may be bonded at relatively low load, if the temperature and conditions of cleanliness are right. Furthermore, it is indicated that the thermal expansion of the uranium may be employed under proper conditions to supply the interfacial pressure without the need of the press.

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To make use of this phenomena, the element must be restrained both radially and longitudinally while its assembled and pre-welded closure end is heated. The collet head was intended to provide such restraint with simultaneous heating. However, heat generation and transfer problems cause trouble in this approach.

N Closure Development. The present brazing practices on NPR fuels utilize braze rings of the Zr-5 w/o Be brazing alloy. This alloy is very brittle and difficult to fabricate; consequently, the braze rings are quite expensive. A possible method of reducing the cost of the brazing operation is to preplace pure beryllium on the Zircaloy end cap and form the braze by diffusion in a high temperature operation. Several methods of preplacement of the beryllium have been considered. Among these methods is the vapor deposition of the beryllium. A vapor deposition apparatus has been set up using an electron beam heating technique. Using this technique it is possible to plate up to several mils of beryllium in a relatively short time. Initial trials have resulted in beryllium films ranging from a few angstroms thick to one-half mil thick and from tightly adhering films to non-adherent films. The results, however, are very promising and it is felt that with refinement in technique, high quality beryllium films can be deposited consistently.

Fuel Deformation Studies. The four cladding studies capsules, charged in DR Reactor in January, are operated with Zr-2 fuel element cladding surface temperatures somewhat higher than desired. The cladding surface temperature in three of the capsules is 360 C and in the fourth capsule is 420 C. Burnup of the fuel in these capsules has reached 840 MWD/T with the goal exposure now set at 2000 MWD/T.

Fabrication of components for a more extensive irradiation of fuel rods with non-uniform thickness Zr-2 cladding is progressing rapidly. All the capsule parts and Zr-2 coextrusion components for the fuel rods have been machined and inspected. Primary extrusion of the fuel is complete and the coextrusion billets are now being machined.

An examination of burst test specimens which had been obtained by dissolution of the uranium from irradiated 0.593 coextruded rods revealed that the Zr-2 had been attacked in the uranium removal process. However, some of the specimens appeared to be suitable for testing, using an expanding mandrel, to establish the strain properties of irradiated cladding. An expanding mandrel is being fabricated which will develop enough circumferential stress to fail the Zr-2 sheath.

To investigate the relative irradiation performance of uniformly enriched and "dual" enriched uranium fuel elements operating at equivalent power generations and surface temperatures, three capsule type experiments are being prepared for irradiation in the MTR.

The formulation and analysis of mechanical models for reactor fuel elements require an understanding of the mechanical behavior of the fuel and cladding materials under the conditions encountered during reactor operation. Incremental strain increases during strain cycling have been studied using Zircaloy-2 specimens at 270 C. For a one percent cyclic strain, a 0.1 percent strain increase per cycle was obtained at the same stress level (50,000 psi) as for the standard reverse strain-cycled test.

An in-reactor strain-cycling capsule for uranium has been fabricated and the instrumentation for this in-reactor test was designed.

A model for fuel element swelling was formulated which accounts for the temperature and burnup distributions. The parameters in the model are to be determined using data obtained from experimental fuel element irradiations. The model was formulated so that the parameters could be adjusted for a least square error fit with an existing program called "NELLY".

Non-Destructive Test Development. No non-destructive testing method currently exists which will differentiate between uranium cores which are thermally stable and those which will rapidly warp under a thermal gradient as a result of very "eccentric anisotropic" grains or residual stress. A device is now under test which will subject cores to a high amperage current while simultaneously observing the position of the longitudinal axis at one end of the cores with the other end being fixed in place. This device uses the Reslec heater as a source of current. In its initial form the sensitivity of the system was demonstrated by heating a core to 140 C and observing the displacement caused by lightly touching a damp cloth to one side of the core. This displacement was readily measurable. Some additional development is required to insure an adequate supply of current to the cores without localized heating and to calibrate the optical system.

Die Lubrication. The use of standard drawing lubricants in the drawing of Zircaloy wire has consistently resulted in lubricant breakdown and poor quality wire products. In the search for improved lubricants, samples of five asphaltum products were obtained, dissolved in xylene, and applied as a solution to the wire to be drawn. The lubricant was allowed to air dry rather than being baked

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on the surface. Of the five samples tested, Flintkote appears to give the best performance. The coating is adherent and does not break down during the drawing process. In these tests, the wire was reduced to 20 percent in a single pass at a rate of 5.8 feet per minute. The performance of the dissolved Flintkote lubricant was markedly superior to that of the standard drawing lubricants. Failure of the Flintkote lubricant has occurred only when the coating was unduly thin and initial wire surface quite rough. In the latter case the roughness was caused by seams and folds produced during previous forming operations.

2. REACTOR PROGRAM

Corrosion and Coolant Systems Development

Hydriding of Zircaloy-2. Zircaloy-2 coupons were exposed over a range of temperatures to helium - 1% hydrogen that had been purified of oxidants (O_2 , H_2O) by passing the gas over calcium metal at 650 C. Sample temperatures ranged from 400 C to 100 C in 50 C increments. Three freshly etched samples and three samples pre-autoclaved 72 hours in 400 C, 1500 psi steam were run at each temperature. All of the samples at 200 C and above failed by accelerated hydriding within seven days. The preautoclaving offered only transitory protection. Exposure of samples at 150 C and 100 C is continuing. These results confirm the oxidant control mechanism of zirconium hydriding and indicate measures should be taken to assure an adequate oxidant level whenever zirconium and hydrogen are exposed together. The data explain some previous "anomalous" low temperature hydriding failures of zirconium and indicate difficulties reported in the literature in hydriding zirconium below 400 C result from insufficient control of the oxidant level of the gases used in the experiment.

The electrical resistance of the ZrO_2 film on Zr-2 remains high, above 100 Kohms, when oxidant is present in the gas atmosphere and drops to a value below 100 ohms in the absence of oxidant. Experiments have shown that the permeability of the oxides to hydrogen may be related to this change in electrical resistance. In an experiment using etched samples the time that accelerated hydriding would begin was predicted successfully three days in advance.

Rupture Testing of Fuel Elements. Tests are being continued in ELMO-4 (TF-9) to confirm earlier results that if an unirradiated fuel element were given a pre-treatment at 850 C, the rupture rate would be about the same as for an irradiated fuel element. Two fuel elements were heat-treated at 850 C for eight hours, defected and tested in TF-9. The rupture rates of both elements were similar to

rates for pre-defected fuel elements irradiated to 1000 MWD/T. However, one difference in rupture behavior was noted. The unirradiated elements failed only by the corrosion process with the core still in one piece; whereas the irradiated core broke into pieces. It does appear, in spite of the difference in cracking of the core, that the testing of unirradiated pieces will give valuable information in fuel element evaluation.

One rupture test was performed during the month to measure the amount of flow blockage during a rupture. The element used was a NIN element (with a 1.25-inch OD) irradiated to 2100 MWD/T and with 25-mil defect on the outside. The element was placed in a 1.5-inch ID test section and the inner annulus of the fuel element was plugged. Test conditions were 300 C, 1650 psi, and 14 fps with a pressure drop of 95 inches of water across the element. The element was tested for 117 minutes after the element had started rupturing with 72 minutes at 300 C. At no time during the test was an increase in pressure drop noted. The element ruptured very severely as indicated by the filter activity, which increased to over 500 R/hr. This information confirms that obtained from previous similar experiments. Ruptures in dynamic systems, even when extensive damage occurs, may not block the passage of coolant to any appreciable extent.

Additional information was obtained to show that rupture damage is much more severe if the element is cooled slowly than if the reactor is scrammed. In one test, after the rupture indication the loop was held at 300 C for five minutes and cooled rapidly (scram), the rupture was mild with loss of 33 grams of uranium; in another test, after the rupture indication the loop was held at 300 C for six minutes and cooled at the normal NPR cool-down rate, the rupture was severe with extensive tearing of the Zircaloy cladding and a loss of 232 grams of uranium.

Inhibition of Stellite Corrosion. One of the major corrosion problems in decontamination is the attack of Stellite by the alkaline permanganate. Previously, some inhibition of the attack was obtained by addition of phenylthiourea to the alkaline permanganate. It was thought that the inhibition was due primarily to reduction of Mn^{+7} to Mn^{+6} , and that other reductants might be more effective. During the past month, two materials, HCHO and Na_2SO_3 , were tested as inhibitors. Both compounds were efficient reductants, reducing the Mn^{+7} to Mn^{+6} and providing stable solutions for the six 2-hour cycles. However, the corrosion of the Stellite was just as serious as without the inhibitors, indicating that more than reduction of Mn^{+7} to Mn^{+6} is required to reduce the attack on the Stellite.

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NPR Descaling Studies. Two mixtures were used on heat exchanger tube rings coated with atmospheric rust. The first solution, inhibited five percent sulfuric acid at 90 C, gave almost complete rust removal in 10 minutes. The second solution, inhibited one percent citric acid at 90 C, gave complete removal in 10 to 20 minutes. Large quantities of a dark brown adherent material were liberated from the inhibitor, Rodine 115, and coated portions of the beaker and test piece holder. Exposure to the air set this material into a hard mass. Neither acid descaling solution caused gas evolution at the metal surface.

Atmospheric Corrosion Tests. Vapor phase inhibitors are being considered for use in protecting the NPR piping systems during fabrication. Present plans call for the VPI (dicyclohexylammonium nitrite or similar compound) to be distributed through the piping system as a solid as fabrication progresses. The same plans call for the VPI to be removed shortly before startup with water flushing. Dissolution studies have been made on VPI using water. Over 90 percent of the VPI is readily dissolved. The residue appears to "ball" and in this state is rather insoluble. Further water treatments failed to dissolve any appreciable amount of this residue.

Temperature Drop Across Films. The new thermocouple train, including a coextruded Zr-2 clad rod with one thermocouple in the center, was successfully pressure tested and the thermocouple checked out. It was charged in KER-1 February 16. Prior to charging, a dummy train and the charge of heater elements (four 17-inch NIE-1) were charged and discharged to ensure the thermocouple train could be successfully discharged. During this trial discharge the supports on the downstream spacers (not part of the thermocouple train) showed some tendency to hang up on the rear nozzle, but not seriously. The two thermocouples measuring the core and water temperatures appear to be working well.

Corrosion of Materials in Columbia River Water. Tests in TF-13 to determine the uniform corrosion rates of 304 stainless steel, 70-30 Cu-Ni alloy, and A202 carbon steel in 210 F, raw Columbia River Water are continuing. Testing of Admiralty metal stressed to yield has been completed after 1150 hours of exposure. No stress-corrosion cracking was found. The uniform corrosion rate was 8 mils/year. The 304 stainless steel continues to corrode at a rate of 0.01 mil/year after 4000 hours of exposure. The carbon steel corrosion rate has dropped to approximately 3 mils/year after 5500 hours of exposure. The carbon steel surfaces are covered with 10-30 mil thick layer of oxide and tubercles, beneath which

are broad shallow pits. These pits appear to be slowing down in their growth.

Corrosion of Materials in NPR Secondary Water. The test in TF-8 to determine the corrosion rates of NPR secondary system materials of construction in simulated NPR secondary system boiler water has been terminated after 2764 hours of exposure. Test conditions were deionized water at 160 C containing 30-50 ppb hydrazine and sufficient NH_4OH to raise the pH to 9.

After 2764 hours of exposure the corrosion rates of 304 S/S, Admiralty metal, 70-30 Cu-Ni, silicon bronze, and A212 C/S were 0.015, 0.11, 0.05, 0.153, and 0.08 mil/year, respectively. No cracking was found on stressed samples of 70-30 Cu-Ni, Admiralty or 304 S/S. Deposition of a nonadherent, black oxide from the carbon steel loop piping was a problem during the test.

Effluent Activity and Corrosion Testing. Aluminum, carbon steel, and Zircaloy-2 samples were exposed in 105 KE facilities SP-7 and SP-8 for 89 operating days. The process water used to cool SP-7 contained small quantities of EDTA for the first 22 operating days, then phosphate ion for the next 18 days, and citrate ion for the final 49 days. Both SP-7 and SP-8 facilities were decontaminated three times with Turco-4306-B (after 0, 22, and 40 operating days). Facility SP-8 was cooled with process water (6.6 pH with 1.8 ppm corrosion inhibitor) during the entire operating period.

Corrosion of the three metals was not affected by addition of the EDTA, phosphate and citrate chemicals. The repeated decontamination with 4306-B, however, greatly increased corrosion of these materials. Total penetration of decontaminated samples averaged over 13 mils for aluminum, nearly 12 mils for Zircaloy-2, and one mil for carbon steel. Total penetration of control samples, whether exposed in the chemically treated process water or in the unmodified process water, averaged 1.4 mils for aluminum, less than 0.001 mil for Zircaloy-2, and 0.4 mil for carbon steel.

Structural Materials Development

Pre- and Post-Irradiation Evaluation of Zircaloy-2 Pressure Tubes. Creep rupture tests at 300 C are in progress on sections of an NPR Zr-2 pressure tube having 18 percent cold work. The table below summarizes the results to date:

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<u>Percent of Ultimate Burst Stress</u>	<u>Hoop Stress Psi</u>	<u>Time (Hrs)</u>	<u>Total Strain Percent</u>
90	53,000	425	0.2
93	55,000	550	0.5
97	57,000	370	0.36
99	58,500	18	0.58

The specimen operating at 99 percent of ultimate strength failed after 50 hours with a total elongation of four percent. This portion of the test is invalid, however, since an apparent failure of the pressure control system caused a sudden rise in pressure. The other three tests are continuing and these specimens appear to be in the early stage of secondary creep. A tentative comparison with tests on rolled strip material at comparable deformations indicate that similar creep rates are attained at stresses 10,000 to 15,000 psi lower than for tubing at this level of cold work.

An extensometer to monitor tube expansion is under development. A section of roller chain is wrapped around the tube and the relative motion of the ends of the chain is sensed by a high temperature, linearly variable differential transformer. The signal from this device is brought out to a measuring instrument at the control panel. Calibration curves have been prepared for room temperature, 150 C, and 300 C.

The design criteria for the facility for elevated temperature burst testing of irradiated pressure tubes has been approved. The design work by Bovay Engineers is estimated to be 25 percent complete. A burst test was performed in the prototype containment vessel. The test specimen was a section of three-inch Schedule 80 carbon steel pipe. Though the pipe failed at an internal pressure of 13,040 psi at 363 C, there was complete containment of the pressure wave caused by the expanding water. Techniques to correct the temperature variance between the top and the bottom of the furnace are being investigated.

Nonmetallic Materials

Reactor Moderator Oxidation. Burnout-monitoring samples in Channel 3478 since August 8, 1961, were discharged recently from D Reactor. The results indicate that conditions were favorable for high graphite oxidation. The burnout-rate profile showed a sharp peak between 90 and 110 inches from the front face of the graphite stack. These results are similar to those in the previous test in this same facility (see "D Reactor Graphite Burnout Interim Report," HW-71751).

A laboratory experiment in which gas flowed at a linear velocity intermediate to that of present Hanford reactors and the NPR over eight 2-inch graphite samples placed end-to-end showed peak oxidation occurred at the second sample whose average temperature was 755 C. To test the effect of flow velocity in-reactor on the peaking of graphite oxidation rates, sample geometries for the next set of three-channel burnout monitors at 105-C have been designed to achieve approximately a four-fold variation in the average linear velocity through the first third of the burnout channel.

NPR Graphite Irradiations. The long-term irradiations of NPR graphite continue in the General Electric Test Reactor. Graphites being tested are NPR samples cut transverse to the extrusion axis of the bar, NPR parallel samples, TSGBF transverse samples, and reference samples of CSF.

The first capsule in the series, H-4-1, is operating satisfactorily for the third cycle. Eight of the nine thermocouples continue to operate properly. Sample temperatures vary from 325 C to 750 C along the capsule length.

The second capsule in the series, H-5-1, is operating satisfactorily in the second cycle. All nine thermocouples are functioning correctly. Sample temperatures vary from 325 C to 750 C along the capsule length and agree within 15 C with the corresponding position in the H-4-1 capsule.

An inspection of the H-5-1 capsule was made during the last shutdown to determine the effectiveness of the capsule modifications made prior to installation. These modifications were made when a wear pattern was observed on a prototype dummy capsule. The stainless steel wearing surfaces showed a very slight amount of wear. No wear was visible on the capsule body itself and no evidence of fatigue of the capsule or leadout system was observed. No further wear was observed on the prototype capsule, and a pressure test confirmed its integrity.

Radiation, Heat, and Water Resistance of Elastomers. Chlorosulfonated polyethylenes are generally considered to possess a good balance of resistance to many deleterious environments. At Hanford, radiation, heat, and hot water are three of these environments which elastomeric materials must successfully resist in service applications.

An exploratory program has been completed which has shown that whereas chlorosulfonated polyethylenes can be formulated to resist any two of the three environments, it is difficult to produce materials with

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satisfactory resistance to all three. It has been found that the cure system is of prime importance with respect to environmental resistance. Other compounding ingredients, although not as important as the cure system, nevertheless can profoundly affect the utility of these materials. To date, several compounds have been prepared which do possess a good balance of resistance to the three environments.

Radiometallurgy Laboratory Studies

NaK capsule specimens GEH-14-94, 95 and 99 were found to be badly swollen and split open. After being cleaned of residual NaK, specimen 94-1 burned up approximately five days after being opened and specimen 99-2 burned up approximately two weeks after being opened. (RM 565)

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

Thermal Hydraulic Studies

Experiments Concerning Fuel Temperatures Resulting From an Inlet Pipe Rupture on a C Reactor Process Tube. Preparations were completed in the laboratory to investigate heat transfer conditions and fuel temperatures which would result from the coolant flow loss following a failure of a hydraulic fitting at the inlet of a process tube at C Reactor. The purpose of these experiments is to define the operating conditions at which the flow of coolant from the rear header following such an event would not be sufficient to prevent melting of the fuel elements.

A test section was installed which is a full scale electrically heated mockup of I&E fuel elements in a C process tube. A 23.6-foot long metal tube, 1.464 inches OD and 0.375 inch ID, represents the fuel elements and is made of materials with different electrical resistivities along the length to simulate a chopped cosine heat generation. This portion of the test section was placed in a standard process tube and connected in the heat transfer apparatus such that the normal flow of coolant could be interrupted suddenly while the inlet line is opened to the atmosphere.

The initial runs with the test section were devoted to establishing a power shutdown rate for the silicon rectifier power supply that would represent a 1400 inhour scram at the reactor. This required minor adjustments of amplidyne control circuits.

Film Boiling Experiments. Laboratory experiments were run in an attempt to obtain heat transfer data during film boiling of water under forced convection cooling conditions. The test sections used were composed of a cermet, molybdenum disilicide (trade name, "Kantal"). This material was selected since maximum temperatures experienced when operating in a water vapor atmosphere approaches 1700 C and would permit operation in the film boiling region where the heat transfer coefficients are low.

The first experiment was run at a water temperature of 235 F and a pressure of 10 psig. The heat generation was slowly increased and at a heat flux of 850,000 B/hr-ft² film boiling conditions were established as evidenced by a surface temperature of 1500 F. At this point a momentary flow reduction occurred during adjustment of the outlet pressure and the test section failed. The failure appeared to be caused by thermal shock rather than melting. The next test section failed at a heat flux of 300,000 B/hr ft² for no apparent reason when the surface temperature was only 300 F.

Molybdenum disilicide is very brittle and has little mechanical strength at low temperature. Since great difficulties were encountered in making test sections, it was decided to change to a different material for further experiments

Experiment for NPR Heat Transfer Experiments Acceptance tests were started on the equipment installed under Project CGH-834 to allow NPR transient experiments to be performed on the 189-D high pressure heat transfer apparatus. The equipment under test included a pressurized water storage tank, an air reservoir to pressurize the water tank, and a set of quick acting valves to isolate the test section from the pump and allow expulsion of the coolant from the test section. During the initial tests involving valve actuation, pressure readings were taken throughout the apparatus with fast response pressure transducers. It was found that pressure pulses of a very short duration occurred and were of a higher intensity than desired. It was thought that a small surge chamber would alleviate this condition, but the tests were suspended due to a leaking valve before further testing could be tried.

Shielding Studies

A shielding survey was made for Process Design in IPD as part of a conceptual study of a superheat power reactor. Five types of thermal shielding were analyzed: ordinary water, borated carbon steel, cast iron, a mixture of magnetite and water, and borated graphite. Gamma dose rates emanating from the ordinary concrete biological

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shield (102 inches from the core) were the same within a factor of three for all types. A wider range existed between neutron doses, 9 to 263 mrem/hr; however, the gamma leakage of 4 to 11 r/hr predominated in each case. In the current study, it was concluded that the use of borated and cast iron types of thermal shielding could not be justified economically, although they possess excellent shielding properties. The magnetite-water mixture was considered the best choice for a thermal shield because of its relatively low cost, excellent heat transfer properties, and the fact that it eliminates the need for cooling pipes in the concrete shield. The light water (which could serve as the reflector also) was ruled out in the present study on the bases of decreased neutron economy and increased fuel enrichment. A report describing the results of this investigation is being written.

The shielding properties of iron serpentine concrete (265 lb/ft³) were calculated using the multigroup attenuation code (MAC). These calculations agreed with experimental results within a factor of four for thermal neutrons and a factor of two for gammas at a distance of 231 cm from the core.

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 Hydraulics Studies

PRTR Flow Characteristics. An analysis was made of flow data obtained at the PRTR during a startup when the temperature of the primary coolant was increasing. The following observations were made:

- (1) The decrease in viscosity of the D₂O as it heats up from 130 F to 460 F causes a two percent increase in the volumetric flow rate.
- (2) The decrease in density of the D₂O upon heating from 130 F to 460 F causes a decrease in the bulk and average tube reading of about eight percent.
- (3) The decrease in individual tube readings upon the temperature increase shows considerable scatter. This probably reflects the difficulty in reading accurately the fluctuating flow meters.
- (4) The bulk flow reading is about two percent less than the sum of the tube flows. (Earlier data indicated that the bulk flow was four percent less than the summation of the tube flows. Since it was believed that the summation of the tube flows was more accurate than the bulk flow reading, the practice at the reactor is to correct the bulk flow reading by the four percent factor.)
- (5) There is about a 20% spread between the minimum and maximum flow tube.
- (6) The tubes in the southeast quadrant of the reactor have on the average about six or eight percent greater flow than do the other tubes. Earlier data indicate that this may change as different primary pump pairs are used.

Studies Concerning the Loss of Pumping Power in the PRTR. The analysis of the heat transfer conditions that would exist in the PRTR following a loss of power to the primary pumps was continued. Early studies had shown that circulation of liquid primary coolant by free convection could adequately transfer heat from the reactor to the steam generator as long as an adequate heat sink is maintained in the steam generator. However, certain events can be postulated which would allow formation of vapor voids in the primary system piping, reducing the convective

head and allowing boiling of the coolant. In such a case liquid circulation would be lost.

Thermal Hydraulics Laboratory experiments have shown that heat removal by boiling water in the process tubes will prevent excessive fuel temperatures, even following a reactor scram from full power operation, as long as the fuel remains covered with water. Calculations have also shown that the steam generator could condense steam as fast as it was being formed in the reactor and that free convection with boiling and condensation would be possible.

However, boiling could have disadvantages. In order for steam to reach the steam generator tubes where it could condense, it would be necessary to form a void of some 80 to 100 cu ft in the upper piping of the primary system. In some cases this void would be accommodated by flowing primary coolant through the pressurizer relief valves into "A" cell. Recovery would be difficult and expensive. Further, this loss of coolant combined with volume loss due to reducing the average primary coolant temperature, or due to leakage from the system could require addition of light water to the primary system to keep the fuel adequately covered.

It was concluded that free convection of liquid should be used wherever possible, and boiling utilized only if liquid circulation were lost accidentally.

The present instrumentation could provide indications that a void was forming in the primary system, but in many cases would not show whether the void was sufficient to initiate boiling. If boiling does begin, it would be necessary to make certain valving changes so that the pressurizer liquid level would indicate when water must be added to the system to prevent uncovering of fuel. These valving changes would allow gas to escape from the pressurizer to the upper piping, increasing the void and, perhaps, prematurely force the system into boiling.

Installation of an instrument to provide an accurate indication of the liquid level in the process tubes was recommended. Such an instrument would eliminate the necessity of valving to use the pressurizer level as an index, and the possibility of driving the system into boiling unnecessarily. It would also simplify emergency procedures and eliminate the reliance on the proper positioning of valves to provide a process tube liquid level indication. Early completion of design and installation of this instrument was recommended.

Component Testing and Equipment DevelopmentPRTR Components.1. Liquid Poison Shim Rods

Laboratory tests continued to determine means of inhibiting cadmium plating and/or precipitation in the proposed liquid poison shim rods. Several additives were mixed with cadmium sulfate solutions. Additives of chromate, nitrate, and nitrite appear to have inhibited the plating and precipitating reactions in the cadmium sulfate-aluminum system.

2. Mechanical Shim Rods

Design of a second generation mechanical shim rod continued. One problem with the present shim rod arises from flaking and corrosion of the lead screws. It is believed that this results from the rolling process used in threading the screws; therefore, the new design will utilize machined or ground threads. Lead screw and ball nut assemblies with ground threads have been ordered to verify the premise.

3. Environment Control Test Assembly

The design of the environment control facility for testing second generation shim rods under reactor conditions (except nuclear) is essentially complete and material procurement has been initiated.

The design utilizes a methyl methacrylate housing tube to permit visual inspection of shim rods in operation. Although the strength of the tube is adequate at the operating conditions, the strength of the joints at the side outlets (for controlling water levels) is unknown. Hydrostatic tests of different joint designs are in progress to establish their strength characteristics.

4. Reactor Auxiliaries Test Facility

Scope design of a facility to test compressors, blowers, and other reactor gas-type auxiliaries was initiated.

Critical Facility Components.1. Fuel Element Transfer Thimble Assembly

The PR Critical Facility fuel element transfer thimble assembly was completed. It appears likely that the physical layout and dimensional limitations of the thimble assembly may preclude remote handling with the existing PR Critical Facility equipment. Reduction in the size of the flexible cooling lines appears to provide a possible solution.

2. Safety Rods

One safety rod has been delivered to the PR Critical Facility. The remaining two safety rods are awaiting the cadmium plated safety rod tubes. In spite of continued attempts, the vendor has not been able to obtain a satisfactory plating with the remaining tubes. New aluminum tubes are being prepared with a machined surface in an attempt to limit surface porosity which the vendor contends has been the reason for his failure to produce a satisfactory plate.

3. Adjustable Weir

An adjustable mechanical stop was designed, fabricated, and installed in the PR Critical Facility adjustable weir.

Rupture Loop Components.1. In-Reactor Test Section

The in-reactor test section has been installed in the EDEL-I 35-foot test pit. Leak collection traps are being fabricated and installed at the various gasket seals of the test section. These traps will be used to determine the leak rates of the seals.

2. Discharge Procedures and Equipment

Discharge procedures for the rupture loop require that cooling water be supplied to the test section during the period when the temporary cooling hoses are being attached, immediately prior to discharge. Tests are being performed to determine if water injection at or above the top of the fuel element will prevent ejection or oscillation of the water column above the

fuel element. The preliminary data indicate that the water surrounding the fuel element must be kept below boiling if oscillation of the water column is to be prevented. If ejection of the water column is to be prevented, the data indicate that the entire water column must be kept below boiling.

Special connectors have been designed for the nozzle outlet and the tube inlet. The nozzle cap has a three-way valve which incorporates an inlet and an outlet water connection. In this way cooling water can flow out of the tube while the discharge hose connection is made.

Hazards Analysis

Process Specifications. Five revisions to the PRTR Process Specifications were initiated during the month.

Organization and initial drafting of Process Specifications for the Plutonium Recycle Critical Facility have been started.

Plutonium Recycle Critical Facility. Studies concerning reactivity addition and control of excess reactivity were incorporated in the final safeguards analysis for the Plutonium Recycle Critical Facility.

PRTR Fuel Element Rupture Loop. Calculations of transient conditions in the Fuel Element Rupture Loop following occurrence of credible failures have been initiated for the final safeguard analysis.

PRTR Technical Assistance

The final report on PRTR critical test results, HW-61900 BA, was completed.

Several neutron flux distribution calculations were completed for fuel element test irradiations to be carried out in the MTR and ETR. Similar calculations were also performed for fuel elements to be tested in the PRTR rupture loop. These results are being utilized to obtain relative heat generation rates, from which fuel element temperatures may be computed.

Plutonium Fuels Development

PRTR Aluminum-Plutonium Fuel Elements. The fabrication of spike fuel elements containing low exposure plutonium has been completed. A

total of 67 Mark I-H, 19-rod cluster fuel elements have been fabricated for the PRTR. A Mark I-H fuel element rod consists of an Al-2.0 w/o Ni-1.8 Pu alloy core clad in low-nickel, Zircaloy-2 tubing which has a wall thickness of 35 mils. Included in the total were six special flux-monitoring elements, made to measure radial and axial flux patterns within a cluster. Of the 59 elements irradiated in the PRTR, 26 have greater than 60 percent of goal exposure and eight have over 80 percent of goal exposure. Goal exposure is 50 percent burnup of plutonium atoms.

The first fuel element rods for three aluminum-plutonium clusters containing high exposure (HX) plutonium 16.5 percent Pu-240) have progressed up to autoclaving. Completion date for reactor delivery is early March, 1962. These HX, Mark I-H, 19-rod cluster fuel elements consist of rods with alloy cores of Al - 2.65 w/o Pu - 2 w/o Ni clad with 35-mil wall, low-nickel, Zircaloy-2 tubing. Forty-eight billets containing the aluminum-HX plutonium alloy were extruded into 96 PRTR spike fuel cores for the fabrication of the rods. The extrusion process was similar to that used for the low exposure core material. These cores were drawn to diameter, straightened, and cut to length prior to loading. Corrosion test pieces cut from front, middle, and rear of each extrusion indicated good corrosion resistance, similar to the low exposure cores containing 1.8 w/o Pu.

Final assembly and welding of 25 sets of Mark I-H end brackets were completed. The fuel rod lifting mechanism was installed in the welding chamber and is operable. Lead rubber curtains, to cut personnel exposure in front of hood gloves, and lead glass have been installed on hoods designated for use during fuel element fabrication with high exposure plutonium.

Irradiated Fuel Element Inspection. Four aluminum-plutonium spike fuel elements were inspected in the PRTR storage basin during the February 5 outage. Two of the inspected elements (5086 and 5087) were charged in September, 1961, as experiments to eliminate rod shortening which in turn causes wire loosening. Graphite dag was used on the cores of 5087 and a minimum cold diametral gap between core and cladding of five mils instead of the normal four mils was used on 5086. The two have been examined periodically and in general, 5086 (with the larger gap) seems to be operating much better than the standard fuel elements and 5087 (with the dagged cores) about the same as the standard fuel elements. Both have been operated to about 45 percent of goal exposure. The other two elements, 5084 and 5085, are standard elements and have been operated

to about 30 percent of goal exposure. Each shows rod shortening of 1/16 inch. All have loose wires and have gaps between wire and rod of from 1/32 inch to 1/8 inch.

Phoenix Experiment. The irradiation and reactivity measurements on the high exposure aluminum-plutonium samples are continuing in the MTR and ARMF. Because the ARMF was shut down, reactivity measurements have lagged on some of the samples. At the present time, the aluminum-plutonium sample with plutonium initially containing 6.25 w/o Pu-240 (GEH-21-1) has received three cycles of MTR irradiation, the sample which contained 16.33 w/o Pu-240 (GEH-21-3) has received two and is being irradiated for its third cycle, and the two samples containing plutonium with 27.17 w/o Pu-240 (GEH-21-9 and 21-19) have received two cycles and one cycle of irradiation, respectively.

Sample number GEH-21-9 was inadvertently cut during manipulation in the MTR canal and was replaced with GEH-21-19. Pre-irradiation reactivity measurements on GEH-21-19 were practically identical with the pre-irradiation measurements on GEH-21-9. This indicates very good plutonium homogeneity in the extruded samples. GEH-21-9 is being returned to Hanford for radiometallurgical examination to determine the effects of irradiation at high heat fluxes. Also, isotopic analysis of the plutonium will be made and this information will be correlated with the exposure to determine plutonium cross sections.

The Phoenix irradiation program will be extended and the possibility of irradiating plutonium-containing samples which yield a flatter reactivity response is being investigated. This may be accomplished by tailoring the neutron spectrum with absorbers to more effectively utilize the resonance capture of Pu-240 and more nearly demonstrate true Phoenix action. Other possibilities for samples with flatter reactivity response are aluminum-uranium-plutonium alloys and samples based on the self-shielded fuel concept.

An underwater work platform and handling tools are being designed to better facilitate underwater work on the Phoenix specimens in the MTR canal.

Irradiation Testing. The first cycle of irradiation for the aluminum-plutonium injection cast seven-rod cluster (GEH-11-6) has been successfully completed in the ETR 3x3 loop. This element was fabricated by injection casting an Al - 2.53 w/o Pu - 2.0 w/o Ni alloy core directly into the Zircaloy cladding. It operated for 12.5 effective full power days during this first cycle at a maximum heat flux of

463,000 Btu/hr-ft² and a maximum calculated core temperature of 453 C. The element will be irradiated for another cycle before it is returned to Hanford for examination.

The UO₂-PuO₂, cosine-enriched, seven-rod cluster element (GEH-11-8) is operating successfully in the ETR 3x3 high temperature loop facility. This element was fabricated by cold swage compacting arc-fused UO₂ enriched with -325 mesh PuO₂. The PuO₂ enrichment was varied along the length of the element in such a way that the longitudinal power generation would be nearly constant. Swageable end caps were used and hence, there was no end gap between the fuel core and the end caps. The element commenced operation on February 18, 1962, and is currently generating about 210 kw. This condition gives an effective vertical peak/average thermal neutron flux ratio of 1.12 as compared to 1.4 for the uniformly enriched case (GEH-11-7). Therefore, this element is generating about 28 percent more power than the uniformly enriched element for the same maximum heat flux of 345,000 Btu/hr-ft² and central core temperature of 1850 C. Irradiation is scheduled to end on March 19, 1962.

The Zircaloy-clad capsule (GEH-14-27) containing an Al - 2.1 w/o Pu - 2.0 w/o Ni alloy core and fabricated by injection casting was discharged from the MTR at the end of cycle 168 (January 29, 1962). The specimen received an exposure of 12.6×10^{20} nvt or 82.5×10^{18} fissions/cu cm. Goal exposure was 12.5×10^{20} nvt. The burnup of plutonium atoms was 60 percent. The specimen operated at 25 kw/ft and with a surface heat flux of 588,000 Btu/hr-ft². The capsule is awaiting shipment back to HAPO for examination.

The four specimens (GEH-21-13, 14, 15, and 16), each containing UO₂ - 0.154 a/o PuO₂ and UO₂ (1.00 a/o U-235) pellets, for irradiation in the VH-4 Hydraulic Rabbit Facility at the MTR have not been charged. The irradiation testing has been delayed because the facility is inoperable until piping and wiring modifications and installation of a backup safety circuit are completed.

The eleven discharged high and low density UO₂-PuO₂ capsules are currently being examined. Results to date are shown in Table I. Of particular interest is the presence of a sphere, 0.1-inch in diameter, on the main axis and 1/8-inch in from one end of the UO₂ - 1.02 a/o PuO₂ core of capsule GEH-14-66. The UO₂-PuO₂ density was originally 65 percent of theoretical. The sphere is very hard and exhibits excellent oxidation resistance. cursory examination of the sphere indicates that the sphere may be an alloy or compound of uranium-plutonium or uranium-plutonium-zirconium plus fission fragments. An autoradiograph of the sample shows the sphere to be very

TABLE I
EXAMINATION OF IRRADIATED UO₂-PuO₂ CAPSULES

<u>GEH-14 No.</u>	<u>UO₂-PuO₂; (a/o PuO₂)</u>	<u>Density (% of Theo.)</u>	<u>Exposure (10²⁰ nvt)</u>	<u>Remarks</u>
22	0.0259	65	28.40	Central Void formed.
66	0.187	65	21.83	Metallic sphere (0.1 inch OD) and central void formed.
68	1.46	65	7.39	Being examined.
69	3.47	64	3.45	" "
71	5.46	63	2.15	" "
73	7.45	63	1.21	" "
20	0.0259	90	18.73	No central void formed.
91	0.0259	90	31.80	Central void formed.
82	1.02	93	11.62	" " " (Burnout analysis - 1.57 a/o)
85	2.57	91	4.79	To be recharged.
86	4.13	91	3.26	" " "

markedly radioactive in comparison to all surrounding fuel material. Investigation of the specimen is continuing.

Except for the bellows, all components for the ICARUS Experiment were completed. Flow tests will be initiated as soon as the bellows are received from the vendor.

Fabrication of parts for the HELIOS Experiment was initiated. The UO_2 - 5 w/o PuO_2 pellets for the assembly for reactivity tests were completed. Seven UO_2 - 5 w/o PuO_2 samples were pressed and a central hole drilled in each green pellet. These pieces were sintered at 1500 C in hydrogen and ground to finish dimensions. Feasibility of fabrication of required thermocouples was discussed with off-site vendors. It appears that at least two different approaches may be possible.

An outline of the equipment specifications for the ARGUS Experiment was prepared. Tentative plans have been made for visits with off-site vendors to discuss technical problems associated with the experimental equipment.

Oxide Fuel Development. Components for the fabrication of sixty swaged UO_2 - PuO_2 PRTR fuel rods are on hand. Loading will begin during the week of February 26. Fuel materials and cladding components for an additional sixty swaged rods are being prepared.

Twelve Zircaloy tubes selected from four weight groups were filled with UO_2 and swaged to determine the degree of length control possible if the tubing weight were controlled. Tubes in each group weighed within one percent of each other. At the desired swaged diameter, the length variance within any group was 3/16 inch. The over-all length difference was 5/8 inch for this set of twelve tubes, which varied four percent in weight.

Accurate length control in the swaging process can be obtained by maintaining weight variance within a group of tubes to one to two percent of an average weight for the group. Present tubing groups may vary as much as eight to ten percent by weight, and the final lengths will of course vary, although the final length of the tubes in each group will be approximately equal. An alternative method, if necessary, would be to control all tubing to one to two percent of a desired weight by etching heavier tubes to the desired weight, or cutting lighter tubes to a length that would give the weight required.

Uniformity of PuO₂ distribution in vibrationally compacted fuel elements made from physical mixtures of UO₂ and PuO₂ appears to be a function of the manner in which the oxide materials are fed into the fuel element tubing.

Three size fractions of UO₂ were carefully weighed, measured, and blended in one-inch long by 1/2-inch diameter increments. The increments were poured into 88-inch long, 1/2-inch ID tubing and subsequent examination has shown: (1) that separation of coarse and fine fractions does not occur during loading and (2) that vibrational compaction does not redistribute the different sizes after loading. It was concluded that the degree of uniformity of PuO₂ in a vibrationally compacted fuel rod is determined by the accuracy with which the materials are loaded into the fuel tube.

Precision automatic weighing and blending machines are available which are capable of metering fuel material into tubing with accuracies of ± 0.2 percent of a given increment size. A combination of three such machines would be capable of automatically controlling PuO₂ distribution within the desired ± 5 percent of the nominal composition. Such a gravimetric feeder could also be programmed to produce heterogeneously enriched oxide fuel elements.

Extended Surface Plutonium Fuels. Two zirconium-clad, plutonium-Zircaloy core fuel plates were made by roll-cladding. A Pu - 15 w/o Zr alloy was made to provide core material for the roll-clad fuel elements. The alloy was cast into a thin cylinder and rolled into 0.020-inch foil. The foil was cut into rectangles to form a core. The core was polished, etched, wrapped in aluminum foil, and assembled into a picture frame between two zirconium plates. This assembly was placed in a protective steel sandwich. The entire assembly was hot-rolled at 840 C to a 93 percent reduction in area (R.A.). After rolling, the steel jacket was stripped from the fuel plate. No contamination was found on the fuel plates. One of the fuel plates was inspected. Ultrasonic inspection indicated complete bonding over the entire plate; in destructive testing the bond could not be broken. Radiography showed two phases present in the core. Preliminary metallographic examination showed some areas of thickening of the core although for the most part the core thickness was uniform. The thicker areas appeared both harder and more brittle suggesting an intermetallic; the source of the intermetallic is thought to be a reaction between the aluminum foil wrapping and the plutonium-zirconium core forming perhaps PuAl₂, although the intermetallic could also form from the eutectoid transformation of the core into Pu₆Zr and PuZr₂. Bonding between the zirconium plates occurs between 90 percent R.A., and 93 percent R.A. Plates rolled to less reductions could be stripped apart readily.

Uranium Fuels Development

Remote Fabrication Studies. The vibratory compaction equipment was tested in the remote fabrication area, where it operated satisfactorily. Remote operation involves positioning the equipment outside the contaminated area and transmitting the vibrational energy through a 15-foot beam to the fuel rods being compacted. Fuel is fed to the rods by a feeding device clamped to the beam.

As the first of a program of tests under way to determine the feasibility of fabricating a complete fuel element in one pass by vibrational compaction, the completely assembled element was suspended from an overhead beam and subjected to vibrational compaction acceleration forces up to 60 g. The results of this experiment are encouraging. Examination of the Mark II-C nested tubular fuel element revealed only minor scuffing at nodal points between the spacing ribs and tubular cladding. The vibration also broke the pin which positions the center rod of the element. This allowed more rattling of the inner rod but caused no additional damage.

Cladding Studies. The high voltage electron beam welding process provides a method of fabricating an "inverted" fuel element and an integral fuel element process tube configuration which, up to this time, has not been accomplished by any other means. This process allows unique shapes to be fabricated from simple part designs. For example, a component may be welded to the inside of a tube by melting through the tube wall.

Fuel elements of the two designs mentioned above are now being fabricated. Spacer ribs on five sets of cladding for eight-foot long nested tubular fuel elements are also being welded by this process.

Evaluation of AISI-406 Cladding. An experimental four-rod cluster, UO₂ fuel element was fabricated for irradiation to study the behavior of 0.009-inch wall AISI-406 stainless steel tubing as a cladding material. The 11½-inch long cluster contained two AISI-406 stainless steel clad rods and two 0.015-inch wall, 304-L stainless steel clad rods. All rods were PRTR fuel rod diameter, and the vibrationally compacted UO₂ (87 percent T.D.) was uniformly enriched to 3.3 w/o U-235. This test element, GEH-4-71, was delivered to the MTR for irradiation.

Physics Test Elements. Vibrationally compacted, cobalt-poisoned fuel elements for physics experiments are being fabricated. Tubing for the six fuel elements, plus eight spare rods, is available for the compaction step. All parts, including the cobalt-poisoned wire, have been fabricated.

Corrosion and Materials Studies

Aluminum Corrosion in PRTR. The feasibility of using 6061 aluminum alloy in a shim system employing a nearly saturated solution of cadmium sulfate was explored. Tests were conducted at 75 C with 300 g/l cadmium sulfate containing various inhibitors to prevent corrosion of the aluminum and plating out of the cadmium. The additions of sodium dichromate appeared the most promising of the inhibitors tested. Tests with 1/2 percent and 2 1/2 percent dichromate showed no measurable weight change of the aluminum and no obvious cadmium plating in eight days. However, minute amounts of unidentified turbidity developed. More refined tests were started.

The corrosion of aluminum alloys 6061, 6063, X-8001 and an experimental alloy in a gas mixture (argon (58%, water vapor (41%), hydrogen (0.6%), and oxygen (0.3%)) at 260 C and 315 C and at atmospheric pressure is being measured. Only slight corrosion was observed after 665 hours at 315 C and after 2700 hours at 260 C. Contaminating the gas mixture with CCl₄ from the gas makeup system caused severe attack of the 6061 alloy after 2719 hours at 315 C but did not adversely affect the other alloys.

High Temperature Oxidation of Zircaloy-2. Initial cubic rate constants for oxidation of Zr-2 at 800 C, 750 C, and 700 C in water vapor have been determined as a function of water vapor pressure. Rates have been determined using water vapor pressures of 5, 10, and 25 mm. It is found that there is a linear relationship between pressure and rate constant for each temperature. The five-fold increase in oxidant pressure causes a 65 percent increase in the rate constant at 750 C and lesser increases at the other temperatures.

Oxidation kinetics of Zr-2 and crystal-bar Zr have been determined in both 25 mm O₂ pressure and 25 mm H₂O pressure over the temperature range 600 to 850 C. The cubic rate law has been followed in all cases during initial oxidation. The crystal-bar Zr undergoes no transition in either water vapor or oxygen. The Zr-2 undergoes transition in both media at all temperatures investigated. The Zr-2 and crystal-bar exhibit an activation energy in oxygen of 49 K cal/mole, the Zr-2 in water vapor 46 K cal/mol, the crystal-bar in water vapor 59 K cal/mole. The similarity in activation energies indicates that the same general rate control mechanism is probably operative in all cases.

Deionization System for Rupture Test Facility. A project was undertaken to review the Fuel Element Rupture Test Facility (FERTF) high flow rate makeup deionization system supplied by the vendor, and to

prepare a startup procedure to place this system in operation. The review of the system and preparation of the operating procedure were completed, and a letter was written discussing the chemicals and equipment required to perform the regeneration. Initial operation of this system should begin in about two weeks (or as soon as it is mechanically ready for operation). It should be noted that this is a special (high flow rate) system and was specifically designed for operation with filtered Columbia River water as the influent. Since this type of water is not presently available at PRTR, 300 Area process water will be used.

Dissolved Oxygen Analyzer. The Industrial Instruments continuous dissolved oxygen analyzer calibration was checked over most of its calibrated range (0-500 ppb). The results indicate that the optimum flow rate through the analyzer is 125-150 milliliters per minute. At this flow rate the instrument generally agrees within $\pm 5\%$ of the values obtained by chemical analysis of samples. This analyzer will be installed in the KER-1 sampling system. Installation (including inlet and outlet deionizers) should be completed in 2-3 weeks. The analyzer will be installed on KER-3 or 4 when KER-1 is taken out of service for modification.

Pre- and Post-irradiation Evaluation of PRTR Pressure Tubes. Three creep rupture tests on section from the annealed portion of rejected PRTR process tubes are continuing.

<u>% of Ultimate Burst Strength</u>	<u>Hoop Stress psi</u>	<u>Hours, at Temp.</u>	<u>Temp., °C</u>	<u>Total Strain %</u>
75	36,200	720	288	1.1
85	39,600	720	288	3.3
88	40,900	325	288	5.1

Failure occurred after one hour at a hoop stress of 43,700 psi (94% ultimate) and at 288 C. A local circumferential elongation of 119% was measured on a one-inch grid across the fracture. The maximum elongation measured over the entire circumference was 60%.

Replicas of the localized corrosion areas at the points of contact with the upper fuel end bracket supports in tube 6061 from channel 1756 were measured. The maximum depth of penetration was 3 mils. The corrosion marks where the lower end bracket supports contacted the tube were 6.5 mils deep. A room temperature burst test of the annealed section containing the lower corrosion marks burst at 96,300 psi hoop stress (8350 psi internal pressure). The rupture

specimen has not been cut for examination and measurement; however, it did appear to be ductile with the fracture occurring in the portion below the fuel element end bracket supports. For comparison an unirradiated section of PRTR tube burst at a hoop stress of 96,400 psi.

In-reactor Monitoring. Six process tubes were examined during February. No change in ID was noted. Gas gaps for channels 1348 and 1748 were near the 50-mil operating minimum with 55- and 90-mil gaps, respectively. No unusual marks were seen in the three tubes which were visually examined, but the quality of visual examination was not as good as previous inspections. The exact cause of the reduced visual quality has not been determined.

Construction of the Mark III Monitoring Unit is estimated at 80 percent complete. The TV picture size when viewed through the omni-scope is approximately two-thirds as large as specified but of excellent quality. The required optical changes to increase the size of the image will be made. The gas gap measuring unit has been redesigned and the machining difficulties encountered with the fiberglass body of this unit will be alleviated by substitution of a phenolic base material. Preliminary calibration of the ID measuring device is complete. The mockup facility in 314 Building is complete. With the exception of the gas gap unit, both the monitoring unit and the mockup facility are ready for preliminary tests.

Radiometallurgy Laboratory Studies

The top section from PRTR process tube 1756 was sectioned for ring flattening tests and burst tests and micro examination of fretting pits. Silicone rubber replicas were made of the fretting zone where the upper end bracket contacted the tube wall. Contact pads on the lower end bracket were examined metallographically but no significant corrosion or hydriding was found. Burst test sample A2C was burst at 8350 psi and began yielding at about 7900 psi (RM 356). Density tests were completed on 16 PuAlSi samples which had been annealed for one hour at 465 C. The samples are now being given a 10-hour anneal (two samples per annealing) at 465 C (RM 666). Thirteen samples were taken from various locations along the 19-rod PuAl PRTR element for an isotopic analysis by the Applied Physics group. In addition, selected samples were sectioned and dissolved in HCl-HNO₃ for burnup analysis which will be calculated as grams of Pu per gram of irradiated element (RM 750).

Low density UO_2PuO_2 fuel capsule GEH-14-68 had begun to sinter at the core; however, the fuel was porous and a central void had not begun to form. Most of the fuel in GEH-14-69 had recrystallized and a central void was found. Discriminating examination of GEH-14-66 disclosed a spherical shaped, metallic mass about 1/8 inch in diameter in the fuel about 1/4 inch from one end. A portion of the internal surface metal from the opposite end cap was missing leaving a dome shaped void in the end cap (RM 654).

Three high density UO_2PuO_2 capsules, GEH-14-20, 82 and 91, were examined metallographically and the cores of all samples exhibited the typical recrystallized, columnar structure (RM 650 and 655).

The fractured surfaces on irradiated UO_2 samples 52 and 54 were replicated and subsequent metallographic examination is in process (RM 634). A composite of the entire cross section of GEH-3-52, UO_2 exposed to 50,000 MWD/T, was photographed at 75X magnification and an autoradiograph was made of the examined surface. A sample of UO_2 run on the x-ray diffraction equipment indicated the presence of UO_2 only, the inclusion concentration not being large enough to yield a diffraction pattern (RM 642). Examination of Rod #2 from GEH-4-62 exhibited about one inch of recrystallized UO_2 near the site of the enriched UO_2 . Composites of both transverse and longitudinal sections of the recrystallized UO_2 were photographed and autoradiographs were made of the photographed sections (RM 643). A portion of a single crystal of UO_2 irradiated to 10,000 MWD/T, GEH-12-242, was polished and etched, but no structure was discernible in a metallographic examination. Microhardness traverses also were run on the single crystal (RM 645).

Approximately one-half of a uranium sphere was annealed for one hundred hours at 650 C. Following the anneal the sample was in small pieces (practically powder) making subsequent metallographic examination impossible (RM 510). A total of twenty samples were dissolved for burnup analysis and four samples were declad for density determinations.

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

2. PLUTONIUM UTILIZATION STUDIES

Plutonium Carbide

The arc-melted Pu-C samples containing between 30-52 a/o C which were annealed for 91 hours in vacuum below the ϵ + PuC phase horizontal

have been analyzed by x-ray diffraction. Several unusual results were noted and are summarized:

1. Lattice parameters in the region 30.5-43.5 a/o C decrease uniformly with composition and range from 4.9739 to 4.9700 Å, respectively.
2. The phase relations in this region are a coexistence of delta plutonium and PuC. Impurity analyses are presently being run to determine whether sufficient quantity of a delta stabilizer exists.
3. From 43.5 to 51.5 a/o C lattice parameters increase uniformly with composition reaching 4.9796 Å at the latter composition.
4. Phase relations in this region are single-phase PuC between 43.5 and 47.5 a/o, and a mixture of PuC + Pu₂C₃ above 47.5 a/o C. The peculiarity here is that the lattice parameter of the PuC phase increases while it coexists with a second phase.

In another experiment, samples over the composition range previously mentioned were cycled 50 times through the zeta transformation temperature of 575 C. They were finally furnace quenched from 615 C. X-ray examination showed a constant lattice parameter of 4.9740 Å for samples containing less than 38 a/o. Only a single PuC phase was seen. From 39.5 to 52.5 a/o C, the lattice parameter remained constant at 4.9790 Å. Again, above 47.5 a/o C, Pu₂C₃ coexisted with PuC. Metallographic examination of these samples will be performed.

Kilogram quantities of PuC have been manufactured by the reaction of plutonium metal and porous graphite in an inert tungsten arc. A few of the buttons have shown a low carbon analysis due to atmosphere leaks, but over 94 percent of them were within the desired composition limits.

Some attempts were made to arc-cast this material in a water-cooled hearth. No difficulty was encountered as long as the castings were small, 20 g or less. As they became larger some porosity was noted and a considerable amount of Pu₂C₃ was found. This is most probably caused by the slow cooling rate of the large buttons through the peritectoid region.

Plutonium Nitride

Synthesis of PuN was accomplished by arc-melting a small plutonium metal rod under one atmosphere of nitrogen. In order to obtain a product containing no residual alpha-plutonium, the specimen was remelted in nitrogen repeatedly. Analysis by x-ray diffraction showed only PuN to be present in the remelted specimen. The single-phase PuN formed is face-centered cubic and has a unit-cell dimension of 4.905 Å.

The melting point of PuN in argon is $2450\text{ C} \pm 50$. The material appeared stable until the melting point was reached; it then volatilized very rapidly. One might speculate a decomposition of the PuN to plutonium plus nitrogen, then vaporization of the plutonium.

Plutonium mononitride is not compatible with boiling water. The PuN oxidizes and disintegrates into fine PuO₂ particles at a high rate.

In massive form, PuN oxidizes relatively slowly in air. A polished specimen showed no signs of oxide formation after being exposed to air for 72 hours. Its ignition point in air is 350 C. A weight gain of 8.96 percent was observed after complete oxidation. This suggests either the existence of some residual plutonium or the possibility of nonstoichiometric PuN.

Plutonium hydride powder was pressed into a small pellet and heated to 650 C in nitrogen. No reaction was apparent after 100 minutes, so the temperature was increased to 800 C and held for 30 minutes prior to furnace discharge. The outer layer of the highly pyrophoric pellet turned from black to the characteristic PuO₂ yellow on removal from the furnace. It was then immediately transferred to a vacuum bell jar. Preliminary x-ray analyses show the reacted pellet to consist of PuN with a trace of PuO₂. Better control of the nitrogen atmosphere should result in a highly successful technique for the synthesis of PuN.

ZrO₂-PuO₂ and MgO - PuO₂ Irradiation Capsules

Mixtures of plutonia with various inert diluents such as ZrO₂ and MgO are of interest for spike fuel element application. Eight capsules containing MgO-PuO₂ pellets and eight capsules containing ZrO₂-PuO₂ pellets are under irradiation in the ETR. Two compositions of each material are being irradiated under different temperature conditions for different periods of time. The effects of plutonium concentration, maximum operating temperature, and exposure on the operating performance of these fuel materials will be determined. Maximum

operating temperatures for the $\text{ZrO}_2\text{-PuO}_2$ specimens range from 1650 to 2800 C and corresponding surface heat fluxes of 390,000 and 506,000 Btu/hr-ft². Maximum operating temperatures for the MgO-PuO_2 samples are 1700 and 2200 C with corresponding surface heat fluxes on the Zircaloy cladding of 666,000 and 1,150,000 Btu/hr-ft². Because of the better thermal conductivity of the MgO-PuO_2 specimens, they must be operated at higher powers to attain temperatures comparable with similar $\text{ZrO}_2\text{-PuO}_2$ samples.

3. UO₂ FUELS RESEARCH

Fused UO₂

Fused UO₂ core material for vibrationally compacted fuel elements must be free of pores if high bulk densities are to be attained. Purchase specifications currently require fused UO₂ to have a minimum density of 10.80 g/cc, as measured by mercury displacement at one atmosphere pressure. If the measurement is made after evacuating the sample, open pores larger than 10 microns in diameter are filled by the mercury, and thus their effect is not included in determining the UO₂ density. Recent measurements with a mercury porosimeter, with which the volume of pores with diameters as great as 100 microns can be determined, revealed that some lots of fused UO₂ have a significant amount of open pores with diameters in the range 9-100 microns. Average density found for two lots of fused oxide decreased from 10.84 g/cc to 10.72 g/cc when the minimum pore size considered was increased from 9 to 100 microns. Metallographic examination revealed that these two lots of UO₂ consisted primarily of polycrystalline material with grain diameters of about 500 microns. A large number of voids approximately 15 microns thick and 200-300 microns wide were present at the grain boundaries.

Fuel Evaluation

Irradiation of UO₂ Single Crystal. Post-irradiation examination of a UO₂ single crystal irradiated to 7400 MWD/T has revealed a number of property changes resulting from irradiation.

- (a) The hardness of the UO₂ increased significantly during irradiation, apparently as a function of crystal orientation. However, wide variations in hardness were observed.
- (b) The melting point increased approximately 50 C from the pre-irradiation value. A noticeable suppression of the sublimation of UO₂ also occurred. No gross fracture or spalling occurred during heating.

- (c) Small differences, primarily in the surface texture, were observed by electron microscopy of the pre-polished surface before and after irradiation. No significant changes were noted in the surface artifacts initially present in the irradiated sample. Detailed comparative examinations are continuing on as-irradiated, repolished, and fracture surfaces of both control and irradiated crystals.
- (d) No detectable amounts of Kr or Xe were released during irradiation, although a calculated 0.18 cc/g were produced. Approximately 0.2 cc/g of hydrogen was released from the crystal.

Irradiation of Swaged UO₂. Swaged UO₂ irradiated to high exposures at central temperatures near the melting point has characteristics distinctly different from those observed in UO₂ irradiated to low exposures under similar conditions. The radially oriented columnar grains are long and extremely thin, and the voids are more nearly distributed uniformly throughout the cross-section of fuel. These characteristics were observed previously in a 0.57 inch OD capsule (GEH-14-177) irradiated to ~ 37,000 MWD/T_U and more recently in a 0.77 inch OD capsule (GEH-3-52) irradiated to ~ 50,000 MWD/T_U.

Fission gas release from the fuel in GEH-3-52 was sufficient to rupture the 0.050-inch thick Zircaloy cladding even though the capsule was contained in a shrink-fit, thick-wall aluminum can during irradiation. Approximately 190 ml (@ STP) of gas were obtained from the capsule; of this, 76.1 percent of the volume was fission gas. The volume ratio of Xe to Kr was 9.57.

Fission Fragment Migration. Cross-sections of fuel elements with 50,000 and 37,000 MWD/T exposures were sampled along several diameters to determine the radial distribution of fission fragments. A fuel core containing heterogeneous 10 w/o ThO₂ - 90 w/o UO₂ irradiated to 1000 MWD/T of uranium was similarly sampled. The specimens were obtained by scratching the polished fuel cross section along a diameter and removing loosened particles with strips of celluloid soaked in acetone. The strips were subsequently sectioned into 0.05-inch segments, each of which was analyzed for long-lived, solid fission fragments. This method of sampling is an improvement over prior techniques which depended upon cellophane tape to retrieve the loosened UO₂ particles.

Distribution of Recoil Fission Fragments. Fission fragment depletion by recoil loss at the boundaries of UO_2 fuel cores was investigated by measuring the concentration and distribution of fission fragments in aluminum irradiated adjacent to a UO_2 single crystal which received 10,000 MWD/T exposure. One-micron "layers" were removed from the aluminum by chemical milling in a 12.5 percent NaOH solution. Each layer was analyzed for Zr-95, Ce-144, Cs-137, and Ru-106. The plot of recoil concentration versus depth of penetration showed a maximum at a depth of approximately 13 μ for each isotope.

Electron Microscopy

Investigation continued on methods of preparing thin samples of UO_2 from bulk specimens for transmission electron microscopy. Thin sections estimated to be a few microns thick were obtained by slow Syntron polishing, but edges tended to chip off or dissolve in the chromic acid lubricant, rather than become thinner. Further polishing with an oil lubricant is being investigated, but it appears likely that some other thinning technique, such as a chemical or electrolytic method, will be required to achieve sufficiently thin areas.

High Temperature-Pressure Experiments

Induction Heating of UO_2 . A pronounced difference in the activity of UO_2 in helium and argon atmospheres at a temperature near the melting point of UO_2 was observed. The UO_2 was melted in a tungsten tube susceptor by induction heating within a controlled atmosphere chamber. Gross vaporization occurred in a helium atmosphere, without obtaining a pool of UO_2 . The vapors deposited in a selective area adjacent to the induction heater coil on the outside of the tungsten susceptor. The UO_2 vapor traveled about one and one-half inches, passing a cool tungsten zone to deposit on the hot tungsten surface. Within the argon atmosphere very little UO_2 vaporization occurred. Molten UO_2 readily flowed through a 0.005-inch gap in the tungsten susceptor and solidified into spheres and fingers as it dropped through the argon atmosphere. Shrinkage during solidification produced large central voids in the spheres and fingers of UO_2 .

Direct Resistance Heating of UO_2 . Simulation of in-reactor fuel heating has been demonstrated by relatively simple, direct resistance heating of UO_2 in a closed capsule. This procedure may provide a simple method of studying certain aspects of relocation, fission product migration, fuel cracking, and void formation in various

nuclear fuel materials. The uranium oxide is heated by its own electrical resistance. Electrical connection is provided at each end of the fuel section by tungsten electrodes which extend through insulated end closures. The outer surface of the fuel section is cooled by water. Micro-examination of the fuel rod from the first test has been completed. There was a central void within a fused core of the specimen; this fused core was surrounded by non-fused, swaged material adjacent to the cladding. Both circumferential and radial cracks were evident. A maximum of 500 amperes was employed in this experiment, to produce temperatures well above the normal operating range of fuel elements in reactor irradiation experiments. Fuel cladding burnout occurred at several points on the capsule and the tungsten electrodes near the central core of the UO_2 were melted, resulting in a selective dispersion of tungsten in the fused portion of the UO_2 .

4. BASIC SWELLING PROGRAM

Irradiation Program

A total of four temperature controlled capsules are being irradiated at this time. These capsules were charged in tandem with two capsules to each test tube. Irradiation temperatures for three of the four capsules regardless of the reactor operating conditions are one capsule at 575 C and two capsules at 625 C. The fourth capsule is operating at a temperature of approximately 400 C only during reactor operation. The heater in this capsule was broken during assembly, subsequently repaired, and had operated satisfactorily during all laboratory tests. However, after the capsule was charged, the heater failed.

One additional capsule has been completed and laboratory tested. Assembly of another capsule has begun. It is also planned to charge these two capsules in tandem.

Two capsules previously irradiated will be shipped shortly to Radiometallurgy for disassembly and post-irradiation examination of the specimens. It is anticipated that further capsule assembly of this design will be delayed until electrical resistance heaters now on order are delivered. Meanwhile, efforts will be directed to the development and testing of the metallographic swelling capsules. This capsule concept has been modified to provide a constant temperature only during reactor operation. Some effort is also being directed into the design of a temperature controlled capsule that will impose a hydrostatic restraint on the fissionable material.

Post-Irradiation Examination

Specimen 4-2, which was annealed at 650 C for 100 hours, was inadvertently dropped during handling; it broke into small fragments and can no longer be subjected to intended metallographic study or density measurements. A diffusion couple, GEH-14-281, which had been annealed at approximately 650 C for 100 hours was repolished and re-etched for replication.

Pore Size Distributions

Pore diameter data from specimens 4-2, 4-3, 4-4, 5-1, 5-2, and 5-3 in the as-irradiated and in the post-irradiation annealed states have been submitted for computer determination of pore volume and pore density distributions. The final results have not as yet been received.

Second Phase Distributions in Dilute Alloys of Uranium

Fourteen of thirty specimens which had received a variety of heat treatments to develop the "delta" phase have been polished, etched, and replicated. The remaining sixteen specimens will be similarly treated and the presence, morphology and distribution of second phase precipitates will be established by electron microscopy.

Restrained Irradiations

In order to gain insight into the influence of Zr-2 cladding on the swelling of uranium, Zr-2 clad uranium rods with selected uranium temperatures, cladding thicknesses, and exposure are being irradiated in NaK-filled temperature monitored capsules. The volume changes for rods 105-1 and 105-2 were incorrect as reported in the previous monthly report and should be 3.4 and 2.2 percent $\Delta V/V$, respectively.

Six additional fuel rods were removed from capsules irradiated in the MTR. Severe splitting of the cladding and oxidation of the uranium occurred. Density and burnup samples from each rod are being processed. One additional capsule containing two fuel rods is being returned from the MTR.

5. IRRADIATION DAMAGE TO REACTOR METALS

Alloy Selection

The determination of the irradiation effects on mechanical and physical properties of reactor structural materials is necessary for the

development of advanced reactor concepts. A program for the determination of irradiation effects on structural materials is being initiated as a coordinated effort among several sites to provide accurate data for several reactor materials.

Efforts during the past month have been directed toward the selection of alloys to be investigated under the program. A cursory literature survey indicates that approximately twelve alloys could be investigated over an extended period of time. These initial alloys would include austenitic stainless steels, nickel base alloys, a ferral alloy, and low alloy ferritic steels.

An extensive effort will be made to insure that all material procured for the program is representative of commercially obtainable heats. This will be assured by contacting various steel manufacturers to gain information from which stringent specifications on chemical composition, mechanical properties, and metallurgical structure can be prepared. Sufficient amounts of each alloy will be procured to supply the needs of the entire program.

In-Reactor Measurement of Mechanical Properties

In-Reactor Measurements. During the past month Capsule III-3 has been used to determine in-reactor activation energies for the creep of Zircaloy-2. The method used in-reactor was identical to that in the pre-irradiation study; namely, the creep rates established just before and just after an abrupt temperature change are measured and ΔH is calculated from the equation:

$$RT \ln \frac{\dot{\epsilon}_2}{\dot{\epsilon}_1} = \left(\frac{1}{T_1} - \frac{1}{T_2} \right) \Delta H$$

The temperatures at which activation measurement was conducted ranged from 300 C to 400 C.

The activation energies obtained for temperature increases were near 85,000 cal/mole. Activation energies based on temperature decreases were generally lower than 85,000 calories. Reproducible values near 52,000 cal/mole were obtained.

The method used in this experiment produces identical activation energies for heating and cooling if the specimen structure is not appreciably altered during the temperature changes. Differences between heating and cooling values can be interpreted as being the

result of structure changes. Ex-reactor studies have shown that no structural change occurs during small abrupt temperature changes so the observed differences in-reactor must be associated with radiation damage phenomena.

The nature of the creep curve derived from Capsule III-3 during shutdown indicates that creep is inhibited by radiation-induced defects which can anneal at 310 C. At any temperature near 310 C in-reactor, a steady state defect concentration will be established which is characteristic of the temperature and flux. The mechanism of creep above 300 C has been established as dislocation climb and is associated with an activation energy of 58,500 cal/mole. Since the number of creep inhibiting defects which defines the specimen structure depends on temperature and the dislocation climb rate also depends on temperature, an activation energy greater than 58,500 cal/mole will be observed upon a temperature if the time to establish the defect level is short compared to the time required to change the temperature. In contrast, if the temperature is lowered the defect level is essentially frozen in, that is, little change in structure occurs, and only the activation energy for climb is observed.

These preliminary data from Capsule III-3 demonstrate the utility of activation energy measurements in interpreting and measuring dynamic effects of irradiation damage.

Capsule and Instrument Development. Specifications are being prepared for 18 additional creep capsules. The capsules will be nearly identical to the third generation series with water cooling. They will be shipped as sub-assemblies rather than completely assembled units. In this form, the capsules can be inspected for any damage during shipping. This will also permit the changing of specimens, or tailoring the heating and cooling capabilities for a particular test. The capsules are to be operable either with or without the water cooling. Operating temperature ranges with cooling will range from 220 to 350 C and 400 to 750 C without cooling. A few leads and parts are being relocated to permit future modifications such as direct cooling on the ends of the specimens or cooling of the micro-positioner and transducer in the event higher or lower operating temperatures are desired.

Calibration checks of the variable permeance sensor were conducted after an in-reactor creep test was terminated. Using new and improved readout instruments gave an accuracy figure that was much higher than originally predicted. Sensitivity, resolution and short-time stability of the variable permeance transducer all approached the limit

of measurement by the mechanical micropositioner which is 20 micro-inches for any one reading. The reliability of the variable permeance sensors will result in a reduction of the required operational life for the mechanical micropositioner. A total of one percent strain in the specimen can be measured with the variable permeance transducer for one adjustment of the micropositioner.

Irradiation Effects in Structural Materials. The purpose of this investigation is to determine the combined effects of radiation and reactor environment on the mechanical properties of structural materials. Special attention is currently being given to the determination of mechanical property changes produced in metals by irradiation at elevated temperature in contact with hot water.

One hundred twenty Zircaloy-2 notched tensile specimens were autoclaved and assembled into quadrants for irradiation in the G-7 hot water loop of the ETR. The 0, 10, 20, and 40 percent levels of cold work and both the longitudinal and transverse directions with respect to rolling are represented. Irradiations will commence with ETR Cycle 45. Tensile, bend test, and notched tensile specimens of Zircaloy-2 were autoclaved and assembled for exposure in an ex-reactor hot water loop. These exposures will duplicate the in-reactor exposures and will permit the evaluation of environmental effects aside from the neutron flux.

Tensile tests at room temperature were conducted on specimens of types 304 and 347 stainless steel. The data are in general agreement with handbook values. In the case of 347 stainless steel, specimens cut both transverse and longitudinal with respect to the rolling direction were tested. Ductility alone revealed a direction effect for the annealed condition, the transverse direction exhibiting the higher total elongation and work hardenability. For specimens cold worked 25 percent, a directional effect was observed for both strength and ductility. In this case the transverse direction exhibited a lower yield strength, higher tensile strength, and lower percent reduction in area.

Tensile tests at 300 ± 2 C were conducted on Zircaloy-2 specimens cold worked 70 percent. With the exception of percent reduction in area, the tensile properties at 300 C showed little or no directional effects. The percent reduction in area was higher for the transverse direction at 300 C. A moderate strain rate effect was observed at 300 C. That is, an increase in cross-head speed from 0.005 to 5 inches per minute resulted in an increase in yield and tensile strength of about 9,000 psi. This effect was observed for

both the transverse and longitudinal directions, and is contrary to an expected small strain-rate effect due to strain aging at this temperature. Higher strain rates had little effect on ductility.

In the course of the strain rate studies it was observed that a significant adiabatic temperature rise occurs for Zircaloy-2 at strain rates above $10 \text{ in/in/min}^{-1}$ and for type 304 stainless steel for strain rates above $0.25 \text{ in/in/min}^{-1}$. This effect tends to reduce the strength increase with strain rate and has been observed to cause a marked drop in strength at high strain rates. To determine the magnitude of this effect, specimens were pulled in tension at various strain rates both in air and water. At low strain rates the curves of strength versus strain rate overlapped; however, the curves deviated from one another above about $0.25 \text{ in/in/min}^{-1}$. For example, the difference in tensile strength of 304 stainless steel for a strain rate of $10 \text{ in/in/min}^{-1}$ was about 10,000 psi. Such a decrease in strength for this material corresponds roughly to a temperature rise of 400 F.

Damage Mechanisms. Impurity elements are known to have a pronounced effect on the physical and mechanical properties of metals. Recent investigations have shown that impurities also are an important factor in determining the response of a metal to neutron irradiation. The purpose of this study is to establish the nature of interactions between impurity atoms and defects introduced by irradiation and to investigate the possibility of neutralizing the deleterious effects of the impurities by means of chemical stabilization.

The first phase of this program consists of an investigation of neutron damage in high purity iron. Controlled additions of carbon and nitrogen then will be made to zone-refined iron. Additions of a stabilizing agent such as titanium will also be investigated. In unirradiated iron containing carbon and nitrogen, titanium ties up these elements as carbides and nitrides; however, it is not known whether these compounds will be stable in a neutron flux. The investigation will cover the following materials, over a range of solute contents: (1) pure iron, (2) iron plus carbon, (3) iron plus nitrogen, (4) iron plus carbon plus titanium, (5) iron plus nitrogen plus titanium, (6) iron plus titanium.

A number of experimental techniques will be employed in this program, including (1) tensile tests, (2) x-ray diffraction studies, (3) electron transmission microscopy, (4) low frequency internal friction measurements.

Addition of impurities and stabilizers will be made in HLO, but it is not yet known whether the high purity iron will be prepared in HLO or purchased from external sources. Analysis of these two alternatives is in progress.

A preliminary irradiation of ingot iron specimens has been completed. The stock material was decarburized in wet hydrogen and then "nitrogenized" in an ammonia-hydrogen atmosphere. Tensile specimens were machined from stock material, from decarburized material, and from nitrogenized material. The specimens were irradiated to five exposures: 10^{17} , 5×10^{17} , 10^{18} , 2×10^{18} , and 5×10^{18} nvt (fast). Six specimens of each composition were included in each exposure. Testing has been completed on the control and low exposure (10^{17} nvt) specimens; no marked hardening was observed, but all materials decreased in ductility, the ingot iron having the greatest decrease in ductility, the nitrided specimens showing a smaller decrease in ductility, and the decarburized iron the smallest decrease. All materials showed pronounced increases in strain-aging response after irradiation. Testing of higher-exposure specimens is under way.

6. GAS GRAPHITE STUDIES

EGCR Graphite Irradiation. The H-3-4 capsule containing EGCR graphite was installed in the General Electric Test Reactor on February 9. This is the fourth period of irradiation in the EGCR irradiation program. Fourteen of the twenty-four samples in the capsule have been irradiated to varying exposures, the maximum of which is 8.7×10^{21} nvt ($E > 0.18$ Mev). This is equivalent to approximately 58,000 MWD/AT in the EGCR, or about 75 percent of the lifetime exposure. Modifications to the H-3-4 capsule were completed, and a new guide tube was constructed to provide protection against wear in the reactor. The capsule is operating satisfactorily and all nine thermocouples functioning properly.

Flux Intensity Test. The GEH-13-8 graphite irradiation capsule was installed in the N-5 position of the ETR on February 14. The objective of this test is to study the effect of flux intensity on property changes at a controlled temperature. The fast neutron flux, $E > 0.18$ Mev, will differ by a factor of six over the length of the capsule. All seven thermocouples and four heaters are operating properly. Preliminary data indicate sample temperatures will be controlled at 625 C or 650 C. The graphites being tested are CSF, NPR core, and TSFBF. Three samples previously irradiated in the GEH-13-5 capsule are included in the GEH-13-8 capsule.

Graphite Oxidation in Air. Oxidation experiments were conducted on samples which had been irradiated to 3.2×10^{21} nvt ($E > 0.18$ Mev) at 1150 C to study the effect of prolonged high temperature irradiation on oxidation rates in air. The results indicate that this irradiation increases the oxidation rate by a factor of approximately two; however, the activation energy is apparently unchanged. Previous tests indicate that an exposure of 1×10^{20} nvt ($E > 0.18$ Mev) at 550 C had no discernible effect on oxidation rates.

A study is in progress to determine the influence of the intensity of gamma radiation on oxidation rates in air. Experiments were recently completed with EGCR graphite in a gamma field of 2.5×10^5 R/hr at temperatures varying from 525 to 650 C. Results indicated that the activation energy at this intensity is approximately the same (50 ± 5 kcal/mole) as in the absence of radiation. In a gamma field of 1×10^6 R/hr an activation energy of approximately 30 kcal/mole was measured.

Irradiation of Fermi Reactor Graphite. The first irradiation of Fermi Reactor boron-impregnated reflector graphite in the Snout Facility was concluded at a goal exposure of $\sim 10^{18}$ nvt at 20 C. Prior to irradiation the samples were preheated to 600 C, measured and weighed. During irradiation the Great Lakes 7 w/o boron sample lost 4.5 percent in weight. A white glassy substance about 1/16-inch thick was observed on the surface of this sample. All other samples lost negligible weight and did not change in length. A second irradiation will be performed at approximately 100 C in an insulated capsule.

Gas Coolant-Structural Materials Compatibility. A high temperature carbon dioxide compatibility test with periodic weight change recording was applied to Hastelloy X tubing and to AISI 406 SS (Carpenter 1-JR) seamless tube sections. Hastelloy X appears to resist attack by dry, purified carbon dioxide flowing at 0.5 cfh at 1875 F (1025 C) for periods of 500 hours as far as weight gain is concerned but is susceptible to preferential surface oxidation. AISI 406 SS on the other hand, at a ten-fold higher rate of oxidation than Hastelloy X after 50 hours of exposure under identical conditions, forms a uniform oxide coating but is apparently susceptible to a breakaway type of edge attack at this temperature.

7. GRAPHITE IRRADIATION DAMAGE STUDIES

Effect of Graphitization Temperature on Contraction Rate of Graphite. The first indication of a decrease in the rate of graphite contraction after extended irradiation at 500 to 600 C has been observed in

samples from a series of experimental molded graphites in which the final heat treatment temperatures were varied from 1370 to 2700 C. The samples were prepared by Battelle Memorial Institute using a Texas coke filler and a medium-hard coal-tar pitch binder. Final bulk density of the unimpregnated samples was 1.50 g/cm³. In previous irradiation of this series of samples an inverse relationship between final processing temperature and the rate of contraction was indicated up to an exposure of 4000 MWD/AT. The graphite processed to 1370 C contracted at an average rate of 0.4 percent per 1000 MWD/AT and graphite processed at 2570 C contracted at an average rate of 0.01 percent per 1000 MWD/AT. Between 4000 and 8000 MWD/AT the average rate for the 1370 C graphite was 0.06 percent per 1000 MWD/AT, whereas no change in the rate was observed for graphites processed at 2300 C and above. The most pronounced reduction of contraction occurred in the 1370 C graphites. Ratios of the average contraction rate below 4000 to that between 4000 and 8000 MWD/AT for 1370, 1800, and 2100 C graphites were 6.7, 1.3 and 1.7, respectively. Maximum contraction observed in the 1370 C graphite at 8000 MWD/AT was 1.85 percent. Further irradiation is required to determine if contraction saturates in the 1370 C graphite or continues at a reduced rate. The samples will be recharged for further hot test hole irradiation.

8. ALUMINUM CORROSION AND ALLOY DEVELOPMENT

Corrosion of Aluminum. Dynamic corrosion tests are being conducted on a high purity aluminum alloy (1-2% Ni, 1-8% Fe in 99.995% Al) at 340 C at linear flow rates of 25 fps. At one month, penetrations on the new alloy were 1.2 mils (average for nine coupons) compared to 1.6 for X-8001 (average for three coupons). Previous static tests on the same alloys (HW-68253) have shown improved corrosion resistance for the new alloy over X-8001 at this temperature. Dynamic tests will be continued to three months.

Heat Transfer Testing. Corrosion testing of X-8001 cladding at 228,000 Btu/hr-ft² continued during the month. The test is being operated at a bulk water temperature of 600 F using high purity de-ionized water, and the temperature drop across the water, oxide and crud films continues to increase slightly during operation and decrease slightly immediately following a shutdown. The test has been in progress approximately 2500 hours. The A-288 alloy continues to show better film retention compared to the X-8001 alloy. In general, this is indicative of lower penetrations.

H-1 Loop. A Zircaloy liner tube, 1.76-inch OD x 1.68-inch ID, has been charged into the in-reactor facility for use during initial

coupon tests. When the loop is revised to operate with fuel, the same liner tube will be used. A nozzle was also installed on the in-reactor test section to permit easier charging and discharging.

Aluminum coupons with encapsulated silver discs have been fabricated with newly designed components to permit adequate weld penetration. These coupons are of alloys X-8001 and A-288. In addition, one coupon contains rhodium. These coupons plus similar coupons of 304 SS and Zircaloy-2 will be charged in H-1 loop and will serve to measure the behavior of these materials in an intense beta field obtained by irradiation of the silver. Any crud obtained will be compared to control coupons placed in the H-1 loop at the same time. The production test authorizing the charging of these coupons is currently being circulated.

The deionizer for H-1 is still being fabricated with vendor shipment due the end of February.

9. USAEC-AECL COOPERATIVE PROGRAM

Ultrasonic Testing of Sheath Tubes

Using a longitudinally oriented OD notch, 30 mils long by 3 mils deep, the ultrasonic response was measured as a function of entry angle for two types of crystals, a cylindrically focused crystal and a flat rectangular crystal. In the longitudinal test as the defect on the tube is rotated away from the crystal there is a succession of signals which rise and fall as they move away in time from the direct surface reflection. To avoid any interference from a direct surface reflection, the first signal is not used. The second major signal was chosen to run the curve of response versus entry angle for the cylindrically focused crystal because the third response was too sensitive to changes in metal path and entry angle to give reproducible results. The range of entry angles encompassed was 8° through 27° . The results show that this transducer is not very sensitive to changes in angle; however, the signal is very sensitive to changes in length of metal path.

Using the unfocused 1/4-inch x 1/2-inch rectangular crystal, the second signal rose to a maximum height and back to zero at entry angles from 10° to about 30° , while the curve from the third signal encompassed a range of angles about three times as wide with a 300 percent increase in signal strength at the same gain setting. The third signal was also less sensitive to changes in metal path than the second signal.

Comparing the results from the two types of crystals, the response from the rectangular, unfocused crystal is greater for the same gain setting over a wider range of entry angles and lengths of metal path than that obtained using the focused crystal. Future testing for longitudinal oriented defects will be performed with the unfocused crystal.

10. REACTOR STUDIES PROGRAM

Advanced Concept Studies

In the studies of compact-core fast reactor concepts for application in space and similar remote environments, attention has been directed to the criteria for control systems for such reactors. Reactors of this type would be required to operate unattended for extended periods of time and would probably be required to start up automatically or on remote command. A primary goal of the present studies is devising of control schemes which would permit stable self-regulated operation of such a reactor, with minimization or elimination of mechanical components and externally-actuated control devices.

Control methods which have been considered to date derive control action through utilization of temperature effects, acting to increase neutron leakage from the core or to decrease reactivity as temperature increases.

Methods which have been considered include:

1. "Thermometer" fuel elements with high rates of thermal expansion, so arranged that at high temperatures a portion of the fuel is expanded out of the core. Such an element may consist, for example, of a plutonium- or uranium-bearing glass, molten under reactor operating conditions, in a jacket which permits expansion of the fluid out of the core.
2. Use of a "variable density" hydrogenous reflector such as zirconium hydride, whose hydrogen content can be made to vary widely with temperature.
3. Control of effective reflector density through evaporation and condensation of a liquid reflector.
4. Utilization of thermal expansion to open gaps in a core or reflector. Remote or automatic startup of reactors of this type by gradual release of a neutron-absorbing fluid from an in-core capsule is being studied.

D. RADIATION EFFECTS ON METALS - 5000 PROGRAM

This study is aimed at establishing how neutron irradiation and carbon impurity levels affect the metal molybdenum. One hundred twenty single crystal rod specimens, one and one-half inches in length, representing the three carbon levels, 20, 150, and 450 ppm, have been ground into uniform cylinders. A reduced section, approximately 0.08 inch in diameter and 3/8 inch in length was then ground in each specimen prior to electro-polishing. The orientation of the gage length sections is now being determined by x-ray methods and specimens are being classified into single crystal, bi-crystal and multi-crystal groups.

The geometry of the single crystal tensile specimens now being processed for encapsulation differs markedly from the bar configuration of single crystal tensile specimens first prepared. The bar specimen geometry, optimum for conducting metallographic and x-ray analysis of deformation processes during tensile testing, imposes a severe problem with respect to proper alignment of the gage section axis with the tensile loading axis. To eliminate this problem and the time required to prepare two perpendicular metallographic surfaces on each specimen (many of which are not single crystals), cylindrical tensile specimens complemented with a limited number of bar bend test specimens will be utilized. The latter specimens will supply over-all length change as well as microstructural deformation information.

Four bend test bar specimens, each with two intersecting polished and etched faces, have been prepared, photographed, and characterized and will be encapsulated for irradiation to 10^{19} nvt.

X-ray diffraction specimens having 100 and 110 surfaces are being characterized as to line intensity, line breadth, and lattice parameter. Six of these x-ray specimens together with 15 cylindrical tensile specimens, representing two diffraction and five tensile samples from each of the three carbon levels, will be contained in a single capsule. The three capsules which are required for respective irradiation exposures of 10^{18} , 10^{19} , and 10^{20} nvt (fast) are presently being filled with characterized specimens.

A variety of specimens have been prepared for polycrystalline molybdenum stock containing 100 ppm C and less than 250 ppm total impurities. The metal had previously been annealed at 1200 C. Nine cylindrical stored energy specimens, two inches in length, 3/8-inch OD, with a wall thickness of 0.1 inch, have been prepared. Three capsules, each containing three specimens, will be irradiated to 10^{18} , 10^{19} , and 10^{20} nvt (fast). Capsules for these specimens have not as yet been fabricated.

Three length change cylindrical specimens, 3/8 inch in diameter, having a two-inch nominal over-all length and hemispherical ends, have been prepared for irradiation. Target exposures are again 10^{18} , 10^{19} , and 10^{20} nvt.

Twenty-one microhardness specimens, seven per exposure level, have been prepared for irradiation. These specimens, 3/8-inch diameter by 3/8-inch long, will provide a sufficient supply of specimens for surveying important hardness recovery stages during post-irradiation annealing. A specimen from each exposure level will yield x-ray lattice parameter change values which will be compared with the growth observed in the physical length change specimens previously described.

Ten tensile specimens, 1-1/4-inch gage length and 0.180-inch gage diameter, have been prepared. Five additional specimens are required to permit irradiation of five specimens to each of the three goal exposure levels (10^{18} , 10^{19} , and 10^{20} nvt (fast)). As-rolled foil specimens, stock for optical hot stage deformation studies and electron microscope studies, have been subjected to a one-half hour annealing treatment at 1900 C to achieve grain growth and carbon homogenization. Grains grew to an equiaxed size as large as 0.3 mm in diameter, and the carbides in the high carbon stock 450 ppm became redistributed on a finer scale. Electron microscope studies for these foils showed that many of the carbide particles were present on dislocation lines and at boundaries. Carbide particles in many instances showed a striated structure which was insensitive to tilting of the specimen foil. Only an occasional prismatic loop was detected. Twenty-four annealed foil specimens, six from each of the three groups characterized by a particular carbon impurity level as well as an additional six-foil specimen of annealed 99.99 percent pure molybdenum, were sealed in evacuated capsules and submitted for irradiation.

Recovery and recrystallization of swaged polycrystalline molybdenum rods have been followed by microhardness measurements. An isothermal microhardness recovery curve at 1000 C has been established for 25 percent reduced material. Isochronal recovery of microhardness over the temperature range 25 C to 1200 C is now being studied.

Swaging experiments aimed at introducing various cold work levels in one-half-inch diameter rods have continued. If a wrought microstructure is obtained, rod sections will be shaped into stored energy release specimens and tested in the calorimetric apparatus currently being assembled.

Construction and assembly of a vacuum manifold and apparatus for annealing and quenching specimens is progressing. Reactivation of electrical resistivity apparatus has begun.

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E. CUSTOMER WORK

Radiometallurgy Service

Examination of transversely cracked dingot uranium elements from 105-F showed evidence of "mud-flat" surface cracking on the internal surface of the fuel in the areas where failures occurred. X-ray diffraction on sample B-3A3 indicated a slight amount of preferred orientation of the uranium in this sample (RM 437). Examination of the transversely cracked process tube from 1486 H disclosed that fatigue failure caused the break. Microhardness values on five similar tube failures showed that hardness was greater immediately adjacent to the fractured edges but was relatively the same over the remainder of the sample (RM 438). An overbore element with severe can wall corrosion was examined, and results indicate the corrosion was caused by two damaged supports that failed to provide adequate clearance and coolant flow on the bottom of the element (RM 439). Preliminary examination of the bumpered, enriched I&E element failure indicates that damage imparted during charging was the cause of the rupture (RM 441). X-ray diffraction work on UC and UO₂ for GE-VAL to determine the reaction products between the core material and the cladding revealed there was not enough reaction product present to produce a diffraction pattern.

Metallography Service

A study of the warping of coextruded fuels has led to an examination of the uranium grain structure along the extruded tubes. A definite correlation between asymmetrical grain structure and warping behavior has been found. The uneven cooling of the billet as it contacts various surfaces while being transferred from the soaking pit to the extrusion die appears to be the cause of uneven temperatures, which, during extrusion result in the asymmetrical grain structure and warping of the fuel. Examination by x-ray showed one section of fuel to have a beta-transformed structure instead of an alpha annealed or cold worked structure. Macro etched sections of this rod revealed the front end to be fine grained alpha (cold worked by extrusion) with a thin rim of coarse recrystallized alpha grains. The rim of recrystallized alpha grains grew progressively larger until near the rear of the extrusion the entire uranium section had undergone recrystallization. The ragged grain boundaries of beta-transformed grains appeared gradually along this transition, but these grains lacked the sub-graining so characteristic of beta transformed uranium. One very small area was found in which grains exhibited both ragged boundaries and sub-graining, indicating that this area at least had been heated into the beta phase during the extrusion.

Renewed interest has been shown in uranium-AlSi compound layers and their modes of formation. A study has been renewed and extended in scope to establish under what conditions certain compounds will or will not form. Identification of compound layers will be by metallographic comparison.

Work has continued on an electron micrographic study of thirty heat treated, high purity dingot uranium samples to which known concentrations of iron and silicon have been added. The size and distribution of the finely divided second phase particles are of interest. Fourteen specimens have been polished and vacuum cathodic etched. Negative replicas from these specimens have been processed. To obtain the detail required for positive identification and accurate measurement of the second phase particles it is necessary to make a positive replica from the original plastic negative replica. The additional step allows representation of the etched specimen as it actually appears, with the second phase particles protruding from the matrix. Difficulty has been encountered in the process of dissolving the second plastic replica following the carbon deposition step. Different combinations of mutually insoluble plastics are currently being tried.

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

Samples Processed During the Month

Total samples	580
Replicas	47

Photographs

Micrographs	426
Macrographs	260
Electron Micrographs	71
	<u>757</u>

Negatives Printed	1641
Prints	6387

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N Reactor Charging Machine

1. Modifications

Installation of the 2 to 1 ratio gears in the vertical lift 4-way mitre box has been completed.

Modifications of the "C" elevator magazine and magazine support mechanism were started.

2. Testing

The cross travel drive system life test (100 cycles) was completed. No major problems were encountered.

Additional tests on the transfer arms indicate that the drive shaft and its couplings are developing excessive backlash. The drive shaft is made up of short lengths of shaft and 26 flexible couplings. Each coupling has a small amount of backlash, which when new was not objectionable. A recommendation has been made to replace the flexible couplings with solid sleeve-type couplings.

Loaded magazines have been driven forward and backward by all three forward drive rollers.

The deceleration cams on each end of the rail have been adjusted and tested satisfactorily.

The test of fuel element foot wear using parts of an actual inlet nozzle and prototypical feet was completed. Considerable galling was present when autoclaved feet were run on an autoclaved nozzle. The test has been expanded to investigate this situation more fully.

Fission Product Transient Samples for Phillips

The development of a coextrusion clad tubular element for Phillips Petroleum Company is continuing. Several dummy elements have been extruded with normal uranium-aluminum alloy cores and aluminum cladding. These showed uniform cladding and core thickness except for a 1/2-inch long end effect on each end of the core. Radiographs and sections showed a "shift" of the core ranging from 1/8 to 1/2-inch. In tandem extrusions the "shift" usually increased at the trailing end of the rod. A "shift" of about 1/4 inch can be tolerated, and it appears that this can be accomplished by closer control of extrusion conditions.

Blisters and areas of nonbonding have been present in the extrusions made to date. Billets of improved design for outgassing are being fabricated to correct this problem.

Plutonium-aluminum core material for the irradiation test pieces has been cast and rough machined and samples have been submitted for analysis.

J. J. Cadwell
For Manager, Reactor and Fuels Research
and Development

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PHYSICS AND INSTRUMENT RESEARCH AND DEVELOPMENT OPERATION

MONTHLY REPORT

FEBRUARY 1962

FISSIONABLE MATERIALS - O2 PROGRAM

REACTOR

N-Reactor Exponential Experiments

Measurements were made in the NPR mockup with 150% flooding and six control rods (vertical traverse), and also with full flooding plus one piece of Masonite in each steam vent and no control rods (vertical and horizontal side-to-side traverse).

No results can be reported until consistent values are obtained for the extrapolation distances in the x, y, and z directions.

The final buckling is sensitive to the value used for the extrapolation length in the vertical direction if points near the top of the pile are included. The vertical fit does not fit the three data points near the top of the pile if the horizontal extrapolation length is used at the top of the pile. The deviation of the data at the highest point can be explained because of its different cadmium ratio, but the next two points should fit well if our model is correct. Two different studies are proceeding with the aim of finding a consistent extrapolation length which gives the best fit to all points concerned for several cases.

The lead-cadmium pieces to be used in N-Reactor for auxiliary startup control were received. Their control strength will be measured later in the NPR mockup.

Control Spline Measurements for K-Reactor

The worth of irradiated boron splines used for supplementary control in K-Reactor is to be measured with the K lattice in the PCTR to determine the rate of burnout of the boron. The measurement will be delayed to April or May because the spline samples melted during irradiation. A new set of samples will be irradiated and two will be kept unirradiated for comparison to the old standards.

Data Correlation and Analysis

A Summary of a paper "Neutron Energy Effects on Exponential Pile Analysis" has been submitted to the American Nuclear Society for presentation at the Annual Meeting in June.

A measurement of the diffusion length in the standard pile was made to provide data on a simple case for comparison with the improved analytical methods now being developed. The precision was poor because the background

was too high with only a single source in the pile. Multiple sources are needed to obtain higher accuracy in the measurement.

The standard pile was used to check the effect of neutron leakage through the cadmium shutter. The shutter is not solid because 0.25-inch graphite spacers are used to support the pile. The flux above the shutter was measured for different gap sizes in the shutter. The results indicate that a small amount of leakage does occur (2-5%) with the present 0.25-inch spacers. The leakage does not affect the result if the shapes of the traverses taken with and without the shutter are similar, as they usually are. However, if the experimental conditions are changed (source closer to the shutter, larger gaps, very low buckling, etc.) the shapes of the traverses could be different and the leakage effect would have to be considered.

A special traverse was taken in the C-Pile exponential to look for a small source type oscillation in the vertical traverse similar to the oscillation found in horizontal traverses. A small oscillation was found, but the magnitude ($\sim 1\%$) was too small to provide a reasonable test of the theory.

Two-group theory was applied to exponential analysis to improve the agreement between experiment and theory for data taken near the bottom of the exponential region where the experimental data is always too high. A vertical traverse was taken in the N-Reactor mockup exponential with the sources at $x = \pm a/4$, $y = 0$ and $x = 0$, $y = \pm b/4$ and the counter at $x = a/6$, $y = b/6$. This geometry reduces the harmonic correction below 1% and thus eliminates harmonic effects as a possible source of the discrepancy. With the usual analysis the measured flux at 8" above the shutter was 1.3% too high and the flux at 16" above the shutter was 2.4% too high. Two group theory showed that the usual reflector effect was present, i.e., fast neutrons from the exponential region slow down in the graphite base of the pile and then return to the exponential region to increase the thermal flux near the boundary. The two-group correction reduced the discrepancy to 3% at 8" and 0.2% at 16". Thus most of the difference is removed by the two-group calculations. This will allow more data points to be used in the fits to determine the relaxation length, thus improving the precision of the measurement. Without this correction, data taken near the bottom of the pile usually has to be discarded.

The new value of the copper resonance integral was used to determine the $1/v$ cadmium ratio inferred by the copper cadmium ratio for use in the measurement of p . The old resonance integral gave values of $CR^{1/v}$ which were 5-15% below the CR^{25} for the concentric tube fuel and 2.5-inch solid fuel. This result was suspicious because $CR^{1/v}$ should be 5-10% above CR^{25} in the fuel. The new resonance integral gives $CR^{1/v}$ about 6% above CR^{25} which seems more reasonable. Fortunately the effect of the change is small: about 0.001 to 0.004 in p and negligible in k_{eff} or β/λ .

The error in PCTR measurements of k_{eff} has been examined for two cases where both the mismatch in regular and adjoint fluxes and the change in the inferred mass of poison due to the mismatch had been measured. In both cases Δk_{eff} from the change in the inferred mass of poison to reduce the multiplication factor to one was 0.006. The values of Δk_{eff} calculated from the product of the flux and adjoint mismatches were 0.0038 and 0.0048. The agreement is an encouraging indication of the validity of the error expression derived from two-group perturbation theory, even though the precision of the results used

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here is not sufficient to be a rigorous test of the theory. The change in thermal utilization due to reduced flux in the moderator was calculated for an exponential pile. The flux in a lattice cell in an exponential is not the same as the flux in a lattice cell in an infinite medium since the over-all flux distribution (cosine or exponential) is superimposed on the cell flux distribution. The result is a slight change in the absorption rates which changes f to a value which is usually called f^* . For a small lattice spacing or large pile, the variation of the over-all flux across a lattice cell is so small that f^* is essentially f . However, the exponential pile with 2.5-inch solid fuel in a 14.5625-inch lattice had only five lattice cells across the width of the pile. For this pile the flux at the cell edge (horizontally, along the cosine) is 6.6% below the flux in an infinite medium. This would imply that f^* is about 0.002 above f . However, in the vertical direction the flux at the cell edge is too high by 6% so the effects cancel almost exactly. Thus for a horizontal tube pile no correction is necessary unless B_m^2 is comparable to the geometric buckling of the exponential. For a vertical tube pile the effect is doubled instead of cancelled ($f^* \sim 0.0035$ high and $B_m^2 \sim 6\mu B$ high), so water tank exponentials and vertical graphite assemblies (used to measure anisotropy) may need corrections.

The finite size effect also occurs in p . However, p is an escape probability and the lower flux in the graphite will make p^* lower than p . Thus the effects in p^* and f^* are in opposite directions. Apparently the largest correction will occur in a vertical tube assembly with D_2O moderator since $f = f^* = 1$ and the effect on p is left with no potential cancelling factor.

Lattice Parameters Derived from Critical and Subcritical Reactor Systems

A summary of a paper on the above subject, treating only uranium-graphite lattices, has been submitted for presentation at a special session of the American Nuclear Society meeting in June 1962.

In connection with this study a logical inconsistency in the usual formulation of heterogeneous reactor theory was noted. This inconsistency leads to errors in the values of bucklings derived from exponential experiments which may be large. The error is 44×10^{-6} out of $602 \times 10^{-6} \text{ cm}^{-2}$ in B^2 for one of the D_2O lattices examined so far. The error is not so large for graphite lattices. The inconsistency is of importance for three reasons. The critical parameter B_m^2 , the material buckling, is a function of the shape and size of the system. The resonance escape probability, and hence the conversion ratio, are dependent upon the shape of the system. The effects above have been heretofore falsely attributed to a migration area asymmetry in the usual experiments performed to measure such asymmetry.

The following description of an exponential experiment to derive B_m^2 illustrates the inconsistency and its solution. The thermal flux ϕ is assumed to obey the equation

$$\nabla^2 \phi + B_m^2 \phi = 0,$$

if the flux is measured only at equivalent lattice points. Such measurements of the flux are taken and fit to the functional form (for a horizontal tube pile)

$$\phi(x,y,z) = A \cos\left(\frac{\pi x}{a}\right) \cos\left(\frac{\pi y}{b}\right) e^{-\gamma z}$$

in the asymptotic region of the assembly of extrapolated width a and depth b . From this fit of the flux the buckling is found from the fitted constants a , b , and γ from the equation

$$B_m^2 = \frac{k_{\infty}-1}{M} = \left(\frac{\pi}{a}\right)^2 + \left(\frac{\pi}{b}\right)^2 - \gamma^2$$

Now the inconsistency arises because it can be shown that for the given system the resonance escape probability p occurring in $k_{\infty} = \eta efp$ should be a complicated function of a , b , B_m^2 , and d^2/τ , where d is the lattice spacing and τ the lattice age to resonance energies. It should not be a p which is appropriate to an infinite system. The dependence arises from the fact that the slowing down density at resonance energies at a fuel element at position $x = ld$ and $z = nd$, where l and n are index numbers, is given by the sum

$$q(ld, y, nd, \tau) = \frac{e^{-(\frac{\pi}{b})^2 \tau}}{4\pi\tau} \cos\left(\frac{\pi}{a} ld\right) \cos\left(\frac{\pi}{b} y\right) e^{-\gamma nd} \sum_{\alpha=-\infty}^{\infty} \cos\left(\frac{\pi}{a} \alpha d\right) e^{-\frac{\alpha^2 d^2}{4\tau}} \sum_{\beta=-\infty}^{\infty} e^{\beta y d} e^{-\frac{\beta^2 d^2}{4\tau}}$$

The actual resonance capture probability is proportional to q , which is a function of the parameters listed above. Further, the value of q derived above is, in general, never equal (sometimes larger, sometimes smaller) to its equivalent for a homogeneous system

$$q(x,y,z,\tau) = \frac{1}{(4\pi\tau)^{3/2}} \int_{-\infty}^{\infty} dx' \cos\left(\frac{\pi x'}{a}\right) \int_{-\infty}^{\infty} dy' \cos\left(\frac{\pi y'}{b}\right) \int_{-\infty}^{\infty} dz' e^{-\gamma z'} e^{-\frac{|x-x'|^2}{4\tau}} \\ = \cos\left(\frac{\pi x}{a}\right) \cos\left(\frac{\pi y}{b}\right) e^{-\gamma z} e^{-B_m^2 \tau}$$

The Modified Heavy Gas Model and Neutron Spectra

Computer programs have been written for calculating from the scattering kernel the variation in slowing down power as required by the modified heavy gas model. Results have been obtained for graphite at 293 K and water at 293 K using the kernels generated by the codes, KRYOS-II and KERNEL. Spectra have been obtained from the neutron thermalization code "SPECTRUM" for a graphite-uranium system and a water-uranium system using the KRYOS and KERNEL kernels. Our next step will be to calculate spectra for these same systems from the heavy gas model, and to then compare with the direct results from SPECTRUM. The graphite-uranium system uses the parameters specified by the British for their system on which they performed fast chopper measurements. Comparison of their experimental results with the spectrum generated by SPECTRUM shows much better agreement for 20 C graphite than it is possible to obtain with a "Hurwitz spectrum." Thus, the "Hurwitz spectrum" discrepancy observed by

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Reactor Lattice Physics is largely eliminated when crystal binding effects are taken into account by the appropriate kernel. The uranium water system uses parameters designed to approximate the system currently being run as a test case on RBU. Comparison of spectra generated by RBU and SPECTRUM is planned.

Lattice Parameter Calculational Methods

The two-group analysis of Hellstrand's measurements shows great promise for this particular method of analyzing the effects of self-shielding on effective resonance integrals. The "round-off" problems mentioned last month turned out to be due to a faulty interpretation of Hellstrand's experimental techniques. The present disagreement between theory and experiment could be attributed to variations in the neutron spectrum incident on the rod surface as different size rods are inserted. This possibility, which could shed new light on the Dancoff effect, is being investigated.

Condensed NPR Lattice Calculations

The calculated value of the U-238 cadmium ratio, using program G-2 with resonance self-shielding factors as described previously, is about 15 percent smaller than the measured value. A large part of the discrepancy results from ignoring the self-shielding of the unresolved resonances from about .6 to 30 kev, where a smooth $\sigma_{a,0}$ cross section has been put on the C-Fine nuclear data tape. A smooth variation of the shielded cross sections would be expected, however, and an ad hoc correction based on this behavior is being applied.

Instrumentation

Instrumentation for testing the nuclear purity of the NPR graphite stack was completed, moved to N-reactor, and tested as a system. Tests at N-reactor showed a higher noise level than had been encountered during initial tests at 100-D. Isolation and better grounding alleviated the noise problem. Further BF₃ proportional counter evaluation tests were performed on four received in February. These were to be spares for the system; however, three of the four had unacceptable pulse height distributions. Two complete spare probes are ready for use in addition to the four regular units. Because of flexibility of the cable shrouds, which had been scheduled to be used to push the probes through the graphite, pull-tapes were prepared and will be used to pull the probes through. A transistor phase shift oscillator was designed and tested for use during the tests to audibly signal all personnel as to the start and stop of each counting period.

Bids were reviewed in detail for the fission counter and cable assemblies for the N-reactor sub-critical neutron monitors.

Discussions, to explain testing methods, were held with GE-APED, CE&UO, and N-Project personnel concerning the NPR Fuel Failure Monitor. All questions were resolved and GE-APED agreed to submit a priority list for the testing of particular system components. In addition, calculations made indicate that the Slow-Scan portion of the system should detect fission product concentrations of 2×10^{-8} gm/ml in coolant water at a 45-second decay time. Such a concentration is calculated to provide a 35% signal increase over

the expected background levels. On February 20, the GE-APED-fabricated prototype Slow-Scan portion, except for the differential alarm module, was received for testing. Following complete acceptance testing, the remaining 23 units will be fabricated.

The scintillation alpha detection monitor, recently developed for FPD for checking NPR fuel rods for uranium surface contamination, was modified and recalibrated. Ten micrograms of uranium surface contamination can now be readily detected with a twenty-minute counting period. For the regular inner rod, the calculated detection level, for uniform contamination, is about 1.5×10^{-8} gm/cm².

Technical assistance continued to be provided IPD regarding NPR instrumentation systems. Discussions with Information Systems, Incorporated, in Los Angeles helped to resolve disagreements over the design of the Integrated Data and Temperature Monitoring System. The main problem remaining is the isolation of the flow logger and trend recorders from the flow monitor transducers. A mockup demonstration of the Rod Position Display System was reviewed at the Anaheim plant of Astrodata, Inc. It was demonstrated that the cathode-ray tube could be readily replaced with a newly-purchased spare without loss of accuracy. Two seminars were held for the NPR Project Section to present analyses of the Primary Loop Control System and its components. More such seminars have been requested.

A review of the progress on process tube distortion traverse mechanisms was given to a group of IPD people. The major part of this review has been written up in a memo to be distributed to all interested persons. As a result of this review the customer issued a work order for the construction of two new traverse mechanisms. Our programs for computing displacement from the traverse data have also been reviewed. No errors were found so the program remains unchanged.

Systems Studies

Reactor flux "noise" signals were recorded on magnetic tape at the 105-KW reactor and analyzed to obtain the flux power density spectrum over the range of 0.05 to 100 cycles per second. Signals were taken from the #3 in-core monitor neutron chamber. The standard deviation of the statistical signal is very small ($\approx 0.005\%$ of the DC signal) at the flux levels involved during power level operation, and some difficulty was experienced in obtaining sufficient amplification. Additional recordings are planned to improve the confidence limits which can be placed on the data and to determine the effects of chamber location and flux level.

NPR simulator scoping continued during the month with major emphasis on developing more detailed simulation circuits. The block diagram intended for use in the scope document was revised to emphasize process flow, rather than signal flow, in an effort to present a clearer picture of the intended simulator scope. The design of more detailed kinetics circuits was completed. Several alternate equipment arrangements and operating procedures were investigated to determine possible methods of simplification and cost reduction. Two memoranda were issued. One general meeting with interested IPD personnel was held to discuss the scope information contained in the two memoranda. The possibility of using an increased amount of combination analog-digital equip-

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ment in the simulator was mentioned and is being investigated in more detail.

The NPR secondary loop system is presently being studied on the EASE and GEDA computers. The main part of the system is simulated on the EASE. Only the controllers are simulated on the GEDA.

The forthcoming reactor instrumentation simulation which uses an eleven-node reactor model, will require almost all of the available analog equipment on the EASE and GEDA computers. The method used in the past for delay group simulation requires one integrator per delay group for each reactor node. In order to get the eleven-node model on the two computers, it is necessary to use an analog circuit requiring only one amplifier per node for the delay group simulation; this circuit employs an amplifier input network consisting of resistors and capacitors. The values of resistance and capacitance were calculated analytically. Representative circuits, using both the old and new techniques, were patched on the GEDA; the two circuits exhibited the same transient response. The resistors and capacitors necessary for the new delay group circuits have been ordered.

Application to reactor problems of Dynamic Optimization techniques recently developed at G-E Research Lab was begun. To complete the first step required devising an extension of the Laboratory's methods and the solution of a set of equations hitherto believed to be insoluble. Both of these have been accomplished and are being written up in report form. This was discussed with Research Laboratory personnel in preliminary form.

The reactor application per se is still in the early stages, but the indications are that this technique will point the way towards a better method of synthesizing a control system.

SEPARATIONS

Experiments with Plutonium Solutions

Further critical mass measurements were made with plutonium nitrate solutions in a 14-inch diameter stainless steel sphere. Types of reflectors used in the experiments consisted of spherical concrete casting shells of four and ten inch thicknesses. The results of these experiments are presented in the following table.

The critical volume was decreased from 22.8 l to 21.5 l or ~six percent when the concrete reflector thickness was increased from four to ten inches. It is thus apparent from these results that for nuclear safety applications, a layer of concrete of four inch thickness should be considered as being essentially equivalent to an "infinite" concrete reflector.

These and previous data may be used to estimate the minimum critical mass for a Pu^{239} -water mixture fully reflected with concrete; this value is ~480-490 g Pu^{239} . The minimum water-reflected value for an aqueous plutonium solution is 510 g Pu^{239} . Thus, from the limited data to date, the

(1) Kreusi, F. E., et. al., Critical Mass Studies of Plutonium Solutions, HW-24514 DEL.

CRITICALITY STUDIES WITH PLUTONIUM SOLUTIONS

IN 14-INCH DIAMETER STAINLESS STEEL SPHERE

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

Date	Exp. Number	Reflector Condition	Pu(g/l)	Acid Molarity	Sp. Gr.	H ₂ O(g/l)	Total* NO ₃ (g/l)	H/Pu (Atomic Ratio)	Critical Volume (Liters)	Critical Mass (Kg Pu)
1-31-62	143056	.036" SS + 4" Concrete	32.6	1.20	1.091	946	118	779.1	23.1	0.75
2- 1-62	143057	.072" SS + 4" Concrete	32.7	1.08	1.093	947	120	776.7	23.4	0.77
2- 5-62	143058	4" Concrete + Mockup of Neck	33.0	1.12	1.093	949	116	771.5	22.8	0.75
2- 6-62	143059	4" Concrete	33.8	1.11	1.093	949	123	753.2	22.8	0.77
2- 8-62	143060	.030" Cd + 4" Concrete	33.8	1.11	1.093	949	123	753.2 ~	25.7	0.87
2-16-62	143061	10" Concrete	33.3	1.11	1.096	944	119	760.5	21.5	0.72
2-22-62	143062	10" Concrete	32.8	1.11	1.095	938	-	767.2	21.5	0.71
2-23-62	143063	10" Concrete	31.5	1.09	1.092	935	-	796.2	22.0	0.69
2-26-62	143064	10" Concrete	30.0	1.07	1.087	933	-	834.2	22.9	0.69

* Chemical analyses of plutonium solutions were incomplete at time of this report.

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concrete appears to be a slightly better reflector than water for dilute plutonium solutions.

In one experiment a 0.035-inch thick layer of cadmium was positioned between the vessel wall and the concrete reflector. In this case the critical volume was increased from 22.8 liters to ~25.7 liters. These data are of interest in respect to certain nuclear safety applications in which cadmium sheet has been used on vessels to reduce the effectiveness of neutron reflectors. The observed volume difference implies the 0.03-inch cadmium sheet to have reduced the effectiveness of the concrete reflector by ~0.7 cm, i.e., the reflector savings is ~0.7 cm less.

Data were also obtained for evaluating the effect of the stainless steel of the vessel on the critical mass of the concrete reflected sphere. The results (when compared with previous data) show that as the solution concentration is decreased and the spectrum becomes more thermal, the stainless steel becomes slightly more effective--which was to be expected.

During the month a new critical assembly vessel of 11-inch diameter was received from Tavco, Inc. This vessel is one of those which will be used to extend the criticality studies of reflected units to higher plutonium concentrations.

Miscellaneous Experiments and Calculations for Nuclear Safety Specifications

Fast Fission Parameter δ

In order to specify safe mass limits for slightly enriched uranium fuel elements, it is necessary to determine criticality for lattices of uranium rods in water. A quantity of importance in the calculation of the critical mass is the ratio of fast fissions in U^{238} to fissions in U^{235} (referred to as δ).

The values of δ obtained from a calculational method reported previously were in good agreement with the experimentally determined δ 's for hollow and internally moderated fuel elements. However, the calculated δ 's for the tube-in-tube element were in considerable disagreement with the experimental results.⁽¹⁾ The results of Monte Carlo calculations of the first flight collision probabilities for the tube-in-tube fuel element⁽²⁾ have since become available and may be used in further comparisons (δ depends directly on the first flight collision probability). The results of the Monte Carlo calculation, given in the table following, are in good agreement with the collision probabilities determined by the method of R. Bonalumi⁽³⁾ which were used in the calculation of δ . Since these two methods are in agreement, the results imply there may have been significant systematic errors in the measured values of the tube-in-tube fuel element.

(1) Wood, D. E., K. R. Birney, and E. Z. Block, Lattice Parameter Measurements for a Concentric Tube Fuel Element, HW-57094.

(2) Tiller, R. E., Private Communication.

(3) Bonalumi, R., "Neutrons First Flight Collision Probabilities in Reactor Physics", Energia Nucleare, 8: No. 5, 1961.

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FIRST FLIGHT COLLISION PROBABILITIES FOR TUBE-IN-TUBE ELEMENT

Method	Coolant	Neutrons Born in Inner Fuel		Neutrons Born in Outer Fuel	
		Pc (Inner Tube)	Pc (Outer Tube)	Pc (Inner Tube)	Pc (Outer Tube)
Monte Carlo	H ₂ O	.223	.213	.073	.213
Bonalumi	H ₂ O	.225	.215	.072	.215
Monte Carlo	Air	.248	.131	.089	.219
Bonalumi	Air	.242	.130	.087	.223

Interaction of Subcritical Systems

The extension of the leakage and albedo method to interacting components gave relatively inaccurate results; for some cases as much as 20 percent too reactive.

The critical matrix iteration method did not converge consistently. The matrix was solved analytically with the albedo as a parameter using Crout's scheme. Refinements of k_{∞} to include transverse buckling destroyed the accuracy of results.

The probability function for neutron interaction between two slabs has been programmed. The fraction of neutrons leaving a small area on one face which enter anywhere on the opposite face has been found analytically. This fraction is numerically integrated, assuming a cosine flux, over the emitting face. It is planned to use these calculations as boundary conditions in an HFN - single component calculation. Preliminary results appear encouraging. The probability function for cylindrical interactions is also being studied.

Consulting Services on Nuclear Safety - Criticality HazardsNuclear Safety in HLO

Comments were submitted to the Fuels Fabrication Development Operation concerning the nuclear safety of handling large uranium tubes (six to seven inches in diameter) in the enrichment range of 1-3% U²³⁵(1). Such billets as these may be used in a future fuel element fabrication program. For latticed rods in this enrichment range, minimum critical masses occur for rod diameters of less than one inch. For rods in the six to seven inch diameter range, the critical mass will be roughly five times higher than the minimum critical value. A hole through the center of the rods has the effect of increasing the reactivity and decreasing the critical mass. It is estimated that for a seven-inch billet, the effect of hole size becomes significant when the ID/OD exceeds 0.2; and for an ID/OD of 0.6, reactivity is highest and critical mass is lowest--roughly two to three times the

(1) C. L. Brown to R. S. Kemper, Jr., Private Communication.

minimum critical mass. More definitive calculations will be made when the program is further along.

Nuclear Safety in Transportation

The nuclear safety of three shipments of fissile material were reviewed and approved.^(2,3,4) The first was for the shipment of 1750 Kg of 1.47% U²³⁵ enriched fuel elements, six Pu-Be sources, and 27 Doe and Rae pieces to ORNL via an escorted Mound Laboratories railway car. The second was for the shipment of four 0.95% U²³⁵ enriched NPR fuel elements to Nuclear Metals, Inc., Concord, Mass., via common carrier. The third was for the shipment of 1.47% U²³⁵ enriched fuel elements to ORNL via common carrier.

Nuclear Safety - Training and Education

A talk was presented to 35 engineers and metallurgists of the Fuels Development Operation on the subject of Nuclear Safety Calculations.

Instrumentation

Work continued on the attempt to develop a satisfactory mathematical model of a pulse column used in chemical separations. The results obtained from one of the tentative models have been highly encouraging. It appears that this model can be used to provide results that are well within the limits of normal engineering accuracy. This model will be used to evaluate the constants for twenty experimental runs. The seventeen-inch linear oscilloscope which was recently acquired has proved to be a very valuable asset in the evaluation of the constants of the mathematical model to give the best fit to experimental data. Work on this study will continue in March.

The control rod drive assembly designed for use at the Critical Mass Laboratory was completed in off-site fabrication and was delivered. Complete testing was started.

Resolving time corrections were determined for the three proportional counters used to obtain fuel addition data during critical approach experiments at the Critical Mass Laboratory. The safety instrumentation was also modified to include spring return pushbuttons and potentiometers to allow upscale checking of trip points, and a spring return switch to prevent false scrams during electrometer range switching.

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- (2) Letter from P. F. Gast to F. J. Zelle, Nuclear Safety Approval for Mound Car Shipment, February 6, 1962.
 - (3) Letter from P. F. Gast to F. J. Zelle, Nuclear Safety Approval for the Shipment of Eight 0.95% U²³⁵ Enriched Uranium Tubes, February 20, 1962.
 - (4) Letter from P. F. Gast to F. J. Zelle, Nuclear Safety Approval for Shipping 1.47% U²³⁵ Enriched Uranium by Common Carrier, February 20, 1962.

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Work has started on a scintillation type log n-period meter for the Critical Mass Lab. Components are being assembled and the instrument will be tested in the near future. Operational amplifiers will be used to perform the necessary computing functions. If the instrument functions as planned, it will avoid the troublesome differentiating circuit associated with most period instruments.

NEUTRON CROSS SECTION PROGRAM

Slow Neutron Cross Sections

Some studies have been started on methods of improving measurements of the spontaneous fission half-life of Pu^{240} following publication by others of a value which is significantly different from the presently accepted one.

Quasi-Elastic Neutron Scattering from Water

Analysis of the previously obtained data on the quasi-elastic scattering of slow neutrons from water continued during the month. The results of least-squares fits to the observed scattering do not give directly error estimates on the parameters which describe the scattering process. Methods of obtaining reliable error estimates are being studied.

Inelastic Scattering of Neutrons from Water

The series of measurements on the scattering of neutrons from a water sample of accurately determined thickness has been completed. These measurements were designed to provide scattering cross sections on an absolute scale. Measurements were made to which all previously obtained data can be normalized. The measurements necessary to determine the inelastic scattering of neutrons with initial energies of 0.1 to 0.4 ev from room temperature water are now essentially complete.

A replacement worm gear was obtained for the gear which failed in the drive mechanism of the second arm of the triple-axis spectrometer.

Fast Neutron Cross Sections

The analysis of the neutron total cross section data obtained during December has continued. Results have been obtained on Li, Na, and K. Systematic effects have been discovered in the instability of the zero time of the time-of-flight system which produce errors in cross sections as large as ten percent. In addition, the cross section data for Na show a discrepancy of 100 kev in energy scale around 3 Mev with some previous results. No cause for this discrepancy has been found.

The rebuilt components of the precision beam-locating system for the Van de Graaff have been installed, and the plumbing and accelerator have been re-aligned following the extended accelerator shutdown.

The new RF deflection system has been completed and is ready for installation.

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An improved sample holder for total cross section measurements has been completed.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE

Non-Linear Aspects of Fuel Cycle Analysis

Theoretical physics work on improving nuclear analysis methods for fuel cycles has led to discerning that, for an infinite homogeneous medium, analytic solutions which include the non-linear coupling between the neutron balance equation and the isotope balance equations appear to be obtainable in terms of hyperelliptic functions (a multiple periodic generalization of the doubly periodic elliptic functions developed by Jacobi and Weierstrass in early nineteenth century work on the mechanics of a rotating rigid body). Further study of the general properties of these "fuel-cycle functions," and of methods of fitting them to initial conditions is in progress, in the hope of providing rigorous infinite-medium solutions to guide fuel-cycle optimization work.

Fuel Cycling in Fast-Thermal Reactor Complexes

Work on the fast-thermal reactor complex is continuing in a joint effort with the Laboratories' Programming Operation (R. Hennig). In the present view, the breeding blanket of the fast reactor is treated purely as a plutonium enrichment facility. Irradiated fuel elements from the thermal reactor are re-enriched in the fast reactor blanket, and then reinserted into the thermal reactor. Fuel costs are expected to be lowered because fuel element jacketing costs can be reduced. Two major problems are presently anticipated. One concerns the rapid buildup of Pu-240 in the thermal reactor, and the incomplete burnup of that isotope during the fast blanket re-enrichment process. The other problem concerns the spatially non-uniform buildup of fissile Pu during blanket irradiation which may have an adverse effect on the thermal performance of the fuel element in the thermal reactor.

Parametric Survey of Fast Oxide Breeders

A parametric survey of fast oxide breeders has been started using the multi-group code HFN, and the Yiftah, Okrent, Moldauer (YOM) cross section set. The reactors are oxide-fueled and sodium cooled, and employ stainless steel as structural material. The power ranges extend from hundred MW to several thousand MW. The purpose of the survey is to examine the potential of fast reactors for production purposes of conventional as well as special materials. Another objective of the survey is to relate its results to the program on fast-thermal interacting systems.

Parametric Survey of Fast Ceramic Reactors

The objective of this parametric survey is to compare the potential of various ceramic fuels for compact reactor designs. Some initial results for carbide systems have been obtained for U-235, U-233, and plutonium. These results indicate that both Pu and U-233 may offer considerable reactor weight savings compared to U-235 fueled systems.

PRTR H_x Fuel Elements

An initial determination was made of the relative power generation and physics characteristics of H_x Pu fuel loaded in a central zone of the PRTR. Fuel concentrations ranged from 1.0 to 2.5 w/o H_x plutonium in Mark I Pu-Al fuel elements. From these analyses, it appeared that nineteen H_x plutonium elements containing approximately 1.0 percent plutonium could be loaded in the center of the core without exceeding the power per element limitation.

Code DevelopmentRBU

Problems arising from the molecular binding effects of the water molecule on the hydrogen atom have been partially solved. The approach used was to attribute all the binding effects to the description of a bound hydrogen isotope.

Using an assumed transport cross section behavior, $\sigma_t(E)$ for hydrogen in light water, and the scattering cross section $\sigma_s(E)$ of hydrogen in light water, the following expression was used to obtain the effective scattering-center mass of bound hydrogen (in H₂O):

$$A(E) = \frac{2}{3} \left[1 - \frac{\sigma_t(E)}{\sigma_s(E)} \right]^{-1} \quad (1)$$

Using equation (1), one forces the transport properties of collisions occurring with the bound hydrogen to satisfy known values of $\bar{\mu}_0(E)$ and $D(E)$ where $\bar{\mu}_0(E)$ is the average value of the cosine of the scattering angle in the laboratory system and $D(E)$ is the diffusion coefficient.

The Monte Carlo program produced an estimate for the thermal diffusion coefficient in a light water solution experiment⁽¹⁾ of $D_{THERMAL} = 0.149$. The eigenvalue estimate was $k_{eff} = 0.989$ as compared with an empirical value of $k_{eff} = 1.001$. It now appears that the Monte Carlo, Diffusion combination is approaching the accuracy expected from diffusion theory. Although the information is not directly available, the reactor appears to be supercritical during the Monte Carlo execution.

Using the above technique, however, does not guarantee that the transfer characteristics of collisions with the water molecule are being described correctly, that is, the detailed balance condition may still be violated. This phase of the problem is now being investigated with the possible use of a separate scattering-center mass variation obtained from some of the water kernels now under development. It is now hoped that, with a relatively simple revision of this sort, both the transfer and transport requirements may be met simultaneously.

(1) Henderson, W. B. and E. W. Klingenberg. "Correlation of Many-Energy Bare Reactor Analysis (Program C-Fine) with Clean Critical Experiments Including Beryllium-Oxalloy and Beryllia-Oxalloy Systems," XDC-60-1-211, 1/25/60, Uncl.

Current results indicate that the transfer requirements are not being met. Spectra show a noticeable distortion towards a cool neutron temperature. Development of the proper scattering-mass transfer variation should correct this difficulty.

All other calculations appear to be satisfactory in the Monte Carlo portion of the program. The current version of the Monte Carlo is being flow-charted in detail.

CLERK JAYNE

A program "CLERK JAYNE" has been written for the IBM-7090. This code takes the output from HFN and calculates power distributions, region average cross sections for each isotope, and region average fission and capture reaction rates. The main purpose of the code is to present basic information to the engineer so that he may form his own conversion ratios and related reactor evaluation parameters.

SWAP

The completion of program SWITCH has increased the case-sequencing versatility of the SWAP code package. The program provides input control and executes synthesis of whole reactor models using information calculated in the same case sequence. In a survey of variations in a cell or reactor type the input required is reduced and consists mainly of specification of changes from one case to the next.

A continuing effort has been the comparison of reactivity effects determined by perturbation calculations using SWAP and experimental values obtained during PRTR operation. Of particular interest has been the spatial effects of fuel element changes in the three zone core.

Program SHUSH

Program SHUSH - Spherical Harmonics Using the Standard HFN, has been used to calculate the thermal flux in a simple slab cell in double P-3 and P-7 approximations. The results are very close to those obtained using SHAG⁽¹⁾ on the same case. The neutron conservation equations are satisfied almost exactly by the results of the SHUSH-HFN combination. The SHAG results obey the equations to an accuracy of a few percent. This difference is probably caused by the different recursion relations used in the two codes. SHAG uses the same relations as did the original FN.

The other transport theory approximations were applied to the Milne problem (compute flux as a function of position near the boundary between a critical half-space and a void). These attempts met with little success, and gave little indication of the reasons for this lack of success.

(1) Wenstrup, F. D., "Program SHAG - A Monocenergetic Multi-region Slab Cell Correction Calculation," XDC-60-8-22, June 8, 1960.

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MELEAGER Modification

The plans for an improved version of MELEAGER, the reactor fuel cycle analysis code, are gradually taking shape. Those features which currently are planned for inclusion in MELEAGER are described below.

The average neutron cross sections will be computed from a multi-group analysis, rather than from the ratio of slowing-down power to absorption rate, as is presently done in MELEAGER. This multi-group analysis will be performed in a unit lattice cell, rather than in the single homogeneous medium considered in the present program. In the interests of high computer speed, this cell analysis will probably only include two regions, consisting of fuel and moderator. The composition of the moderator will not change with burnup.

The group structure (i. e., energy limits on the various groups) has not been determined. The average group cross sections will be obtained from a combination of presently available slowing down codes.

Coding has started on an ex-MELEAGER version of the cell code, program ANNE.

External Distribution of Hanford Machine Analyses

The improved double- S_N reformulation of multi-energy transport theory (GE-HAPO program S) was distributed, upon request, to four more external installations doing AEC-sponsored reactor development work (Argonne National Laboratory, Comitato Nazionale per l'Energia Nucleare, Space Technology Laboratories, and Aerojet General Nucleonics). A fifth request for this machine analysis is being processed (for Westinghouse Atomic Power Department).

Effective Resonance Integral of Pu^{240}

The object of this work is to determine the effective resonance integral of Pu^{240} relative to the dilute resonance integral of Pu^{240} . The analysis of the data obtained earlier is continuing.

The part of the analysis now being done is the determination of the reactor sensitivity to the addition of dilute Pu^{240} . In principle this could be obtained from a linear fit through the points in the reactor response curve for which the Pu^{240} concentration was low enough to be essentially dilute. The difficulty is that the reactor response was small for these cases, and the points have a relatively large fractional error. For this reason, it is desirable to make an approximate correction for self shielding on several additional points for which the self shielding is small. These corrected values will then be used along with the dilute points to help determine the best linear fit for addition of dilute Pu^{240} to the system. As long as the correction is small, it does not need to be especially accurate in order to make the points useful in determining the linear fit.

A study has been started of the work which has been done and reported in the literature in the area of calculation of effective resonance integrals for cylindrical rods. The purpose is to determine which calculation, if any, is directly applicable to the problem of obtaining a calculated correction for the points in this experiment with a small amount of self shielding.

Plutonium Recycle Critical Facility

The descriptions (EW-71214) of the Physics experiments which will be conducted at the startup of the Plutonium Recycle Critical Facility have been distributed to those from whom approvals will be solicited. Comments have been received and a few modifications made to the document in agreement with current organizational changes.

Assistance has been provided in setting specifications for storage of fuel elements in the reactor cell.

Studies which pertain to the kinetics of the Critical Facility have given results of two types. A set of tables which relate asymptotic periods to milli-k, to inhours, and to cents have been calculated. This table is preliminary and the final calculations will not be made until the results of kinetics experiments which are being planned for the PRCF are available. The time behavior of the neutron density for various step increases of reactivity and for various operation times before the increase has been studied. A general conclusion is that the period being measured at some time after a step increase in reactivity differs from the value of the asymptotic period by less than 0.5% within 80 seconds ("waiting time") after the change. This is true for changes which result in periods between 15 and 170 seconds if the reactor has been at critical for 120 seconds (buildup time) before the change. This is to be compared with waiting times of 5 times the asymptotic period for the same reactivity changes of the subcritical reactor. For buildup times at critical operation greater than 120 seconds the waiting times become longer.

Neutron Rethermalization

A paper entitled "Neutron Rethermalization in Graphite and Water" has been prepared for inclusion in the proceedings of BNL conference on Neutron Thermalization. This paper is also a submission to Nuclear Science and Engineering. An abstract of the paper is given below.

"Spatial distributions of radioactivities of Cu^{64} and Lu^{177} induced by thermal neutrons near temperature discontinuities in graphite, and graphite and water systems have been analyzed with a multithermal-group diffusion model of the space-energy distribution of the thermal neutrons. Rethermalization cross sections introduced in the model provide a mechanism for the transfer of neutrons between the thermal groups. Values of these rethermalization cross sections for graphite and water have been inferred from the Cu^{64} data. For graphite at a physical temperature of 300°K the cross sections vary from 0.009 to 0.069 cm^{-1} for neutron spectra with characteristic temperatures in the range 144 to 828°K , respectively. For graphite at physical temperatures in the range 144 to 828°K the cross sections vary from 0.016 to 0.044 cm^{-1} respectively for a neutron spectrum with a characteristic temperature of $300 \pm 20^\circ\text{K}$. For water at a physical temperature of $300 \pm 20^\circ\text{K}$ the cross sections vary from 1.25 to 0.83 cm^{-1} for neutron spectra characterized by temperatures in the range 410 to 720°K , respectively. Effective masses of graphite inferred from these cross sections with the heavy gas model agree to within $\pm 20\%$ of the results of Coates and Gayther obtained from differential measurements of the energy distributions

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of neutrons in the graphite of a graphite-uranium lattice. Relaxation lengths also inferred from the experiments agree to within 20% with those calculated from results of theoretical investigations of elastic and one-phonon inelastic scattering of hot neutrons in 300°K graphite reported by Kothari and Khubchandani. Calculations of activities of Lu^{177} made with the multithermal-group model and the rethermalization cross sections obtained in this work agree to within $\pm 10\%$ with the observed activities."

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

Experiments to determine nuclear parameters for 1.8 w/o Pu-Al fuel are being conducted in the tank in the TTR reactor room. The experiments will consist of approach to critical and exponential measurements with lattice spacing of 0.75, 0.85, 0.90, 0.95 and 1.05 inches.

The approaches to critical using the 0.75 and 0.85 inch lattices have been completed. Critical loadings of 563 fuel rods (4009 gms Pu) for the 0.75 inch lattice and 493 fuel rods (3510 gms Pu) for the 0.85 inch lattice were indicated by extrapolation of the inverse multiplication curves. The highest multiplications occurred for loadings of 95% of the critical masses. These results may be modified to some extent when the data are analyzed by the least squares method.

These measurements will help establish nuclear safety specifications for PRTR fuel reprocessing, and will supplement similar measurements to be made later in the PRP-CF.

Spectral Matching by Reactivity Measurements

It has been hypothesized⁽¹⁾ that spectral matching in a test lattice in the PCTR can be attained by adjusting buffer fuel in the PCTR until the reactivity worths of the central cell and a buffer cell, both relative to a void, are equal. The hypothesis has been proven to the extent that a two-group model of a homogeneous core is valid. Experimental data have been obtained to test the hypothesis using an oxide-fueled, graphite lattice. This data was used to predict a k_{∞} for the lattice of $k_{\infty} = 1.258 \pm 0.013$. This value is derived solely from reactivity data as opposed to the cadmium ratio method. An earlier measurement of k_{∞} by the latter method gave $k_{\infty} = 1.256 \pm 0.009$. However, fewer mismatched cases were available for the analysis that led to this latter value than were available for the reactivity-matched case. A report of this work is being written for inclusion in the Nuclear Physics Research Quarterly Report.

Instrumentation and Systems Studies

The scope drawings were reviewed for the collimator and detector shield of the fuel rod and activated wire scanning facility for PRP physics experiments. Detector shielding was increased when it was learned that the 100 mr/hr background levels might be encountered at the detector location.

(1) Originally suggested by F. C. Engesser.

Fabrication, on an emergency basis, was started on the prototype PRTR liquid effluent monitoring system. Following discussions with PRTR and Radiation Protection personnel, it was decided to fabricate one transistorized system to monitor effluent fission product levels from about 10^{-7} $\mu\text{c/cc}$ to 10^{-1} $\mu\text{c/cc}$. Selectable trip-level alarming will be used.

A divider network was designed for the PRCF differential galvanometer to provide 16 factors of 2. This will facilitate the measurement of reactor doubling times.

The feasibility of using a microwave resonant cavity as a sensitive linear displacement transducer for in-reactor metallurgy creep measurements was further investigated. Two transistorized preamplifiers were built for the crystal detectors; these improved the signal-to-noise ratio by about 10 decibels. With a silver-plated transmission cavity, a detectable change of two microinches could be observed at all positions tested within a total range of 0.35 inches. A rough draft of a report on this project was completed.

An over-all review of the existing PRTR control systems has been initiated with the objects of determining the areas where improved system performance would be beneficial to the PRTR operation and whether such improvement is economically feasible. The pressurizer, degasser, and moderator level systems will be studied first. The equipment required for initial tests on these systems is being procured and assembled, and initial testing is planned for the next scheduled shutdown.

The four-transducer, ultrasonic resonance probe assembly for measuring the wall thickness of the PRTR process tubes has been assembled and bench tested. The tests indicate that none of the four transducers have exactly the same sensitivity or response. This will undoubtedly present some problems to the user but it is expected that they can be overcome by generating operational procedures which compensate for the differences between transducers. The tests also indicate that this probe assembly will maintain the transducers in intimate contact with the tube walls during traverses, thus eliminating the acoustic coupling problems experienced earlier with smaller area transducers. The probe is now undergoing final assembly, and when completed, will be delivered to Structural Materials Operation for operational calibration and testing.

PHOENIX FUEL

ARMF-MTR Experiments with Plutonium Fuel

The results of flux measurements during the irradiations in the MTR have been received. The measurements were made with 0.02% Co-Al wires which surround the plutonium samples.

The reactivity effect caused by the heating of the moderator by gamma rays from the irradiated samples has been estimated empirically in the ARMF. The gamma ray contribution to the reactivity amounts to $\sim 1\%$ and $\sim 4\%$ for samples which have been allowed to decay for two weeks and four hours, respectively, before the reactivity measurement.

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PRTR "Phoenix" Fuel Experiment

Another approach is being tried to achieve the spectral hardening necessary for the "Phoenix" fuel experiment. The central rod in the central cell of the PRTR is considered to be 1.8 percent Pu-Al; the absorption cross section of the process tube in the central cell is then varied from its present value to 200 times this value, and at the same time the U-235 enrichment in the remaining rods in the central cell cluster is varied from natural to 30 percent. All calculations are performed by means of the SWAP code.

The results indicate that the required spectral hardening can be obtained with little difficulty; however, analysis of the change in power density in the central rod is presently being undertaken, and this may indicate the need for further central cell rearrangements.

NEUTRON FLUX MONITORS

A dual chamber fission counter was designed and fabricated for measuring breeder-foil sensitivity as a function of in-core exposure time of the foils. The fission counter operated satisfactorily in tests. Methods were established for the preparation and encapsulation of plutonium salts to be used in mass spectrometer analyses. The salts will be in quartz capsules for the irradiations in the reactor. As an alternate method, PuC will be deposited on pyrolytic graphite for exposure. The irradiation experiments were planned and it was decided to perform several preliminary irradiations over a six-week period before starting the principal experiments. The preliminary tests will be used to better define the reactor neutron spectrum to provide better calculation accuracy for Pu-240 burnup. This should reduce the number of samples required for the main experiments. Design was started on the carbon containers that will hold the quartz ampoules for in-core irradiation. The ampoules will contain the plutonium salts and foils. Irradiation test facilities at 100-DR reactor were inspected and plans were formulated for plutonium foil calibrations in the neutron beam.

A rough draft report was completed summarizing a preliminary investigation of possible uses of Nuclear Magnetic Resonance (NMR), Electron Paramagnetic Resonance (EPR), and Microwave Spectroscopy (MS) for in-core neutron flux monitoring at high temperature and flux. Calculations showed the possibility of detecting a minimum flux of about 2.5×10^{12} nv with EPR using the $\text{Cl}^{35}(\text{n}, \gamma)\text{Cl}^{36}$ reaction. If freon-18 (CClF_3) is used, the thermal dissociation is expected to be small enough so that no free Cl or F will be present in the device at 600°C . The minimum flux detection level appears higher for NMR and MS for the measurement methods considered so far; however, other methods which might lower this threshold have been conceived and will be investigated. Experimental tests were delayed again because of slow delivery of ordered equipment.

A study was made of detection of microwave noise due to temperature of the wave guide and degree of ionization of gas in the waveguide. Calculations showed that fluxes in the range of 10^{13} to 10^{15} nv may be detectable by this method using the $\text{B}^{10}(\text{n}, \alpha)$ reaction.

Patent application papers were signed by J. T. Russell for his invention HWIR-1297 - "Nuclear Particle Detectors" (Microwave spectroscopy methods).

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

The stability and linearity of the amplifiers in the transformation section of the multiple parameter eddy current test equipment were checked. The isolation amplifiers which accept the balanced direct current output signals from the phase detectors were found to have a significant direct current zero offset voltage at the outputs and an undesirable nonlinearity. The four isolation amplifiers were rebuilt on a single chassis. New input chopper transformers were installed, replacing some unshielded 60 cycle transformers which were found to be inadequate. Gain of the new circuit is constant to 0.1% over the desired input range of 0 to ± 50 millivolts. The gain is 7.91. The direct current offset at the output is only 0.15 millivolt, thus it is below the noise and resolution limits of the rest of the system. Short term drift is also low.

Several additional changes in the system may yet be needed. The zero set potentiometers of the phase detectors are rather coarse, wire-wound devices, making it difficult to zero within less than 2% of full scale deflection. Fine zero settings are quite unstable and must be repeatedly checked.

The Fuels Preparation Department is preparing test samples for use in evaluating the multiple parameter test for residual can wall thickness measurements of canned fuel elements, and the determination of minimum wall thickness in coextruded fuel elements. A roller-type fuel element rotator and a wheel-type fuel element conveyor were borrowed for use in these evaluations. A motor driven lead screw and coil mount were added to the rotator to permit translation of a test probe along the fuel element. A memorandum was issued which outlined a possible approach and gave tentative schedules over the next nine months for the evaluation of the multiple parameter eddy current test to problems of interest to the FPD Testing Methods Operation.

A memorandum was issued which described 60 cycle eddy current test equipment used in the laboratory to identify untreated, stress relieved, and annealed steel pipe. Description was included for fabricating a device of this type for use in the field.

Heat Transfer Testing

Two radiometers for the emissivity-independent heat transfer testing system have been designed and are being built in the optical shop. The plasma arc torch has been repositioned on the lathe tool carriage to make room for the two radiometers.

Velocity measurements on the lathe spindle during heat transfer tests have shown slight changes in velocity during each test. It is believed these velocity changes are due to excessive end thrust on the spindle bearings which results from expansion of the fuel element during heating. To eliminate the change in thrust, a spring-loaded tailstock for the lathe has been fabricated.

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For initial test it has been decided to use an Ampex variable-time-delay frequency-modulated tape system for the time delay needed in the emissivity-independent system. The Ampex system is expected to be available on loan from the Systems Research Operation, and components necessary for adaptation have been ordered.

X-Ray Stress Analysis

An existing X-ray diffraction stress analysis method has been evaluated, and a new one is being developed for nondestructively determining whether or not welds in steel pipes have been stress relieved.

The existing method makes use of the shift of the (211) ferrite diffraction peak due to stress. A striking difference between data from stress-relieved and non-stress-relieved pipes was noted. However, there was only a small difference in data between pipes that had been stress relieved at 800°F, and at 1325°F for one-half hour. Further analysis of the data will be necessary to determine whether this difference is significant.

The newly developed method makes use of the difference in broadness of X-ray diffraction lines from stress-relieved and non-stress-relieved cold-worked metal surfaces, an effect reported in last month's report. Tests on sand-blasted steel samples heat treated one-half, one, and two hours at 800°, 900°, 1000°, 1175°, and 1325°F showed a sharp change in line broadness between samples heat treated at 1000°F and those treated at 1175°F. The time at temperature had no appreciable effect on the diffraction peak shape. Because of its strong temperature dependence, the X-ray diffraction line shape method appears to be useful for determining if welded steel pipe has reached a proper stress relief temperature. One sample (one-half hour at 1000°F) failed to behave as expected. A slight difference in actual and measured furnace temperature may have been responsible. Several additional samples will be heat treated to check this conclusion.

Zirconium Hydride Detection

Measurements made by Testing Methods, FPD, on Zircaloy-2 pipes containing small, highly hydrided spots, showed that ultrasonic attenuation was greater in the hydrided than in the non-hydrided regions. However, no difference between attenuation in a small Zircaloy-2 plate containing 7000 ppm hydrogen uniformly distributed (microscopic platelets) and one containing no appreciable hydride could be detected. This difference is attributed to large stresses in the vicinity of the hydrided spot in the pipe. The hydrided area expands during hydriding and produces large compressive stresses between it and the surrounding metal. Since the uniformly hydrided plate expands uniformly, the over-all stresses in it should be much smaller. The effect of stress on attenuation in hydrided and non-hydrided Zircaloy-2 was previously predicted qualitatively on the basis of dislocation theory. Further experimental checks on the attenuation-stress effect in hydrided Zircaloy tubing are being made.

Slow neutron absorption measurements, performed at Argonne National Laboratory in response to our request, indicate there is some difference between absorption in Zircaloy samples containing 10 ppm, and those containing 1000 ppm hydrogen (by weight). However, the main usefulness of this method

appears to be above 1000 ppm where the sensitivity would be greater. Further samples are being loaned to Argonne for these studies.

In an effort to detect local hydriding in Zircaloy-2, eddy current testing techniques are being applied to measure the Hall coefficient and resistivity of a sample with a single test probe. Work has continued toward optimizing the test sensitivity to changes in Hall coefficient. With a bismuth sample in a one kilogauss magnetic field, the ratio of voltage-out to voltage-in (measured at the test coil terminals) has been increased to about 3×10^{-3} volt out / volt in. In an effort to increase test stability, the test coil assembly

has been potted into a solid block, and sharply tuned filters placed between each amplifying stage. In addition to better test stability, this also improved the signal (1 kilogauss applied) to null (0 kilogauss applied) ratio to about 600:1.

Six Zircaloy-2 samples, measuring $1/16"$ x $3"$ x $5"$, have been fabricated and hydrided to levels of 0, 1500, 3000, 5000, 7000, and 9000 ppm (by weight) of hydrogen and will be used as sample specimens for subsequent testing.

Methods for Ultrasound Imaging

To reveal more fully the mechanisms by which ultrasound is made visible in water, studies of the literature pertaining to diffraction and refraction of light by ultrasound were undertaken. Several significant points about the phenomena were established: Refraction techniques which use light beam widths that are small compared to ultrasound wave lengths are best suited to ultrasonic frequencies below 0.5 megacycles. Diffraction techniques are necessary at frequencies above about one megacycle because light collimation slits, necessary for refraction studies, are themselves sufficiently narrow to cause light diffraction. Diffraction is the mechanism which makes ultrasound visible by the schlieren method.

Since ultrasound frequencies in the megacycle range are of immediate interest, diffraction techniques must therefore be employed. Quantitative diffraction studies depend on mathematical relationships which will accurately describe the physical situation. Quantitative studies reported in the literature for measuring ultrasonic propagation velocities and distortion due to harmonic buildup in liquids appear reliable. Methods for extending the measurements to solids are briefly discussed, but have not been developed. The application of diffraction methods to the development of Lamb wave group velocities, frequency components, and mode attenuation leading to the better understanding of propagation phenomena may be possible. Mode conversion studies also appear feasible.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Transducer response measurements with the small ball bearing technique were temporarily delayed because of equipment failures during the month. Transparent photographs of beams from three types of transducers were taken using the schlieren system. These were scanned in detail using a commercial densitometer. Comparison of these scans with response data previously obtained

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with the ball bearing method shows some general similarities. However, the densitometer data must be corrected for effects not present in the ball bearing data before direct comparisons can be made. For example, the schlieren method tends to smooth out the detail of the ultrasound beam giving an integrated measure of response. The adjustment of the schlieren data must be obtained by careful control and measurement of the associated variables, which make the method an involved one. Though the schlieren method could prove to be a more absolute measurement of transducer variables, the ball bearing method appears to be more practical at the present time.

A series of tests to determine ultrasonic instrument center frequencies, frequency bandwidths, and voltage linearities was formulated. The application of these tests will help elucidate the instrument response to different Lamb wave modes and transducer frequencies.

Further studies of optical imaging methods for ultrasound show that these may be useful for the following purposes of immediate interest in sheath tube testing: Experimental verification of Lamb wave mode curves for flat plates, curved plates, and tubing. Detailed study of mode generation as a function of frequency, entry angle, and Zircaloy thickness. This may lead to an alternate method for determining system bandwidths. Measurement of Lamb wave group velocity and ultrasound dispersion in Zircaloy.

Initial measurements of group velocities were made and applied to the mathematical expression for approximating the product of frequency times thickness. The results were somewhat erratic and it appears that more precise methods for measuring the ultrasonic path length in the metal will be necessary to obtain the required accuracy.

PHYSICAL RESEARCH - 05 PROGRAM

Mechanism of Graphite Damage

The apparatus for thermal conductivity measurements was completed and tested and found to have too much heat leakage out the sides. Increased insulation did not eliminate the leakage. A method of correcting for the leakage was conceived and is being tried. A specimen of graphite was irradiated 30 hours at 20 microamps for thermal conductivity measurements.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Diffusion experiments at Cape Canaveral, Florida, under the direction of personnel from Atmospheric Physics were suspended on February 5, one month after startup, to accommodate the customer, the U. S. Air Force. Failure of the instrumentation contractor to complete installation of a weather information and display system on schedule caused the delay. A total of 27 experiments were completed in this second series, providing diffusion data during both stable and unstable meteorological conditions for the winter regime. All samples were received at Hanford, assayed, and the results edited for analysis on the IBM 7090 computer. Preparations were completed for the third and last series at this site to start on March 5.

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At Vandenberg Air Force Base, California, the second series of diffusion experiments started on February 5, but experienced a major setback when heavy rains washed out roads and flooded sampler locations. The extensive repairs required before resumption of a full-scale program progressed during the month.

Two successful field tests were conducted at Hanford during the month, both during unstable meteorological conditions. One test was designed to measure the diffusion during a light snowfall. Measurements were made on both the horizontal and vertical sampling grids, downwind of a source located near ground level.

Comparison of Hanford diffusion data with that collected over grassland in the Midwest showed some important differences. The plume centerline exposures showed about the same behavior with distance during unstable conditions. However, during stable atmospheric stratifications, the median maximum centerline exposure over grassland exceeded the Hanford results by a factor of four. The tracer material used at Hanford was zinc sulfide particulate, whereas, the tracer used in the Midwest experiments was sulfur dioxide. The horizontal dispersion was somewhat greater at Hanford, but would not account for the difference. A combination of vertical dispersion and deposition probably account for the apparent enhanced dispersion at Hanford.

In our rain scavenging, an error analysis was started in order to assess the confidence intervals on the scavenging parameters computed from the experimental data. Errors arise from individual sources as follows:

- 1) Variation from drop to drop in the integrated concentration along the path of the raindrop falling through the tracer plume;
- 2) Variation in background level on the raindrop sampler collecting surface due to dry deposition;
- 3) Combined sampling error in determining the relative size distribution of collected tracer on filters, in drops, and from deposition;
- 4) Sampling error in determining the tracer size distribution in the plume;
- 5) Non-Isokinetic sampling error;
- 6) Combined error in determining raindrop size; and
- 7) Uncertainty in determining the plume strength from dosage samplers.

Generally, the standard error from individual sources was 10 percent or less, except for source 7, which has not been evaluated. The scavenging efficiencies determined to date have been considerably smaller than those predicted by Langmuir's theoretical computations, but have been similar to those found by other investigators using the zinc sulfide tracer.

Dosimetry

The apparatus for I-131 detection at the whole body counter was calibrated a third time. The results agreed with the last calibration and indicated a detection limit of about 15 pc rather than the 5 pc first reported.

In an attempt to minimize Zn-65 interference with P-32 bremsstrahlung counting, the counting is now being done at the head of the subject. The degree of interference was found to be much less and to be much more closely correlated with counts made with the big crystal. Measurements of subjects who had eaten Columbia River whitefish indicated their body burdens had dropped to undetectable levels. This is in accord with the expected seasonal decline of the amounts in the fish.

RPO was assisted in testing a modified tape punch-IBM system for the whole body counter. The system worked satisfactorily. They were also assisted in analyzing body burdens of groups of subjects in two contamination incidents. One of these cases involved Co-60. Almost the entire body burden was excreted within a few days. Bioassay measurements of the amount excreted the first day, all in the feces, agreed closely with the reduction in body burden found with the whole body counter.

The positive ion Van de Graaff was shut down most of the month to permit changes in the accelerator and associated equipment. The helium ion source we developed was installed. A new ion-getter vacuum pump was installed. It has performed very well in producing good vacua. Its main purpose is to maintain suitable vacuum conditions over weekends without requiring overtime for servicing with dry ice; it has not yet been used for this purpose. A new demagnetizing system was installed to permit finer control of the analyzing magnet. The control equipment for the accelerator was mounted in cabinets to improve the appearance and safety conditions in the laboratory. While these changes were being made a sensitive foil counter was set up in the laboratory and its background rate under different conditions determined, and a series of long counter intercomparisons was made with radioactive neutron sources.

A PuBe source for precision long counter standardization was accurately compared with our standardizing source for the Livermore laboratory.

We were informed that the National Research Council of Canada has completed two precision long counters. After intercomparison, one of the two will be given to the International Bureau of Weights and Measures at Sevre, France.

It was found that a BF₃ counter made of oxygen-free copper showed much less deterioration of its counting characteristics due to gamma ray irradiation than a brass-wall counter. This finding may greatly reduce difficulties in the use of the precision long counter with certain neutron sources.

As part of a measurement of neutron fluxes at near background levels, a measurement of the neutron background dose rate was made in the 700 Area. The result was 3.0×10^{-8} rad/hr. About two years ago, a value of 7.4×10^{-8} was obtained in the 329 Building, 300 Area. The difference between

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the two is probably real. There was a well-shielded small neutron source in the 329 Building at the time of the measurement that could have provided most of the difference. Also, differences in building construction could have resulted in different cosmic ray absorption and neutron production effects.

A jig was completed and tested that permits reproducible readings of our large tissue-equivalent chambers with a vibrating reed electrometer. Reproducibility depends on accurate relative location of the chamber and electrometer. Reproducibility of 0.2% at about 1 mrad was achieved.

The half-life of Sh-124 obtained by calorimetric measurements is now estimated to be 60.20 days. The result depends on data obtained in two different sets of measurements. The standard deviations of the two sets fail to overlap by about 0.02 day. Although this difference is small, an explanation for it is being sought.

An attempt was made to develop realistic allowable ingestion rates for phosphorus 32.

The Federal Radiation Council Report No. 2 develops RPG's for bone and further develops RPG's for the allowable average daily intake of radium, strontium 90 and strontium 89. No specific guidance is given for phosphorus 32 but the recommendation is made that calculations be done in a similar manner.

The ICRP values are essentially identical to those of the FRC for Sr^{90} , about a factor of 10 higher for Sr^{89} and about a factor of 30 higher than a provisional value calculated for P^{32} after the manner of the FRC.

Definitive values for P^{32} cannot be determined without access to more detailed knowledge of phosphorus metabolism in the body.

The Biology Operation is now planning some experimental work with large animals to determine uptake and distribution of P^{32} in bones, particularly in the rapidly growing bones of foetus and newly born animals.

Radiation Instruments

The measurement method for the experimental field-type airborne luminescent (ZnS) particle monitor was changed to pulse counting from the original dc method. With appropriate discrimination, the indicated detection sensitivity was improved by a factor of four to about 10^{-8} gm/ft³. The resultant sensitivity is only a factor of 5 from that of the tedious, 30-minute integrated counting of fixed filters following chemical treatment. Other advantages of the pulse-count system are the lack of temperature dependence and the use of transistorized circuits.

Circuitry development was completed to permit use of an existing airborne particle detector of the light diffusion type with a multi-channel analyzer for particle size analyses. The circuit accepts positive pulses greater than ten milliseconds wide and converts them, in linear proportionality, to negative pulses of about ten microseconds width for direct multichannel analyzer use.

A number of quartz fibers, 8 to 12 microns in diameter, were fabricated for use in the dosimeter pencils used in the register-indicating miniature pocket dose-meters. Specifications and prints were sent off-site to obtain commer-

cial fabrication of the pencil dosimeters. Actions are under way to expedite procurement of a number of prototype units of this dose-meter and also the dose-meter with the aural alarm.

Experiments were carried out for the use of developed coincidence-counting methods to measure alpha activity on collected four-inch by eight-inch sample filters used throughout the plant. The standard method entails a four- to five-hour wait before counting to permit decay of radon-thoron products entrapped on the filter. Typical filters, after 24 hours of on-line use, can now be checked in two minutes with a detection ability of 2×10^{-12} $\mu\text{c/cc}$ continuous airborne plutonium concentration level. The present HAP0 standard dual-phototube alpha-beta-gamma scintillation filter checking monitors can be converted to the new system by merely obtaining an additional amplifier-scaler. Tests to date, with actual filters, have been satisfactory.

Further design and development work was completed on the circuits for the Atmospheric Physics experimental portable mast system. A complete and detailed timing sequence design was finished. Other circuits completed in design included wind speed storage scalers, gating and interconnection circuits, clock-driven ring commutators, the display circuits, and the parallel-to-serial converter for driving the paper tape punch. In addition, specifications were written for a digital voltmeter, a paper tape punch, a digital printer, and a number of computer circuits. The circuitry used in the commercial anemometers was examined and found to be of inferior design. Originally, it had been proposed to use these anemometers in the system; however, because of the design, a new simplified circuit was developed in replacement. The new circuit was decidedly superior, with an output signal improvement factor of 20.

The wind speed and direction transducers of a number of data stations of the Atmospheric Physics Radiotelemetry System were inspected. In general, the wind speed units were operating correctly; however, all of the wind direction transducers were indicating in error from 20° to 100° and several had defective potentiometers. Maintenance personnel were directed to rectify the problems. In addition, the internal batteries and desiccant material were replaced in the inspected data stations. Tests of the servo units of the wind direction portion revealed bad servo amplifiers, poor gain correspondence, bad choppers, and general deterioration in many cases. Maintenance effort and programming was materially modified.

Construction of the 400-channel analyzer for the positive ion accelerator is under way. Purchase specifications were written for the oscilloscope to be used in the analyzer and most of the other parts and equipment have been ordered.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples as received during the month. In addition, analyses were provided on eight uranium samples for Fuels Preparation Department.

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During the month it was found to be necessary to replace the electron multiplier which is used in the ion detection system of the mass spectrometer. Three new multipliers, RCA 6810A, were installed. Each multiplier had either unsatisfactorily high d.c. current leakage or cracked electrical glass seals. Although the third multiplier had excessive electrical leakage current, its gain was high enough to be used in the ion pulse counting mode of operation of the ion detection system. Efforts are in progress toward putting this counting system into operation.

Work continued on putting the ion-optic test bench into operation. It was found necessary to install magnetic shielding to reduce the leakage field of the ion pump magnet. The magnetic field in the region of the ion optic system was reduced by about an order of magnitude to about 0.4 gauss by suitable shielding. Viton O-rings were successfully used to replace waxed joints on the electrical lead-throughs which had been a source of leaks. Experimentation showed that a non-lubricated Viton O-ring on a teflon shaft was a satisfactory system for introducing rotary motions into the vacuum system at pressures of the order of 10^{-7} mm Hg. The sylphon bellow systems which were used to provide slit motions and which incurred several failures are being replaced by teflon shaft-Viton O-ring systems.

TEST REACTOR OPERATIONS

There were no scheduled experiments performed in the PCTR during the month. Operation of the PCTR was limited to reactor calibrations. There was one unscheduled shutdown caused by electronic failure.

The worth of the moving face was measured from 0 to 24 inches open.

It is planned to replace the U-Al and UO_2 -Pb fuel elements in the driver rod extensions with available enriched U foil. Several fuel worth experiments were completed to provide data needed for this substitution. Additional driver rod assemblies will be made from the elements now used in the driver rod extensions. When more driver rods are available, it will no longer be necessary to use the individual 6 and 8 inch U-Al pieces. More uniform fuel loadings can be made and radiation exposure will be reduced.

Operation of the TTR continued on an intermittent basis. There was one unscheduled shutdown due to electronic failure. The TTR facility was made available five nights during the month to the University of Washington Graduate Center.

A pile modulator was designed and installed in the TTR for use of the University of Washington Graduate Center.

The critical approach facility was used during the month for 1.8 percent Pu-Al rods in a light water system.

CUSTOMER WORK

Weather Forecasting and Meteorological Service

Consultation service was rendered on meteorological and climatological aspects of an air conditioning problem for IPD. Consultation was given PRTR

on humidity measuring techniques for determining heavy water losses. An extreme value analysis of annual crests of the Columbia River was completed and summarized to accompany the river forecasts for 1962 to be issued next month. Meteorological services, viz., weather forecasts, observations, and climatological services were provided to plant operations and management personnel on a routine basis.

Weather Summary

<u>Type of Forecast</u>	<u>Number Made</u>	<u>% Reliability</u>
8-Hour Production	84	86.2
24-Hour General	56	84.2
Special	141	97.2

In spite of a cold wave during the last six days of February, temperatures for the month as a whole averaged a little above normal. Total precipitation was also somewhat above normal. A weather highlight was the occurrence of a thunderstorm on the 22nd.

Instrumentation

System tests were conducted on the alpha-beta-gamma Automatic Conveyor-Type Laundry Monitor with known alpha and beta radionuclide sources attached to coveralls. Minor difficulties of light leaks in two alpha probes and of troubles in the reset circuits become apparent and were corrected. The alpha channel false alarms are now less than one per three hours with 1000 d/m of Pu^{239} being readily detectable in all tests. Sketch-type drawings were started for the complete system with detailed drawings to be prepared after the system has been proven out in field use.

Bids were reviewed concerning instruments for a neutron counting and spectrometry system for Finished Products Chemical Technology, CPD. A discussion was held with FPCT personnel concerning this project and planned ones.

Advice and assistance was provided to Chemical Effluents Technology, HLO, regarding testing and modifications to a magnetometer system and regarding a design of a glow-transfer tube driving circuit.

A special holder for a ten-inch-diameter NaI crystal was designed for Analytical Laboratories, HLO.

Fabrication continued in the 328 Electronics Shop on the Coincident-Count Alpha Air Monitor and the Columbia River Monitor, both for Radiation Protection Operation.

A short study was made regarding a problem of radiation from a radio-frequency generator in the 326 Building metallography lab. The 150 megacycle apparatus, which is used to etch samples, radiates a level of energy many times the FCC limit. General instructions were compiled for fabrication of a suitable shielding enclosure.

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Technical help was provided Physical Metallurgy in planning equipment for the automatic logging of thirty-six temperatures and pressures in experiments related to the effects of radiation on fissionable materials. A study is being made of the desirability of expanding the equipment to be acquired to include the logging of data from other experiments at the same 105-KW facility.

FPD has requested assistance in improving the performance of the induction-heated cap melting control system used in one of their fuel preparation processes. Data obtained from preliminary tests indicated that incorrect controller settings were being used. Performance was improved by changing the controller gain and the integral and derivative time constants to new values obtained by calculations based on these tests. Although system performance is now within stated process limits, an undesirable overshoot occurs on approach to setpoint temperature. Further study of this problem is in progress.

Calibration of micro-displacement readout systems to be used by Physical Metallurgy Operation for in-reactor creep measurements, was temporarily discontinued during February. However, an initial analysis of previous data was received from Operations Research and Synthesis this month. It appears as though the automated IBM-7090 program still contains a few faults, but when operating properly, this program should produce an automatic, single-pass analysis of the transducer calibration data. Reference system precision appears to have been improved to ± 13 microns for the 0 to 0.025 inch range.

Optics

A long-working-distance, high-temperature microscope is being designed for Ceramic Fuels Development Operation. The mechanical components are being fabricated off-site under the control of Tech Shops. Some mirrors and windows have also been ordered. The most important optical component, a rhodium-plated quartz elliptical reflecting objective, will be made in the Optical Shop. Our design effort includes this objective and the mounting and cabinet for several flat mirrors.

Specifications were written for the procurement of a pyrometer system for use in a reactor experiment.

During the four-week period (January 28 through February 25) included in this report, 440 man-hours of shop work were done. The work completed includes:

1. Fabrication of a quartz photometer for Process Control Development, HLO.
2. Repair of two crane periscope heads for Purex.
3. Assistance in the repair of one tank periscope for 200-W Area.
4. Fabrication of an adapter for a measuring microscope for Corrosion and Coatings, HLO.
5. Repair of one electrophoresis cell for Radiological Chemistry.
6. Fabrication of a set of 10 glass bearings for Spare Parts (Stores).
7. Repair of two microscopes for Biological Analysis Operation.

8. Repair of one underwater viewer at 105 KW.
9. Fabrication of components of a radiometer for Physical Measurements.
10. Fabrication of two traversing mechanisms for Irradiation Testing, IPD.
11. Fabrication of 10 ceramic dies to be used in hot extrusion of fuel element components for Fuel Fabrication Development, HLO.

Physical Testing

Service testing work proceeded routinely. A total of 6,619 tests were made on 5,099 items, representing some 31,185 feet of material. The greater part of this material continues to consist of tubular components. Test work included: borescoping; dimensional measurements (micrometric); eddy current; heat transfer (infrared); magnetic particle; mechanical tests (bend, flattening, hardness, and impact); metallography (macro and micro examination, fracture, and field replication); penetrant (fluorescent O.D. and I.D.); radiography (gamma-ray, and X-ray); stress analysis (electric resistance wire strain gages, and X-ray diffraction); surface treatment (alkaline cleaning, conditioning, pickling for autoclaving, steam detergent cleaning, vapor degreasing, and vacu-blasting); and ultrasonic (flaw detection and thickness measurements). Work was done for 31 different HAPC components representing most of the operating departments and service organizations, and other AEC contractors. Advice was given on 49 different occasions on general testing theory and applications.

Testing and treatment of the NPR Zircaloy-2 pressure tubes was completely suspended for the month of February. Work was stopped pending receipt of 27 tubes necessary to complete the order. Advantage was taken of the suspension in NPR activity to work on spare PRTR pressure tubes. All of the remaining spare tubes (34) had the remaining tests completed, were conditioned (6 were vacu-blasted on both the O.D. and I.D. surfaces), and were pickled preparatory to autoclaving upon resumption of NPR testing. Two tubes were found to contain discontinuities larger than the acceptable limits. The defects were not removable because of encroachment on the minimum wall tolerances.

Special dimensional measurements were made on a KER tube to determine eccentricity of the bore with respect to the outside surfaces.

Testing of fuel element sheath tubing proceeded routinely.

Field testing activity consisted of X-ray and gamma-ray work at 1706-KER (Cells 1 and 2), PRTR (Rupture loop and primary coolant system), NPR (instrument leads and test coupons), and steam lines in the 100-H Area. Fluorescent penetrant and radiography of over-bore nozzles proceeded routinely. Additional stress data were obtained at the 105-C reactor to supplement information obtained in the original tests. Ultrasonic thickness measurements were made on vessels in the 309 Building (PRTR), 200-W Area, 105-H reactor, and 100-KW Area.

The major effort in the 300 Area lab continued to be NPR primary loop piping tests. Five feet of fusion-welded pipe were ultrasonically tested by shear and longitudinal methods. The results were graphed for correlations with subsequent fatigue and bend test samples. Additional field replication work

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was done on header outlet nozzles to verify metallurgical structure conditions previously encountered. Nondestructive and destructive evaluation was started on hand-welded pipe specimens. The welds were ultrasonically tested by shear and longitudinal examination, fluorescent penetrant tested, radiographed, and scribed for macro and bend specimens. To aid in the evaluation of side bend samples, fracture testing has been initiated. A bent sample is specially prepared in the vicinity of discontinuity so that it can be preferentially fractured at the discontinuity and microscopically examined to determine the nature of the origin of the discontinuity. Hardness tests and impact specimens were made of a weld in an NPR forging sample. Work was continued on the longitudinal crack length study of the NPR fusion weld pipe samples. A sizeable quantity of radiographs of NPR fusion welded pipe were re-read to evaluate suppliers' practices. A flattening test was applied to a venturi inlet connector with discontinuities detected by a magnetic particle test. Correlations were obtained, indicating the discontinuities could reliably be detected. It was successfully demonstrated with X-ray diffraction techniques, that adequacy of stress relief heat treatments of NPR piping could be evaluated. Reporting of all work completed to date is being done by problem area and will incorporate all of the sectioning data accumulated, macro and micro photographs, and mechanical test results.

A section of pipe was ultrasonically examined in place at 182-K Building prior to and after welding. The test was made to show if any detrimental effects would be caused by welding on pipe filled with water. No discontinuities were found within the limits of the ultrasonic test used.

Six test plate standards, four stainless steel and two inconel, were fabricated and ultrasonically evaluated to show the quality of the plates. The plates will be used off-site in the qualification of suppliers' equipment.

The immersion techniques for ultrasonically detecting cracks in process tubes appears to be feasible over the range of wall thicknesses from about 0.025 through 0.072 inches. Artificially fabricated notches were used to simulate actual cracks. The entry angle for all wall thicknesses was about 15 degrees. Special transducers mounted at this entry angle have been ordered. A preliminary mechanical device for holding and translating these transducers has been designed, and is being built. One of the tubing samples used for exploratory work had surface scratches which gave signals comparable to the fabricated notches. Sensitivity for crack detection will be dependent to a large extent on reactor tubing surface conditions. The two-crystal immersion technique for defect detection in scintered aluminum product (SAP) capsule welds has not proven feasible. As with other techniques tried, there were noise signals which apparently masked indications from real discontinuities. A single crystal contact method using the first Lamb wave modes was also considered. The sensitivity of this test was not satisfactory. Undesirable signal noises were also troublesome. Some of the noise signals could be attributed to small surface irregularities which were exposed after a light surface etch. The surface etch was given prior to evaluating dye-penetrant techniques for discontinuity detection. In view of the difficulties in developing a satisfactory ultrasonic test, a sectioning plan was evolved for the test weld evaluation and further work on the non-destructive evaluation has been temporarily suspended.

ANALOG COMPUTER FACILITY OPERATION

The major analog computer problems considered during the month were:

1. NPR Secondary Loop
2. CPD "C" Column
3. Reactor Instrumentation

The GEDA was 88 percent and the EASE 90 percent operational during the month.

<u>GEDA</u>	<u>EASE</u>	
49	99	Hours up
68	46	Hours scheduled downtime
0	0	Hours unscheduled downtime
<u>35</u>	<u>7</u>	Hours idle
152	152	Hours total

Over-all maintenance of the computers has been good, with the major portion of the scheduled work being done on time. The installation of eight new differential relays has been completed and placed in operation. The capacitors associated with the integrators, and all adjustable resistors associated with the amplifiers on the EASE, were checked and adjusted.

INSTRUMENT EVALUATION

Satisfactory evaluation field testing continued on the prototype scintillation transistorized gamma-background-compensated beta-gamma hand-shoe-clothing monitor at Purex. Nearly three months of testing has been completed with no maintenance or adjustment necessary. Further tests will be made at 100-B. In total, the instrument has now completed five months of satisfactory testing at 100-D and Purex.

After two months of successful use at Redox, the scintillation, combined alpha-beta-gamma hand-shoe-clothing monitor was moved to Purex for further tests. The instrument was attached to 110 VAC mains at Purex, energized, and placed in service without any adjustments being necessary. In the two months' use at Redox, one light leak was noted and repaired and a new reset switch of a preferred type was installed for a total maintenance time of 30 minutes.

Since both the prototype hand and shoe counters have been accepted by Radiation Protection Operation for general plant use, it was suggested that appropriate RPO personnel coordinate the needs of the plant into one large off-site order to secure economical acquisition of desired units. Complete, detailed HAPO drawings pertaining to both instruments are now nearly complete in all respects.

Slight grounding problems, with attendant oscillations suspected as being due to aluminum oxide film buildup, have occurred with a number of Model II Scintrons fabricated in an order of 30 to HAPO design by GE-APED. The sub-chassis, chassis, and cases of the units are aluminum held together by screws. Previous 15 units, all of steel fabrication, have never shown the problems

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and continue to work satisfactorily. The present off-site order of 65 units will be of all-steel construction. It may prove necessary to install a buss-wire connecting all chassis ground points in the 30 aluminum-constructed units. Exhaustive investigations have been carried out with poor grounding being the only item noted.

To alleviate low temperature performance problems with the prototype transistorized portable scintillation alpha "poppy", the battery holder was redesigned to use alkaline batteries instead of the higher-energy-per-unit-volume mercury batteries originally used. The change lowered the operating cost (battery replacement) by a factor of two and permitted instrument use from -10 F to +140 F. Continuous operation battery life is 100 hours. With the changes made, the prototype has performed correctly in all tests.

Calibration and evaluation tests were completed on a series of miniature halogen-quenched GM tubes in cooperation with IPD. One tube, a specially-shielded Amperex 18509, produced a logarithmic response in the circuit used from 10 mr/hr to 100 r/hr using a Co^{60} source; however, with the electron Van de Graaff set at 1.9 Mev, the response was poor and a decidedly different curve was obtained. It appears that considerable testing with different shields is in order before satisfactory performance can be obtained.

Paul F. Gast

Manager
PHYSICS AND INSTRUMENT RESEARCH
AND DEVELOPMENT

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CHEMICAL RESEARCH AND DEVELOPMENT OPERATIONRESEARCH AND ENGINEERINGFISSIONABLE MATERIALS - 02 PROGRAMIRRADIATION PROCESSESReactor Radioisotope Reduction Studies

The sodium silicate addition to a reactor tube was interrupted on January 23 by a reactor shutdown. At that time, the test had been underway for 21 days and the concentrations of As-76, Np-239 and Cr-51 in the effluent water had been reduced by a factor of about two. The P-32 concentrations increased markedly at first but were about 30 percent lower than the control when the test was terminated. Uncertainties in the condition of the tube and elements and only intermittent operation of the silicate addition facilities have prevented meaningful data from being obtained from the test since the reactor start-up following the January 23 shutdown. A better test of the effect of silicate addition will be made on a cleaned tube with new elements and under addition conditions where pH control can be maintained. This test will start when facilities and equipment are available. Laboratory tests of the relative effectiveness of various commercial sodium and potassium silicates have shown very little difference and will simplify the choice of agent for the test.

Laboratory tests of treating reactor process water to a pH of 6.7 have shown that best removal of arsenic parent material is obtained with about 8 ppm alum addition. Although the optimum amount of alum required may be a function of other water treatment variables, and thus differ from this value, it should be possible to determine the optimum value by zeta potential measurements on the floc particles. A full reactor test using zeta potential controlled alum addition was proposed.

The last three samples of effluent water from two reactor tubes charged with nickel clad slugs show an increase in the P-32 concentrations of about 20, and As-76, Zn-65 and Ba-140 - La-140 concentrations about twice that in the total effluent. In addition, concentrations of an isotope with an 0.81 Mev gamma ray have been observed which have been identified as Co-58. This could result from an n,p reaction on stable Ni-58 of the cladding.

Sorption of Nuclides on 100-Area Soil

Research to characterize the adsorption of radiologically significant nuclides by 100-Area soil continued. Iodine-131 was not adsorbed by soil and consequently can be expected to migrate at or near the rate of water movement. The adsorption characteristics of other radionuclides tested were summarized previously.

An estimate of cation migration rates from experimentally determined equilibrium distribution coefficients (K_d 's) can be calculated using an empirical equation developed by Kaufman⁽¹⁾:

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

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$$T'_{OA} = T_{OA} (1 + K_d \frac{b}{f})$$

where: T'_{OA} is the cation travel time from O to A

T_{OA} is the solvent travel time from O to A

K_d is the equilibrium distribution coefficient

b is the soil bulk density, and

f is the soil porosity

The migration rates for Np-237 and Zn-65, based on K_d 's from batch experiments, are 360 and 2 ft/yr, respectively. These values were computed using an assumed solvent linear flow rate of 12,000 ft/yr. This water flow rate is believed to be as much as an order of magnitude greater than actual rates in the field. Based on this same water flow rate, a neptunium migration rate of 290 ft/yr is calculated from soil column breakthrough data.

Soil Permeability of NPR Decontamination Wastes

Soil permeability of the proposed NPR decontamination waste liquor (combined waste, fully neutralized and filtered) was initially about 20 percent that of water. The permeability increased to over 40 percent that of water after seven days of testing. This unusual condition is attributed to the dissolution of soil constituents by the caustic waste.

Ground Water Temperature Studies

Ground water temperature profiles were obtained for all monitoring wells on the project during the past month. Profiles of wells between the river and 200-East Area indicate that some of the thermally warm water discharged to the ground near the Purex plant may be traveling toward the east at considerable depths rather than near the surface of the water table. The average ground water temperature, excluding the effects of the thermally warm water being sent to ground, appears to be between 16 and 17 C. This value tends to be somewhat lower near the Columbia River during high water stage when cold river water recharges riverbank aquifers. Several sites on the project which are not close to the river also show seasonal ground water temperature fluctuations. In these cases the water table is sufficiently close to the surface that air temperature changes affect the water temperature.

Detectable concentrations of beta emitters were reported in Well 699-62-43, located north of Gable Mountain. The isothermal contour map indicates the source of this contamination to be either 100-B or 100-K Area rather than the 200 Areas. Special isotopic analyses will be made on water from this well to further determine the source.

Treatment of NPR Decontamination Wastes

Additional experiments were conducted to compare scavenging from a simulated phosphoric acid waste with scavenging from a "combined" waste. The "combined" waste results from combining single-pass phosphoric acid cleaning process waste

with waste from a three-step recirculating process. The relative merits of MnO_4^- , Fe^{+2} and Ca^{+2} as scavenging agents for Co-60, Ru-106, Sn-113, Zn-65 and Sr-85 from both types of wastes were evaluated.

Relatively poor scavenging of Co-60, Ru-106 and Sn-113 resulted from combining the waste. For example, 200 ppm Fe^{+2} removed more than 99 percent of Co-60 from the phosphoric acid waste but removed only 88 percent from the combined waste. However, more than 95 percent of the Sr-85 and Zn-65 were removed from both wastes by the addition of Fe^{+2} or Ca^{+2} .

Increasing the MnO_4^- concentration in the combined waste from 100 ppm to 400 ppm increased Co-60 removal from 88 to 95 percent. The additional MnO_4^- also improved scavenging of Zn-65 from the combined waste, but in all other instances the additional MnO_4^- either reduced the scavenging efficiency of Fe^{+2} and Ca^{+2} or had no effect.

All five nuclides were effectively scavenged from the phosphoric acid waste by either the Fe^{+2} or Ca^{+2} . Additional MnO_4^- with either Fe^{+2} or Ca^{+2} did not improve scavenging. The scavenging of Ru-106 from the phosphoric acid waste is noteworthy. About 60 percent of the nuclide was removed by neutralization. The Fe^{+2} addition removed 95 percent, but when MnO_4^- and Fe^{+2} were both added, only 60 percent of the Ru-106 was removed.

These results and those reported earlier indicate the two types of NPR decontamination wastes should be processed separately to achieve maximum scavenging. The phosphoric acid waste should be neutralized after the addition of Fe^{+2} and Ca^{+2} and before disposal to a seepage pit. The following percentages of each nuclide would be expected to remain with the solids at the disposal site: Sn-113, 70; Ru-106, 95; Co-60, 99; Zn-65, 99; and Sr-85, >99. Scavenging of radionuclides from the three-step waste should also be adequate according to previous laboratory work.

Airborne Particulates in Reactor Operations

Several additional filters were examined to characterize particulates released to the reactor work area during tube removal operations. Particle size analyses showed few particles larger than 3 μ diameter. Water solubility of the gross beta emitters present was similar to that earlier reported and ranged from 5 to 20 percent. Other than the gamma emitters previously reported, C-14 and P-32 were tentatively identified from beta particle absorption and decay curves. At month's end more intensive and reliable sampling are in progress at 100-D during a current outage.

SEPARATIONS PROCESSES

Disposal to Ground

At the request of Finished Products Chemical Technology Operation, a core drilling program is being undertaken on the 216-Z-9 Recuplex Crib. Soil samples will be obtained from four cores, each 4 to 5 feet long, from the bottom of the crib, and four cores, each about fifty feet deep, around the periphery of the crib. The samples will be analyzed by personnel in the 222-S laboratory to determine the extent of plutonium migration.

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Iodine Sampling in Separation Areas Stacks

Several runs were made to establish the relative efficiency of the 50-foot and 196-foot-level samplers at the Purex stack for I-131. Although very low iodine stack emissions reduced the accuracy of these runs, results from two laboratory analyses indicated that the sample drawn from the 196-foot-level was only slightly lower in I-131 than the sample drawn from the 50-foot-level.

Ion Exchange Contactor Studies

The operability of the simplified semi-continuous jiggle with 50-100 mesh resin has been successfully demonstrated. During a run comprising five 8-hour shifts over a period of eight days, a 5-inch plug movement of resin in the 4-inch column was effected every time it was required without a single instance of failure to move. There was no evidence of resin attrition nor did the pressure required for movement tend to increase in the bellows.

Feed containing 5.5 gm/liter of thorium in 7 M nitric acid was fed to the apparatus at 640 ml/min. A waste loss of 2.5 percent was noted at steady state over the 16 to 18 inches of resin in the absorption section.

Ion Exchange Resin Stress Studies

Measurements have been made on the stress normal to the tube wall by resin beds flowing counter- and co-currently to water flowing through the bed. An unbonded strain gage is used to measure "resin pressure." The 1/2-inch diaphragm of the gage is mounted flush in a four-inch Lucite tube where it senses the combined water and "resin pressure." The water pressure is measured simultaneously and subtracted, thus netting the desired measurement. Initial results indicate that the ratio of the radial stress at the wall to the average vertical stress is in the range of 0.7 to 0.8 for both the 10-20 mesh and the 20-40 mesh resin. This ratio remains essentially constant throughout the height of the bed.

A difference in the coefficients of friction of 10-20 and 20-40 mesh beads was noted; that is, the coefficient of the larger beads was about 0.11 and that of the smaller beads was about 0.06. These observations were made in both co- and counter-current operations.

A significant increase in the coefficient of friction occurs under counter-current conditions when the radial stress exceeds 4 psig. At this stress, degradation of the resin becomes apparent. Further work is needed to determine how this attrition affects the stress data. If resin degradation is not occurring, the average vertical resin stress can be represented by the equation:

$$\sigma_{z/a} = \frac{A D_t}{4 K_a \mu'} \left[\exp \left(- \frac{4 K_a \mu' Z}{D_t} \right) - 1 \right]$$

where: A is the effective resin density, that is, the hydraulic pressure plus the bulk density of the resin bed in the liquid

D_t is the tube diameter

μ' is the coefficient of friction

K_a is the ratio of radial to vertical stress, and

Z is the height of the resin bed.

Photochemical Reduction of Uranyl Nitrate

Bench scale studies on the photochemical reduction of uranyl nitrate to uranium(IV) nitrate were continued. Purex plant UNH solution was used exclusively. With hydrazine as the reductant, the reduction rates were essentially independent of acidity (0.5 to 2.5 M HNO_3) and of temperature (25 to 55 C). This is at variance with previous laboratory results where the reduction rate increased with both acidity and temperature. A pilot plant photochemical reduction unit to produce uranium(IV) nitrate for a plant test is being designed.

Removal of Carbon from Incinerator Off-Gas Scrubber Solution

A caustic scrubbing solution from 234-5 Building was investigated to find means of removing very finely divided carbon. Centrifugation in an open-bowl machine with 5-minute residence time at 2000 G removed only 70 percent of the solids. Heating to 150 F for a few minutes induced flocculation, but the flocs were so fragile that the least disturbance of the liquid caused re-dispersal of the solids. Filtration on Whatman #1 filter paper gave good results, but clear filtrate was obtained only after adding 2 g/l of Celite[®] analytical filter acid (diatomaceous earth) to the scrubber solution.

WASTE MANAGEMENT

Solvent Extraction Removal of Fission Products from Purex IWW

Batch contact studies were made to compare the efficiency of HEDTA, citric acid and tartaric acid in suppressing extraction of unwanted constituents (other than strontium and rare earths) of Purex FTW by D2EHPA-TBP-Shell Spray Base extractants. At equal concentrations and at the same equilibrium pH, citric acid was superior to tartaric acid; both citric acid and tartaric acid were inferior to HEDTA. However, because citric acid is appreciably less expensive than HEDTA, efforts to develop a satisfactory flow-sheet using citric acid as the complexing agent will be continued.

Tracer-level mixer-settler runs (simulating 1A column extraction) were made with feed solutions prepared from synthetic FTW solutions of predicted 1965 composition. Feed solutions contained 0.19 M HEDTA and 0.35 M sodium acetate and were adjusted to pH 4.5. Extractant was unwashed 0.368 M D2EHPA-0.2 M TBP-Shell Spray Base. The mixer-settler was operated with seven extraction stages and at an ℓ/v of 0.5. Extraction of strontium and cerium was greater than 99 percent; extraction of promethium was about 75 percent. Aqueous raffinate pH was 3.3 under steady state conditions. Attempts to reduce the pH change in the aqueous phase by operating at higher ℓ/v 's failed because of flooding in the mixer-settlers.

Solutions of EDTA, DTPA and HEDTA in acetate-buffered systems (initial pH = 4.5) were irradiated in a cobalt-60 gamma field.

The radiolytic destruction of EDTA and HEDTA was first order with respect to complexant concentration. Doses corresponding to 50 percent destruction were 3.9×10^7 R for EDTA and 2.5×10^7 R for HEDTA. The mechanism of DTPA destruction was more complicated but the overall rate of decomposition was similar to that for EDTA and HEDTA.

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Solvent Extraction of Neptunium from Purex FTW

Encouraging results were obtained in preliminary experiments to determine the feasibility of extracting neptunium from Purex FTW with a D2EHPA-TBP-Shell Spray Base solvent. Over 99.5 percent of the neptunium present in a synthetic FTW solution at 0.5 M HNO_3 was extracted in a single batch contact with an equal volume of 0.368 M D2EHPA-0.2 M TBP-Shell Spray Base. Initial neptunium(V) content was 0.005 M. Neptunium was reduced to neptunium(IV) with iron(II); hydrazine was present as holding reductant.

Over 99.9 percent of the extracted neptunium was removed from the organic by contact with 1 M NaOH. Neptunium was precipitated in the aqueous phase along with some iron(III) which is extracted from the FTW. Attempts to strip neptunium with nitric acid solutions (6 - 10 M) and with sodium nitrite solutions (hoping to oxidize neptunium to non-extracted neptunium(V)) were not successful.

Waste Sludge Transfer

Waste sludge core drilling operations, conducted in cooperation with the Chemical Processing Department, continued with two attempts to (1) definitely locate the bottom of the 241-A-103 tank, and (2) obtain a complete sample of the sludge. Results of the two drilling operations include (1) the tank bottom was located about three inches above the "as built" bottom by conductivity probe (see next item, below), and by "feel;" (2) a redesigned steel core bit successfully penetrated practically the entire sludge layer (first bit plugged after drilling three inches during the test in January); (3) the tank appeared to have a crust, with a soft layer underlying the top layer of hard sludge; and (4) the core drill assembly exhibited good mechanical operability. Final evaluation of the core bit is awaiting sample unloading.

Waste Storage Tank Bottom Detection

Ultrasonic depth sounding techniques are being investigated as methods of mapping waste tank sludge and bottom contours. Laboratory tests were made with a one-megacycle transducer to determine the effects of temperature, sludge depth, and particulate matter on the response of the instrument. At this frequency, the metal bottom was not detectable through 1/2-inch of simulated sludge, although a lower frequency signal will probably be effective for greater sludge depths. The presence of air bubbles or particles in the solution causes loss of the return pulse.

Several lower frequency transducers are being procured and additional experiments have been devised using larger tanks prior to an actual system test in a Purex waste tank.

In another, simpler approach to the problem, a resistance measuring device was used in conjunction with a core drill to locate the bottom in a Purex tank. The instrument functioned as anticipated and indicated contact with the bottom about 3-1/2 inches above the location shown on as-built prints.

Waste Packaging

A preliminary scouting evaluation was made of the thermal stress and expansion that could exist in waste packaging containers during processing and storage. For example, longitudinal and diametral thermal expansion or contraction could be as much as one inch in large vessels. Further, depending on design details and waste material properties, longitudinal and hoop stress as high as 300,000 psi could be generated. From these preliminary results, it appears that thermal stress will need careful consideration in vessel design.

Filtration packaging of formaldehyde-treated IWW (FTW) received further study. Solids were precipitated from simulated FTW of predicted 1965 composition by treatment with gaseous ammonia at 90 C. Laboratory bench scale filtration of the resulting precipitate produced a solids volume of one cubic foot per ton uranium processed. Precipitation was made from two-fold concentrated simulated FTW equivalent to 41 gallons of waste per ton U.

Recovery as filter cake was as follows:

<u>Cation</u>	<u>Percent Recovery</u> <u>(weight percent)</u>
Fe	99.78
Al	99.98
Cr	99.19
*Ni	90.60

*Nickel concentration in feed was inadvertently high by a factor of seven.

A similar procedure, in which the ammonia treatment was followed by addition of caustic in an effort to re-dissolve the aluminum, produced a cake representing a solids volume of 2.2 cubic feet per ton uranium processed. A small part of the relatively high solids volume (compared to that obtained by ammonia treatment only) is due to filter aid added to permit filtration, but a major part must be due to increased "bulkiness" of the precipitate. Analytical results from filtrate samples are not yet available, hence the degree to which the aluminum was redissolved is not known.

Larger scale tests are being conducted at present to investigate the feasibility of carrying out the ammonia precipitation of FTW in the Fission Product Packaging Equipment. An initial run confirmed that filter cake volumes of one cubic foot per ton uranium processed would result from projected 1965 FTW. Equipment performance was excellent. Rapid and complete filtration was accomplished with essentially complete filling of the product annulus.

Recovery of Cesium on Clinoptilolite from Purex Alkaline Waste Supernate

An experiment was completed in the 325 Building High Level Cell with a clinoptilolite column two inches in diameter and about four feet long. Cesium-137 was removed from a 1:1 dilution of Purex 103A tank supernatant solution by acid washed, 20-50 mesh mineral. A flow rate of 1.3 ml/min/cm², equivalent to a two-hour residence time, was maintained. The temperature of the column was controlled at 18-20 C.

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Cesium-137 breakthrough was first detected at 14 column volumes of undiluted waste (1 percent breakthrough). Fifty percent breakthrough was reached at 18 column volumes which is within experimental error of that predicted from laboratory data. This is equivalent to removing 98 percent of the cesium from the first 17 column volumes passed through the bed. The 50 percent breakthrough point of 18 column volumes for clinoptilolite can be compared with 13 column volumes for Duolite C-3 resin with untreated waste and 24 column volumes for Duolite C-3 with a waste containing excess caustic (2 M NaOH).

The cesium was eluted with 5 M NH_4NO_3 at 50-60 C, and more than 99 percent was removed in a three-column volume fraction. A two-column volume fraction had a Cs-137 concentration over 15 times that of the influent concentration. The eluate Cs-137 concentration dropped to one percent of the feed concentration after six column volumes. No unusual difficulties were encountered in filling or emptying the column with clinoptilolite.

Recovery of Cesium on Clinoptilolite from Purex Formaldehyde-Treated Waste

Several laboratory runs can be summarized as follows:

1. Increasing the sodium content of the synthetic waste from 0.6 to 1.2 M decreased the column capacity by 40 percent.
2. Increasing the iron concentration of the waste from 0.17 to 0.34 M decreased the column capacity by 20 percent.
3. Increasing the temperature from 25 C to 60 C decreased column capacity 40 percent, and decreasing the temperature to 16 C increased column capacity by six percent.
4. Contacting the synthetic waste with TBP in kerosene had a negligible effect on column operation or cesium capacity.
5. Dilution of the synthetic FTW (54 gal/ton) by factors of 3 and 10 increased the column capacity for Cs-137 by 21 percent and 36 percent, respectively.
6. Equilibrium experiments with Cs-137, Ce-144, total strontium and Ru-106 indicate that only Cs-137 is removed from the acid waste.
7. Exploratory experiments indicate that the removal of strontium from Purex formaldehyde-treated waste with clinoptilolite is not practical for process application. Simulated formaldehyde-treated waste solutions were prepared with citric acid to complex interfering ions such as ferric ion. The solutions were neutralized to various pH levels up to pH 4.7 with lithium hydroxide. The lithium hydroxide was used since clinoptilolite had previously exhibited high trace strontium uptake in the presence of lithium ion. The strontium K_d 's were less than ten in all cases.

Cesium Solvent Extraction

New information was obtained during the month on the hot-cell experiments (mentioned briefly last month), on the effect of pH on cesium extraction, on the relative extractability of other fission products and cations, and on the

effectiveness (as cesium extractants) of nitrated diphenylamine compounds containing fewer nitro groups than DPA (dipicrylamine).

Two hot-cell experiments were performed. In one, tartrate and caustic were added to 100 ml of full-level Purex 1WW, resulting in a final volume of 200 ml, and this solution was contacted with an equal volume of 0.01 M DPA in nitrobenzene. The cesium extraction coefficient (E_a^O) was about 2.75. The single-stage decontamination factor from sodium was 322. The decontamination factors from fission products ranged from 250, for ruthenium, to 1140 for zirconium-niobium. Three successive extractions removed a calculated 98 percent of the initial cesium. The extracted cesium was readily stripped with 0.5 M or 1 M HNO_3 with back-extraction coefficients (E_a^O) of 140 and 80, respectively. Continuously contacting the organic phase with the 1WW for 190 hours resulted in a decrease in E_a^O of only about 10 percent and in no visible breakdown of the organic or formation of solids. Similar results were obtained with 103A tank supernate, except that the cesium extraction coefficient was somewhat higher (3.6) and the decontamination from other fission products somewhat poorer (possibly due to lack of tartrate complexing or to analytical uncertainties because of the high cesium to fission product ratio).

Variation in pH, at least in the range 5.7 to 11.5, had little effect on the extraction of cesium from tartrate-complexed, simulated FTW, the extraction coefficients ranging only from 2.8 to 3.6. Extraction coefficients of a series of other elements were measured, and all found to be much smaller than those of cesium. On the basis of these results, an improved analytical method for cesium has been developed using DPA extraction. Precision and accuracy of the method are excellent, and there is far less interference from other fission-product activities than is the case with the currently used tetraphenylboron method.

A series of additional compounds, identical to dipicrylamine except for the number of nitro groups, were synthesized and tested as cesium extractants. Cesium extraction was very low ($E_a^O \sim 0.01$) unless one ring contained three nitro groups and the other ring contained at least one. Location on the second ring had little effect. Thus, extraction coefficients for cesium (from 1 M NaOH) into 0.01 M solutions of 2,2', 4,6- tetranitrodiphenylamine, 2,3', 4,6- tetranitrodiphenylamine, and 2,4,4', 6- tetranitrodiphenylamine were 3.47, 3.85 and 4.0, respectively, compared to a value of about 24 for dipicrylamine itself. Substitution of a trifluoromethyl group for one nitro in the 2' position was less effective and decreased E_a^O to 0.17.

Burning experiments were performed with a liter of nitrobenzene-dipicrylamine solution. The solution was difficult to ignite, due to its high flash point, and burned with a dark red-orange flame giving off a thick, black sooty flame. Addition of dry dipicrylamine or of a slurry of dipicrylamine in nitrobenzene had no effect.

TRANSURANIC ELEMENT AND FISSION PRODUCT RECOVERY

Oxalate and Sulfate Precipitation Studies

Additional B-Cell and laboratory experiments have been performed in support of current Purex strontium recovery and future possible cerium and promethium recovery. Direct oxalate precipitation of rare earths from 1WW (or FTW) may be

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desirable either as a first step in rare earth recovery or to reduce the centrifuge heat load in strontium processing. B-Cell experiments show that near quantitative precipitation is achieved provided sufficient oxalate is employed, and that the minimum amount required varied with different batches of 1WW-- presumably because of different iron contents. From 2 to 3.3 volumes of 1 M $\text{H}_2\text{C}_2\text{O}_4$ were required per volume of plant 1WW.

Tracer-level laboratory experiments on double sulfate precipitation revealed, surprisingly, that addition of hydrogen peroxide strongly inhibited the precipitation of rare earths. Additional work is planned to explore and elucidate this phenomena.

Technetium Sulfide Preparation

Technetium sulfide (Tc_2S_7) is a convenient starting material for the preparation of most technetium compounds. The technetium recovered in the A-Cell anion exchange runs was accordingly converted to this form. The procedure which was finally developed follows: The hot-cell product was in eight liters of 8 M HNO_3 . This was evaporated to half a liter (14.4 M HNO_3) before technetium began to distill. An equal volume of hydrochloric acid (12 M) was added, heated to boiling, and formaldehyde added drop-wise to destroy the remaining nitrate. Ammonium persulfate (5 g/l) was added and the solution heated to boiling to oxidize all of the technetium to the heptavalent state. After cooling to room temperature, the solution was saturated with hydrogen sulfide to precipitate the technetium. The as-precipitated sulfide was amorphous and could not be filtered; however, heating to 82 C for 30 minutes converted it to a form which was easily filtered on a coarse-grade glass filter. A portion was converted to TcO_2 for use as an analytical standard.

Bulk Fission Product Packaging

Binding Agents - Studies of binding agents for fission product powders continued. All binding agents are co-deposited with the fission product precipitate. Strontium peroxide filter cake has been reduced to a dense fused product by calcination with 10 percent ceramic binder, 3 percent sodium hydroxide, or 2 percent lithium fluoride. (Percentages are weight percent of total weight of uncalcined filter cake.) About 10 percent of the ceramic-bonded cake adhered to the filter. The sodium-fluxed cake reacted slowly with water or moist air. The lithium-fluxed cake was a rock-like product that was unreactive toward water or air. It dissolved slowly in dilute acid with the evolution of an as yet unidentified gas (possibly oxygen remaining as a result of incomplete calcination). The strontium density of the fused cake was about 2.5 g/cc., approximately ten-fold higher than the unfluxed filter cake.

Ten weight percent lithium fluoride fluxed strontium carbonate as a fluid melt at about 900 C. Lithium fluoride does not flux ferric oxide (calcined waste).

Low melting glasses numbers 1990, 7570, 7572 and 7575 by Corning Glass Works were evaluated as binding agents for ceric oxide. Numbers 1990 and 7572 showed some promise in this regard, giving firm but friable and porous cakes when 20 percent of the powdered glass was mixed with cerous oxalate before calcining at 800 C.

Filtration Canister - In continued studies of the prototype loading station, the 18-inch-long canister was shortened by one inch to bring the bottom of the can more fully within the field of the induction heating coil, thus allowing a more uniform temperature over the lower half of the can. The maximum outside temperature difference from can mid-length to can bottom is now about 50 C versus 300 C before the change.

Three runs in the loading station have been attempted this month. In one run the strontium peroxide slurry bypassed the filter, apparently because of improper seating of the gasket sealing between the slurry inlet and the filtrate outlet. The other two runs were taken to completion. In the last run, the filter stick had a small leak, evidenced by initial cloudiness of the filtrate. This filter stick had previously been through two complete pyrolysis cycles.

Remote Welding - Remote welding closure studies continued with welding of 5-1/2 inch 304L caps in the inert-gas tungsten-electrode welding station. The welding conditions were: (1) amperage, 100 to 200 amps; (2) welding speed, 6.7 inches per minute; (3) arc length, 0.025 inches; (4) argon flow, 14 cubic feet per hour; (5) automatic start and tail slope of amperage; and (6) cap to can gap of 0.030-inch diametral clearance. Seven of the samples had flush face joints and three had 1/8-inch raised face joints. No leaks were detected in the raised face joint welds. Six flush face joints welded at less than 200 amperes had cracks in the joint, and the leak rate was higher than could be measured by helium mass spectrometer. A flush joint welded at 200 amps had a leak rate lower than could be detected. Various raised face joints will be evaluated.

Modifications of the torch positioning mechanism were completed. Positioning of the torch is greatly improved, but further development will be required to get a torch positioner suitable for remote operation.

ANALYTICAL AND INSTRUMENTAL CHEMISTRY

Emission Spectrographic Determination of Metal Impurities in Water

A direct emission spectrographic method was developed for metallic impurities in water samples. The method showed an average standard deviation of ± 4.6 percent for Fe, Ni, Cr, Zr, Al and Cu, and an average spike recovery of 95 percent for Fe, Ni, Cr, Zr, Mn, Al, Cu and Co in water samples. Previously, water samples from the 100-K recirculating loops had required time-consuming and costly concentration to permit spectrographic analysis. A systematic evaluation of the spectrographic parameters affecting sensitivity was made. By choosing the most sensitive conditions found, a sensitivity was reached which permitted direct analysis. Those parameters include use of the Jasco spectrograph, emulsion type 103-0, a high voltage spark with a 150 volt primary input, and the graphite residue electrode method.

Spectrographic Analysis of Impurities in Zirconium

The carrier distillation method used for analyzing impurities in uranium has been adapted for determination of impurities in zirconium.

Zirconium oxide prepared by oxidizing at 900 C turnings or filings from an etched metal surface is mixed with gallium oxide and cobalt oxides as carrier and

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and internal standard, respectively. A 100 mg sample is compacted into an 8 mm crater electrode. The sample is DC arced at about 11 amp, during which the impurities are distilled. The resulting plate is relatively free of the zirconium spectrum, only the most intense lines showing faintly. Working curves were developed for Fe, Al, Cu and Cr. The aluminum curve showed greatest variation due perhaps to the refractory nature of the oxide. A single analysis of pure zirconium metal showed very good agreement with off-site analysis.

Strain Measurement Near Pipe Welds

Research is completed on the feasibility of using an X-ray spectrometer to determine lattice parameters of iron crystals as a measure of strain near pipe welds. In addition to showing that laboratory-type equipment has adequate sensitivity for this measurement, the work led to a simple and rapid test of the efficacy of heat treatment in relieving welding stresses. The test evolved from the following observations:

- (1) Vapor or sand blasting a steel specimen developed severe residual strain in the outer 5-10 mil surface layer. This type of strain is easily detected qualitatively from the appearance of a diffraction scan past a single iron line in the back reflection region. A single, very broad but weak peak indicates considerable strain in the specimen surface. Complete resolution of the $K\alpha_1$ - $K\alpha_2$ doublet together with narrower, more intense peaks indicates absence of strain.
- (2) Heat treatment of sand-blasted pipe at 800 F did not relieve either the surface strain or the deeper strains produced in the steel adjacent to the welds. Heat treatment at 1325 F, however, relieved both types of strain.

For examining sand-blasted, welded, and heat-treated pipe, then, the test involves a two-degree scan to traverse the diffraction peak, and visual examination of the scan to determine the degree of resolution of the peaks of interest.

EQUIPMENT AND MATERIALS

Hydraulic Equipment

With increased radioactivity and hence heat generating levels in process solution, means must be found to transfer boiling solutions under emergency conditions in the event of loss of coolant.

Preliminary tests with water cooling jacket around the intake of a 3 GPM jet was found to permit transfer of hotter than usual solutions. Additional cooling of the influent stream by admission of cold water (resulting in further temperature drop of about 12 C) resulted in steady state transfer at a rate of about 28 percent of the nominal cold water transfer rate with 11 percent dilution of the boiling water being transferred. Additional tests will be conducted with a more efficient jet rated at 75 GPM under conditions simulating plant operations. In these tests, the aim is to establish or derive formulae whereby the maximum gain in transfer efficiency can be predicted with the minimum of dilution. Fabrication of the test stand is essentially complete.

Simultaneous operation in transfer of boiling water by means of a thermal siphon or steam operated "air-lift" was completely satisfactory in a bench-scale study. The equipment is being mocked up to prototype size.

A Hastelloy C canned motor pump with 50 mil cans has been in operation without incident pumping hexone for approximately 544 hours. Operation has been very smooth and quiet. Inspection will be made for determination of wear of critical parts after either 1500 or 2000 hours of operation.

An electrically driven ball valve failed after 64,800 cycles of operation when the motor cam-control spring separated. The unit was re-installed and again failed after a total of 336,960 cycles when the drive gears wore severely. Trouble was traced to improper length of the mounting spacers creating a deformation of the gear casing which allowed only partial meshing of the faces of two of the drive gears. This reduction or inadequate contact area caused accelerated wear and finally failure. Prior to failure and throughout operation, the test was extremely smooth, reliable and leak-proof.

Development of the waste transfer diverter (replacing diversion box technique in future installations) was completed. With modifications, the system exhibited flooding rates above 200 GPM--about two-fold higher than required.

Mechanical Equipment

To meet the demand for a small, remote process connector, an end cap, originally designed as a closure for fission product containers, has been redesigned. The new design permits the unit to be used as a process connector. The design incorporated desirable features such as all stainless steel, semi-remote operation, minute leakage rates and simplicity of design. Development and testing are now in progress.

Experience has shown that Teflon[®] is not suitable for gasket material in high radiation fields unless it is completely restrained. Tests indicate that linear polyethylene may be substituted for the Teflon gasket used in the B-plant process connector if the gasket is used for low temperature chemical service. The linear polyethylene gasket was subjected to service in a two inch B-plant connector at 11.5 psig at room temperature. No leaks were observed with water as the pressurizing medium.

Sonic Energy Application

A study has been instituted to investigate the foam destruction properties of sonic energy generated by a stem-jet whistle, with initial application being aimed at destruction of foam in uranium dissolvers. Tests indicate the output of an experimental whistle to be about 147 db. at the nozzle orifice. Experiments performed in an insulated sound attenuation vessel have demonstrated that this whistle, when directed at a detergent foam, can control the foam produced by air-sparger flow rates of up to 55 cfm. Fabrication of two additional types of stem-jet whistles has been completed as had the fabrication of a mock-up dissolver tower base testing assembly.

Corrosion of Titanium in Fluoride-Containing Purex 1WW

Long-term, single-period exposures of titanium to the vapor from synthetic Purex 1WW containing 0.02 M HF have been completed. Test samples were removed from the environment at one-week intervals. Longest exposure was 40 days. Relatively

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high corrosion rates, about 0.5 mil/mo, were observed for all samples. The mode of attack suggests corrosion by condensate from refluxing solution rather than true vapor phase corrosion.

Corrosion of 304-L in Concentrated Uranyl Nitrate-Aluminum Nitrate Solutions

In connection with a proposed Redox back-cycle system involving the 2DFS and 3DFS streams, corrosion rates for 304-L stainless steel in acid deficient uranyl nitrate-aluminum nitrate solutions are being obtained. In a single 10-day test without solution changes, samples exposed in the liquid, vapor and interface of boiling 3.1 M UNH - 2.1 M $\text{Al}(\text{NO}_3)_3$ - 0.2 M acid deficient solution corroded at rates of 1.4, 0.07 and 0.43 mils/mo, respectively. Attack on liquid and interface samples was intergranular; corrosion rates for longer exposures would probably be significantly higher.

Non-Metallic Materials

In pursuit of materials of construction for molten salt plutonium processing, vapor deposition of tungsten metal is being studied. Tungsten has been successfully deposited onto a copper sheet by reducing tungsten hexafluoride in hydrogen at approximately 650 C. The deposit varies from dense and smooth to dendritic. The density of the smooth deposit is approximately 95 percent of theoretical.

Two attempts to coat a 10 percent TiO_2 - 90 percent MgO ceramic part have been only partly successful as the tungsten does not adhere well.

PROCESS CONTROL DEVELOPMENT

Neutron Multiplication Monitor

The neutron detection instruments of the experimental multiplication monitor were relocated in the Purex N-Cell to a position providing higher neutron flux. Multiplication factors up to 2.5, corresponding to k_{eff} of 0.6, are expected in the new location adjacent to the N-7 receiver tank.

The diffused junction silicon neutron counter is correctly indicating changes in neutron flux but due to its low geometry and low conversion efficiency, in relation to the BF_3 tube, the resulting counting rate is only about one percent of that of the BF_3 tube. Various methods of increasing conversion efficiency of solid state devices have been reported in the literature and are being evaluated.

Data are being recorded to enable calibration of the system in terms of neutron multiplication. Some difficulty has been encountered in obtaining sufficient information concerning concentration and liquid levels in the N-7 tank, since these data are not normally recorded for operational purposes.

C-Column Instrumentation

Several modifications were made to the experimental C-Column to further improve its versatility as a test facility. New heat exchangers were installed on the LCX and LCF streams to extend the operating range of the facility and improve temperature response characteristics. A semi-conductor strain gauge is being tested on the column as a column density transducer. The instrument is also used to display cyclic pressure variations.

After over two years of continuous operation, a mechanical failure necessitated disassembly of the mid-column photometer. Failure was caused by a leakage of solutions splashed on the instrument into the space containing the filter slide device. Subsequent corrosion of the sliding metal parts by these solutions caused the filter slide to bind in the phototube block and stop its normal operation. The photometer was repaired and will soon be returned to service.

C-Column Data Processing

The calibration data obtained for the LCX turbine flow meter was fit by least squares with a linear model which relates the throughput flow to the digital volt-meter reading. The data was found to have a standard deviation around the model of 12.8 ml/min for flows between 500 and 2500 ml/min. Compared with the deviation for the presently used Fischer-Porter transmitting rotameter of 40 ml/min, over the same range, it can be seen that use of the turbine meter substantially improves the accuracy of the LCX flow measurement.

An improved technique was developed for fitting a mathematical model to the unequilibrated aqueous and organic uranium concentration as a function of position in the column. Presently, unequilibrated organic uranium concentration, U_o , is determined from measurements of aqueous uranium concentration, U_a , and analysis of an equilibrated sample for U_a^* , U_o^* , and E , where these values represent equilibrated aqueous and organic uranium concentration and the fraction of organic phase in the sample. The U_a and U_o can then be fitted to a model by a least squares technique. However, the correlation between U_a and U_o complicates this approach excessively. The complication is eliminated if U_a and W are fitted where $W = E U_o^* + (1-E) U_a^*$, which represent completely independent measurements at a given column height.

In making the fit to U_a and W , the deviations of the observations of these quantities from the model predictions will be weighed inversely according to the errors squared associated with each observation. The C-Column data reduction code was modified to calculate, print, and punch out W and the weights to be used with U_a and W .

REACTOR DEVELOPMENT - O4 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Co-Deposition of Uranium and Plutonium Dioxides - Attention has once again been turned to the problem of co-depositing PuO_2 and UO_2 from molten chloride salt solutions of uranyl and plutonium(III-IV) chlorides. In previous work, mixed oxides were produced by cathodic deposition from a KCl-NaCl melt at 750 C, following a moist air sparge. These conditions, however, gave poor rare earth decontamination and yielded dendritic products with undesirably high G/U ratios.

This past month it was attempted to electrolytically co-deposit UO_2 and PuO_2 at 600 C from chlorine-sparged melts selected from the LiCl-KCl or LiCl-NaCl-KCl systems. The melts were made up to contain 4-6.5 w/o uranium and enough plutonium to give a U/Pu ratio of either 1.0 or 4.4. It was hoped that the lower temperature and relatively high plutonium concentration would both work toward putting more plutonium into the deposit.

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As expected, the chlorine sparge allowed the formation of a dense, crystalline product. However, plutonium separation factors across the deposition step were about the same (45-110) as in earlier experiments employing hundred-fold lower plutonium concentrations at 600 or 750 C. For example, a very low U/Pu ratio of unity gave a deposit containing only two percent plutonium. Thus, this approach does not appear highly promising.

Co-Deposition of Uranium and Thorium Dioxides - Electrolytic deposition of UO_2 from molten chloride salt solutions of UO_2Cl_2 and $ThCl_4$ is now under study. The first series of experiments, using KCl-LiCl melts at selected temperatures between 500 and 700 C, yielded the following conclusions:

1. Solid solutions of at least 14 w/o ThO_2 in UO_2 can be formed as polycrystalline deposits by this technique.
2. The oxide deposit can be enriched in thorium, as compared to the melt, by factors at least as high as 2 or 2.5. The degree of enrichment decreases with increasing $ThCl_4$ concentration in the melt.

The incorporation of ThO_2 in the oxide deposit probably occurs by reaction of $ThCl_4$ with the UO_2 as it is laid down on the cathode, by the reaction

By study of this reaction and system, it is hoped to develop information useful in connection with co-deposition of UO_2 and PuO_2 .

Uranium Chemistry in Molten Chloride Salt Solutions - Recent work on the chemistry of uranium in molten chloride salt solutions has been directed toward the kinetics of the oxidation of uranium(IV) with air. A current interest in precipitating PuO_2 from chloride melts by sparging with oxygen-chlorine mixtures has now prompted an examination of the rate of oxidation of uranium(IV) with oxygen-chlorine. Comparison of the two reactions was carried out at 600 C using 40-gram melt samples (LiCl-KCl, 2.6 LiCl-KCl, and 2.6 LiCl-NaCl) containing about 8 w/o uranium(VI) and 0.5 w/o uranium(IV). Sparge rates were about 150 ml/min. The results may be summarized as follows:

1. With an air sparge, the oxidation proved to be zero order in uranium(IV) concentration and to proceed at a rate of about 0.9 w/o per hour. Thus, an air sparge caused complete oxidation in one-half hour in each of the three melts.
2. With a 30 percent oxygen-70 percent chlorine sparge, slower oxidation was observed, especially below 0.3 w/o uranium(IV). In this case, complete oxidation of the uranium(IV) required more than 70 minutes in 2.6 LiCl-KCl, more than 100 minutes in 2.6 LiCl-NaCl, and more than 120 minutes in LiCl-KCl.

Engineering Process Development - Pilot plant electrolysis studies with a 60 w/o KCl - 40 w/o LiCl melt were initiated this month. Melt preparation for the first run consisted of (1) drying the uranium-free salt by chlorination, (2) dissolution of U_3O_8 feed with chlorine, (3) drying the resulting molten salt solution

of uranyl chloride with chlorine, and (4) pre-electrolysis for four hours as a final drying step. The pre-electrolysis cathodes were then used as anodes for a product electrolysis carried out at 0.75 reference volts and 310 amperes. Forty pounds of UO_2 with an O/U ratio of 2.0266 were produced in 24 hours at a current efficiency of 47 percent.

The product crystal size was smaller than previously observed for that with a good O/U ratio. This was tentatively attributed to an interfering impurity in the melt since similar crystal characteristics had been observed in laboratory electrolysis in melts containing 0.2 w/o thorium. Analysis of the melt indicated the presence of 0.35 w/o iron introduced with the U_3O_8 feed.

The comparatively high electrolysis voltage was also postulated as a possible cause for the crystal characteristics.

A submerged gas lift recirculator was tested during dissolution and drying. The lift was a 14-inch-high, 3/4-inch-diameter quartz tube connected at the bottom to a gas inlet tube and at the top to a support rod. The tube bottom was cut at an angle to provide an unobstructed flow path. Qualitative results indicated that the lift kept the greater part of 50 pounds of undissolved U_3O_8 and inert sludge circulating with only three liters per minute chlorine flow.

UO_2 Pulverizing, Classifying and Transfer - A broad program to define the problems of handling electrolytic UO_2 continued with studies of two pulverizers and a hydraulic classification/transfer system. Both two-inch and four-inch milling cutter pulverizers operated smoothly and produced a suitable product (i.e., without excessive fines). However, the two-inch cutter required force-feeding to "bite" into the relatively large chunks of UO_2 . Surprisingly good separation of various mesh particles was obtained by flowing water at varying rates through a bed of pulverized UO_2 thus transferring increments of solids out of the bed. Further studies to define the precision of this method of classification will be made.

RADIOACTIVE RESIDUE FIXATION

Ruthenium Studies

The ruthenium tetroxide decomposition study was completed with measurement of the effects of nitric acid vapor, nitric oxide (NO_2), and reaction-chamber surface area. Nitric acid vapor at the concentration expected in the off-gases from the radiant-heat spray calciner (equivalent to 2 M in condensate) caused a marked decrease in extent of decomposition--from over 90 percent RuO_4 decomposition at 150 C in the absence of nitric acid to less than 25 percent when nitric acid was present in the gas stream. Addition of NO_2 , on the other hand, increased the extent of decomposition and resulted in irreversible "absorption" on a silica gel bed. An increase in surface area (by packing the reaction chamber with glass beads) had very little effect. A detailed report on this project is being prepared.

In-Cell Calciner

Assembly of the in-cell spray calciner equipment in the cold mock-up area of the 325-A Building is very nearly complete, and testing has begun. The major uncertainty in the design of the unit was the adequacy of a heated section only five feet long (dictated by head room in the cell). A test run, with simulated Purex FTW,

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shows that it is. The run was made without filter blowback or use of the vibrator in order to see where solid build-up might occur. Operation was very smooth and the inside of the column was quite clean after the run. A few particles adhered to the top cover but the area around the water-cooled nozzle was completely clear, and there was no hint of stalactite formation. The off-gas line was also completely clean, with no visible trace of powder. The solid (powder) product had an excellent appearance and contained only 0.033 weight percent nitrate.

The tests revealed some deficiencies in the resistance heating of the column due to excessive power loss in the leads and inductive coupling between the two Variacs used to supply the upper and lower sections of the column. Thus, it was not possible to maintain the 800 C design temperature when calcining feed at the design rate of one gallon per hour, and the above runs were made with temperatures of about 700 and 530 C, respectively, in the upper and lower sections. More insulation will be added to the insulating cans and, if that fails, a larger transformer (from the demonstration unit) will be employed.

Fabrication of the cart for the melt pot is proceeding well at Minor Construction and should be completed within two weeks. The electrostatic precipitator, also constructed by Minor Construction, has had to be returned twice to correct fabrication errors but should be ready for installation by March 1. Other items being constructed or completed include an in-cell wash sink (for exterior decontamination of melt pots), some additional melt pots, and the hooking up of certain drain, scrub, chemical addition, and air lines. Detailed design of step plugs and bus bars for carrying electrical power through the cell wall, has been completed and construction started.

Decontamination of A-Cell is proceeding and is anticipated to be complete well in advance of in-cell installation of the calciner. The clinoptilolite was pushed out of Column 5 with concentrated ANN with very little difficulty, and flushing the tanks and columns reduced the radiation level in the cell to 60 rads, before exterior spraying was commenced. A simple water wash decontaminated the glass jugs (used for collecting strontium product) sufficiently that they could be disposed of in unshielded cardboard cartons.

Kinetic Studies of Ion Exchange Materials

The cesium loading kinetics of lithium, sodium, potassium, hydrogen, calcium, and barium-based clinoptilolite were determined by a "shallow bed" technique in the film and particle diffusion regions. Results showed lithium exchange for cesium in solution to be the most rapid reaction in both diffusion regions. Exchange of hydrogen on the clinoptilolite for cesium in solution is among the slower exchange reactions. The lithium-cesium exchange, for example, proceeded at a rate of 0.00192 meq Cs/ml of bed/second, compared to 0.00064 meq Cs/ml of bed/second for the hydrogen-cesium exchange.

Hydrogen ions, having a liquid diffusion coefficient approximately three times as large as that of lithium cations, should unload faster than the lithium-based clinoptilolite in the film diffusion region of cesium concentration if liquid diffusion is the limiting factor. However, the assumption ignores the specificity or intensity of the clinoptilolite cation replacement series. Film diffusion rates alone would be rate determining only if the zeolite were non-selective.

The exchange sites on clinoptilolite prefer cesium as it is the least hydrated cation of the replacement series. Lithium, the most hydrated cation of the series, is most easily and readily replaced by cesium.

Low- and Intermediate-Level Wastes

Trace strontium and cesium exchange with several zirconium compounds in dilute ammonium ion solutions was investigated. The compounds included zirconium phosphate, zirconium tungstate, zirconium molybdate, and hydrous zirconium oxide. Zirconium tungstate and zirconium phosphate showed the highest strontium and cesium uptake, respectively. The uptake of trace strontium by zirconium phosphate in 0.02 M ammonium ion is about an order of magnitude less than that of a sulfonated polystyrene resin. The trace cesium uptake by zirconium phosphate is somewhat less than that of clinoptilolite.

Micro Pilot Plant Run 22 was terminated when Cs-137 in the effluent from the Amberlite IR-120 column exceeded its MPC_w after the treatment of 10,800 column volumes of waste. The activated carbon column used to remove residual organic prior to passage of the waste through the ion exchange column was regenerated at atmospheric pressure with steam superheated to about 250 F. MPP Run 23 was performed to evaluate the ability of the steam regenerated activated carbon column to remove residual organic from steam-stripped Purex Tank Farm condensate. This carbon column was effective in removing organic during the entire run treating 1800 liters of waste; however, the pressure drop across the column steadily increased during the run while the fresh carbon column was able to treat over 1500 liters before the pressure drop started to increase. It is believed that the regeneration operation created enough fines to impair the hydraulic characteristics of the bed.

The Amberlite IR-120 column was regenerated with 8 column volumes of 4 M NH₄OH followed by a water rinse, and then this was repeated using 8 column volumes of 8 M NaOH and a water rinse. Preliminary measurements indicate that the ammonium hydroxide was effective in removing 10 to 15 percent of the column activity, and the sodium hydroxide removed the additional 85 to 90 percent.

BIOLOGY AND MEDICINE - O6 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Geology and Hydrology

Field visits made to key basalt exposures in the plant area and environs together with stratigraphic knowledge, obtained in part from U.S. Geological Survey personnel, resulted in the determination that the flows exposed in Gable Mountain overlie those exposed in Umtanum Ridge and, therefore, constitute the youngest and stratigraphically highest part of the basalt series. These flows have not been systematically studied or described because they evidently center beneath the Hanford Works where they are but poorly exposed. Determination of the stratigraphic position of the Gable Mountain basalt flows will help resolve the sequence along the Columbia River, the geologic history, detailed structure, stratigraphic relationships with other formations and, accordingly, the behavior of ground waters in the upper part of the basalt series.

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Discussions with National Bureau of Standards personnel at the Boulder, Colorado, laboratory revealed that the computational methods they employ in mapping radio-wave propagation characteristics of the ionosphere can be applied directly in the mapping of ground water potential patterns. These methods are needed at Hanford to enable use of the irregularly spaced well observations in describing potential distributions over the project and in predicting ground water behavior. A request was made to obtain a copy of the subroutine program from the N.B.S. Laboratory.

Four plastic piezometers were installed at different depths below the water table in Well 699-37-82A to measure the ground water potential head at the various depths. The tubes were embedded in coarse sand and each sand layer was separated by a 30-foot grouted section. Potential measurements made three days after installation of the piezometers indicated head variations of from 0.2 to 1 foot with the minimum head occurring about 140 feet below the level of the water table. These measurements correlate qualitatively with readings previously obtained with the vertical flow meter which showed increasing downward flow velocities with depth in this well. Such measurements of potentials with depth are needed for three-dimensional analog simulation of the ground water flow regime beneath the project.

Field Apparatus - Development and Application

The An/ASQ-3 fluxgate magnetometer is undergoing laboratory testing. A malfunction noted in the detector circuit was due to a faulty 2000 cycle filter; a new filter is being obtained from the U.S. Geological Survey. All other components operated satisfactorily. The magnetometer will be used to obtain data needed for complete evaluation of the airborne survey of the underground basalt.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition in Conduits

Several deposition experiments were completed in a 60-foot long, one-half-inch diameter vertical tube. Zinc sulfide powder with a mean size of about 2μ was used in these shakedown tests. At the higher velocities used (Reynolds No. about 8000), 91 percent of the powder deposited in the conduit. Comparison with estimates based on derived correlations awaits measurements of particle size distributions; qualitatively, the deposition is about that expected.

Columbia River Sediment Studies

As part of a radioisotope depletion study, measurements were made on the relative amounts of radioisotopes on particulates (as separated by millipore filtration) in reactor effluent water (REW) and Columbia River water. These studies were aimed at discovering any differences which might have resulted from use of high alum and other changes in the reactor process water treatment procedures. The data are tabulated below, together with the results of measurements made in 1959:

<u>Radioisotope</u>	<u>Percent Particulate</u>					
	<u>1961</u>					
	<u>Sample 1</u>		<u>Sample 2</u>		<u>1959</u>	
	<u>REW</u>	<u>River</u>	<u>REW</u>	<u>River</u>	<u>REW</u>	<u>River</u>
As-76	1.8	6.4	5.8	4.7	2.6	3.7
Cu-64	7.8	22.3	6.6	16.3	26.0	15.0
Na-24	0.96	1.2	0.71	1.7	0.7	0.6
Cr-51	0.99	3.4	1.1	3.3	3.4	3.3
Np-239	4.0	7.1	4.0	8.0	0.95	1.4

The amounts present on particulate material do not differ greatly except in the case of Np-239 which may indicate an increase in the rate of loss of this isotope to the bottom sediments.

Counting Techniques

The use of dimethyl-POPOP (or DMP) in liquid scintillation counting has increased the sensitivity for tritium counting by 17 percent when compared with the standard method, using POPOP. Greater improvement may result by further varying the concentrations and components of the scintillation liquid. The advantage of DMP over POPOP appears to lie mainly in its approximately four times greater solubility in toluene.

Manager
Chemical Research and Development

WH Reas:cf

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BIOLOGY OPERATION

A. ORGANIZATION AND PERSONNEL

R. H. Schiffman was appointed Acting Manager of Aquatic Biology effective February 1, 1962.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Columnaris

Repeat experiments on virulence of a radiation-induced antibiotic resistant mutant columnaris strain showed no difference in virulence compared to a wild type organism. Both cichlids and salmon were used in these studies.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Strontium

The absorption of Sr^{85} and Ca^{45} from the perfused small intestine of the rat was affected by 10 mM sodium citrate when the solution was devoid of alkaline earths. When 5 mM calcium were included, citrate increased the absorption of both Sr^{85} and Ca^{45} . A solution containing 5 mM citrate reduced Sr^{85} and Ca^{45} absorption.

Lymph collected during the perfusion of the intestine with Sr^{85} and Ca^{45} contained less activity than the plasma, but had the same ratio of Sr^{85} to Ca^{45} .

Comparison between the average freezing point depression of blood plasma of control and experimental trout fed $0.5 \mu\text{c Sr}^{90}\text{-Y}^{90}/\text{g}$ food ad libitum for four weeks showed no statistical difference.

One miniature pig fed 3.1 mc of Sr^{90} per day for 118 days was killed when semi-moribund. The body burden will be determined on the basis of radioanalysis of the skeleton which is now in the process. In vivo radiation dose measurements were made by the Radiological Development and Calibrations Operation at various locations in the body that were external to bone using Victoreen pocket dosimeters. Readings varied from below the detection limits up to 7,600 mrad per day. Most of the variation was attributed to locations of the measurement sites relative to their proximity to bone or to various portions of the gastrointestinal tract which contained unabsorbed $\text{Sr}^{90}\text{-Y}^{90}$. A summary of the measurements made in the body cavities is given in Table 1. The high reading at the location of the ovaries is related to their close proximity to the colon. Measurements made immediately adjacent to some of the major bones ranged from 3 to 6.7 rads/day.

Table 1. Dose Rate Measured by Victoreen Pocket Dosimeters at Various Locations in a Swine Fed 3.1 mc/Day

<u>Location</u>	<u>Dose Rate (mrad/day)</u>
Ovary	2,000
Heart	800
Left Uterine Horn	100
Right Uterine Horn	600
Vagina	1,000
Antero-lateral abdominal cavity	600 - 900
Postero-lateral abdominal cavity	100 - 200

Widespread hemorrhages were evident throughout the body of this swine, a probable consequence of the very low platelet count observed previous to death. Aplasia of the bone marrow was evident grossly. The gastrointestinal tract was grossly normal, and contained no free blood as observed in an animal of this group recently autopsied. More severe lesions would probably have been evident if the animal had not been euthanized, since it would probably have lived 24 to 48 hours longer. (Since all the cellular blood elements were very low for some time prior to death, serum from this animal was preserved for future use in attempting to demonstrate factors that will stimulate erythrocyte, leukocyte, or platelet production. Theoretically, the serum should be high in such factors since the bone marrow was not able to meet body needs due to the radiation assault.)

A litter of four offspring (6 weeks old) were weaned from a sow that had received 625 uc/day for approximately 6 months including the entire gestation period. Two of the offspring were of normal size and weight, and two of slightly sub-normal size and weight. The average skeletal radiation dose for these animals was estimated to be 5 to 8 rads/day at birth and 30 to 40 rads/day at weaning. (A second litter of six pigs at this feeding level are showing normal growth and will be weaned during March.)

Neptunium

Preliminary results suggest that about 0.1 per cent of a dose of neptunium nitrate administered intragastrically to rats was absorbed. Approximately 0.03 per cent of the dose appeared in the liver.

Liver and kidney samples taken 30 minutes after the intracardial injection of Np^{237} were studied for the intracellular distribution of Np^{237} . A marked difference was observed between the distribution in the kidney and liver. The liver contained 26, 42, 20, and 12 per cent of the total activity in the nuclear, mitochondrial, microsomal, and soluble fractions. In the kidney, the same fractions contained 22, 17, 20, and 41 per cent of the total activity.

In continuation of the study on the acute chemical toxicity of Np^{237} , one ewe which received 6 mg/kg (4.1 uc) died four days post-injection. Post-mortem

findings included icteric appearance throughout the body, severe fatty degeneration of the liver, and hemorrhages in the ileum and the mesentery. Two young ewes which received 3 mg/kg (2.1 μ c) survived, but showed impaired liver function as measured by I^{131} Rose Bengal blood clearance.

Plutonium

Rats were partially hepatectomized and injected with Pu citrate at various times after surgery. Hepatectomy reduced the liver deposition by about 50 per cent. Bone deposition was affected only slightly.

Intracellular distribution studies of Pu in liver and kidney showed that Pu was distributed somewhat differently than was Np. The nuclear, mitochondrial, microsomal, and soluble fractions of liver contained 35, 26, 15, and 24 per cent, respectively, of the total activity while the same fractions in kidney contained 43, 11, 11, and 35 per cent of the activity.

Four miniature gilts were intradermally injected with 0.0016 μ c to 5 μ c of $Pu^{239}(NO_3)_4$. This makes a total of 10 swine bearing 595 injected sites that are being held for lifetime observation. One 5 μ c site injected 540 days previously still contains about 2.4 μ c of Pu^{239} . This site has a hard, brown, horny-like scab. This site is unusual in that all other 5 μ c sites on this animal contain less than 0.015 μ c of Pu^{239} . A one μ c site which is adjacent to the 5 μ c site described above contains about 0.2 μ c of Pu^{239} and has a similar type scab except somewhat smaller in size. It appears that the degrees of tissue reaction and quantity of Pu^{239} retained is a function of depth of deposition.

Two dogs were exposed to $Pu^{239}O_2$ aerosols as a repeat of an earlier experiment in which was observed a relatively rapid translocation from the lung of plutonium, deposited as particles with a MMD of 0.22 μ .

A dog exposed about two and one-half years ago to $Pu^{239}O_2$ aerosols is showing high respiratory rates of 80 to 90 per minute, slight cyanosis. Another dog in the same experiment showed CNS disturbance for the second time. Two dogs have shown these temporary symptoms. However, whether this is related to the plutonium treatment is not known.

Bone Marrow

Control rats irradiated with 950 r died within 10 days. However, animals injected with bone marrow within three hours after irradiation were alive and healthy after 30 days.

Cellular Biology

Deuterium effects in fermentation are different from effects on respiration. D_2O grown cells show an increased endogenous fermentation whereas exogenous fermentation with glucose is inhibited up to 75 per cent. Furthermore, the rate of fermentation does not fully recover when D_2O grown cells were

transferred to ordinary water.

A repeat experiment on the effect of D₂O on fish confirmed that 50 per cent D₂O is lethal. At 40 per cent fish showed signs of stress but survived for seven days. No effects were seen at D₂O concentrations below 40 per cent D₂O.

Radiation-induced changes in constituents of rat's liver include decreases in the potassium content of the soluble fraction in 12-hour X-irradiated regenerating liver.

Plants

Preliminary data obtained on coleus plants indicate that crossover of Cs¹³⁷ between vascular bundles occurred. This is contrary to the polar hypothesis which states that ions travel in a vascular bundle without crossover.

Rattlesnake Springs

Moisture in the upper 3 dm of soil from Hanford sagebrush and greasewood communities decreased during the period February-August at an average rate of 0.4 liters/square meter/day. Maximum rates of decrease, ranging from 1.3 to 1.8 liters/square meter/day, occurred between the middle of March and middle of April which is when the deciduous greasewood defoliate.

Columbia River Limnology

Sampling of plankton populations continued at Vantage, Priest Rapids, and downstream from 100-F Area.

Fallout

Radiiodine concentrations in Colorado deer thyroids decreased steadily from 2×10^{-4} $\mu\text{c/g}$ wet weight at the beginning of the month to less than 1×10^{-4} $\mu\text{c/g}$ wet weight at the month's end.

Project Chariot

Plankton counts were completed on about 200 samples from Arctic ponds and lakes.

Space

Different life cycle stages of the flour beetle, Tribolium confusum, were maintained two weeks in a field essentially void of magnetic forces at the General Electric MSVD laboratories at Philadelphia. Subsequently the cultures were maintained under optimum environmental conditions.

Increased lethality and sterility occurred in beetles that were exposed as young larvae. The normal sex ratio was not altered. No changes were observed among those treated during pupal and adult female stages.

General

Frank Swanberg was a guest of Biology's Internal Emitter Committee meeting. Research related to the deposition of radioactive materials in human beings was discussed.

Manager
BIOLOGY OPERATION

HA Kornberg:es

C. Lectures

a. Papers Presented at Society Meetings

Bustad, L. K. Radiation biology at Hanford. American Society of Mechanical Engineers, Richland, Wash. February 15, 1962.

b. Off-Site and Local Seminars

Bustad, L. K. Radiation biology at Hanford. Greater Seattle Assoc. of Science Teachers, University of Washington, Seattle, Wash. February 9, 1962.

Rickard, W. H. Jr. Some ecological aspects of fallout in regard to livestock operations. Montana Nutrition Conference, Bozeman, Montana. February 12, 1962.

Uyeki, E. M. Radiation therapy-protective agents. In-Service Institute on Radiation Biology, University of Washington, Seattle, Washington. February 14, 1962.

Bustad, L. K. Somatic effects of chronic low-level radiation. Radiology Department, University of Washington, Seattle, Washington. February 20, 1962.

c. Biology Laboratory Seminars

Van't Hof, J. Mitosis. February 13, 1962.

Went, Hans A., Washington State University, Pullman, Washington. Modification of the mitotic apparatus and cleavage patterns in sand dollar eggs. February 14, 1962.

Uyeki, E. M. Biological properties of DNA. February 20, 1962.

Erdman, H. E. Biochemical polymorphism in genetic systems. February 27, 1962.

d. Miscellaneous

None

D. Publications

Documents:

Schiffman, R. H. The uptake of strontium from diet and water by rainbow trout. HW-72107. (December 1961).

Open Literature:

Bair, W. J., D. H. Willard, and L. A. Temple. 1961. Plutonium inhalation studies. I. The retention and translocation of inhaled $\text{Pu}^{239}\text{O}_2$ in mice. Health Physics 7: 54-60.

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1-E

OPERATIONS RESEARCH AND SYNTHESIS OPERATION
MONTHLY REPORT - FEBRUARY, 1962

ORGANIZATION AND PERSONNEL

There were no changes in organization or personnel during the month.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Fuels Preparation Department

Two previously designed production tests were redesigned to account for changes made in some of the independent variables. The heat treatment production test was changed to incorporate core blank size as an additional variable, while deleting brine concentration as a variable. The test was further changed to disallow the use of an oil quench on fuel elements heat treated in the solid form. The ingot-dingot; rolled-extruded test was simplified by the elimination of ingot uranium, making the test wholly one of evaluating different fabrication routes. Rolling, hollow extrusion, and solid extrusion are the three routes. Further assistance was given in the design of the test to evaluate alternate cladding processes.

Some data have accumulated in connection with the Quality Certification Program which permits a comparison of ingot and dingot uranium. Such data were analyzed. It was pointed out that since the earlier dingot material was heat-treated primarily at FMPC, and the later material at MCW, it was impossible to separate any potential-site-of-heat-treating effects on dimensional stability from other time associated, but unidentified, variables.

Work is proceeding on up-dating the empirical models relating dimensional instability to reactor variables.

Consultation was provided in the development of a generalized data reduction program designed to summarize the masses of fuel element data into a few meaningful statistics. This is being developed primarily for pre-irradiation data.

Assistance was given in interpreting measurement error results from an analysis of pre-irradiation measurement station data. Two operators were involved, each making three sets of measurements. The end purpose of the analysis is to associate a total measurement standard deviation with each type of measurement, and to indicate how much of this is due to imprecision, and how much is due to bias between operators and/or runs. A report is being prepared to indicate how to use Mercy results to generate these numbers in the general case. In this connection, a presentation of the Mercy program was given members of the Quality Control Operation. It is further planned to incorporate additional calculation steps in the 7090 program to minimize the amount of hand calculation presently required to complete the analysis.

2-F

Data from two welding tests were analyzed. One was an evaluation of single-pass versus double-pass weld; the other evaluated weld speed. In presenting the results, suggestions were given on how to improve future experimentation in this area, and these suggestions were incorporated in a subsequent test.

A study is being made of the sources, measurement and other, contributing to the over-all variation observed in shoulder heights of bumpers, both before and after welding.

Further work is being done in the evaluation of coextrusion process variables affecting fuel element warp. Behavior of warp magnitude along the extrusion is being characterized. Also, several test designs were submitted in an attempt to isolate the effects of several variables which are largely confounded in existing process data.

An investigation is being made of wall thickness data from a large number of NPR outer extrusions in attempting to locate optimum sets of process conditions based on this criterion. The data are currently being prepared for submission to the computers.

Results obtained from the first trial run of the manufacturing scheduling model prove conclusively that the model is unfortunately one of those whose discrete and continuous variable optima do not coincide. This precludes the use of standard linear programming techniques to solve the model. Although there does exist a well known algorithm guaranteed to obtain the desired discrete variable optimum, the additional effort in programming and computation time virtually prohibits its use on such large models.

Irradiation Processing Department

Work continued on the problem of providing a mathematical model for the statistical evaluation of data from guided bend tests on samples of welding from the primary piping for the NPR project.

Work continued on the provision of a reliability analysis of various reactor safety systems employing a k out of n type of trip logic. Current effort is directed toward incorporating into the analysis a consideration of routine periodic trip checking of devices. Figures of merit are being calculated for specific cases.

A presentation was made to the area analysts in connection with the objective appraisal of the effectiveness of downtime utilization. The specific example used was concerned with the control of charge-discharge work rates, but it was emphasized that this was presented primarily as an example of how similar types of data could be analyzed. Further work is planned in this area as a result of comments solicited at this meeting.

Some 44 weeks of craft overtime data were analyzed to determine if reassignments of personnel can, in any instance, be expected to result in less overtime. A report was written presenting the results of this rather rough analysis. It was emphasized in the report that more valid answers should

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result from the computer simulation study of reactor operations currently under way. All supporting information files, and programs to progressively update them, have been completed. During the month the reactor experience data delineating cause and down activities were obtained. These data have been reviewed to establish file structure and field definition, and are currently being put into machine language. Tentative analytical procedures to establish the major starting points have been defined and will shortly be translated into machine programs. From these results, it is anticipated that the methods for deriving the necessary probability distributions for the simulation model can be developed.

Chemical Processing Department

Preliminary analyses were made of dimensional measurements data for tails and plugs. The large amounts of available data, and the potentially large number of variables which can affect the measurements, dictate that a complete analysis will have to be made using machine computation. The purpose of the preliminary study was to give indication of the type of analysis that should be made on the complete set of data, once it is made available.

A review was made of the feasibility of using measured specific gravity and nitric acid concentrations to estimate plutonium concentration in the separations plants. At present, the primary measurement point for nuclear material control occurs at the plutonium nitrate transfer point, which does not have an independent verification. While the use of specific gravity is in some respects inadequate, it may prove to be a valuable indicator of unsuspected bias.

Several sets of data used in the acceptance sampling of pipettes were analyzed using available computer programs.

Further consultation was provided relative to determining specific areas within Z Plant where direct application of the process control computer appears most feasible as a result of the experience obtained from the demonstration run. A review and critique of the demonstration evaluation report was also furnished as requested.

Discussions were held concerning the possibility of writing a program to convert manufactured parts gauging data from the design coordinate system to another which is more convenient for specific gauges.

Relations Operation

Salary curves continue to be fitted as requested. In addition, minor amounts of consultation have been given in connection with the interpretation of certain of these curves.

Contract and Accounting

Consultation was given in connection with graphically depicting data which relate to costs and procurement costs.

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STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HLO2000 Program

Pulse Column Facility

Further work was done on the statistical evaluation of the twenty pilot study experiments run on pulse column last summer. The average value mass transfer indices, fraction of total uranium throughput transferred and absolute transfer rate, are being correlated with column control variables.

Because of an inability to monitor low flow rates the original flow rate instrument, a rotometer, was replaced by a turbidimeter. An experiment was designed to calibrate this instrument.

A nonlinear least squares procedure is being debugged, which will allow the estimation of several parameters in the system of nonlinear differential equations, which expressed the mass transfer dynamics of the extraction process as a function of column position.

Nondestructive Testing

A number of sets of X-ray diffraction data were processed in connection with an investigation of a new method for assaying the strain relief characteristics of annealing on NPR piping welds. A correlation analysis is currently under way using the summary parameter estimates of average strain and material heterogeneity obtained from each of the above analyses. It is hoped that these parameters can be related to certain variables which describe the annealing process.

General

A numerical method was devised for measuring the thermal conductivity of materials at extremely high temperatures. The method is based on the use of measured temperatures at selected points on the sample in formulas obtained from approximate solutions of the diffusion equation with black-body radiation boundary conditions.

Several discussions were held with interested persons concerning the possibility of building a materials testing raw data file and statistical analysis system in conjunction with the storage and retrieval system which is currently in the developmental stages.

Data from the chromate-dichromate autoclave treatment test are currently being analyzed. This was a 4 factor, - 3 level test consisting of 27 observations blocked in blocks of size 9, and is concerned with developing an autoclave film which provides improved resistance to uniform corrosion attack.

Further work was done in connection with the determination of empirical relationships arising from buckling experiments.

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3000 Program

Machining Development

Sample magnetic tapes designed to test specific characteristics of the proposed Gorton lathe automatic control mechanism have been prepared on the 7090 for the first time. These tapes are to be used in a series of tests on the control mechanism prototypes at the General Engineering Laboratories in Schenectady.

4000 Program

Waste Fixation

Data from a recent mineral bed waste column experiment are being analyzed. The current experiment is the second in a series to study column optimization as a function of three factors, waste solution, residence time, and bed particle size. Several column efficiency indices, including the number of theoretical plates and the number of column volumes to fifty percent breakthrough are being correlated with these factors. Further experiments are contemplated once the direction of improved column operation has been established from the present analysis.

Swelling Study

Further programming was done of statistical procedures for detailed analysis of micrograph pore size distributions. The routine evaluation of void fraction and density estimates for photographs of uranium samples continues.

PPF

Further discussions were held concerning the evaluation of D₂O inventory procedures.

Several precision and accuracy studies are being done. Currently, analytical methods of estimating trace boron content and trace plutonium content are being evaluated.

6000 Program

Biology and Medicine

The results of a statistical analysis of data from an experiment involving rats subjected to a partial hepatectomy were written and forwarded to the interested person.

Work continued on an IBM program to find parameters for a model for the mice inhalation study.

Initial orientation was begun on a fish feeding experiment to which the multi-compartment model will be applied. The final draft of an article on cesium content in milk was completed, and will be submitted for publication in "Science" magazine.

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Other

Atmospheric Diffusion Studies

Some progress was made toward improving the tentative diffusion curves for the APE data. Work continues in this area.

Carl G. Bennett

Manager
Operations Research and Synthesis

CA Bennett:dgl

PROGRAMMING OPERATIONFEBRUARY, 1962REACTOR DEVELOPMENT - 04 PROGRAMPLUTONIUM RECYCLE PROGRAMSummary of Bred Fuel Value Calculations

At AEC request, the values of plutonium and U-233 in five different simulated reactor concepts with various uranium price schedules and various fuel inventory charges were being calculated with the MELEAGER CHAIN code. The reactor types simulated in this study are the APWR, the BWR, the HWR, the GCR, and the OCR as described in a document which was issued during the month (HW-72217 -- "Fuel Cycle Analysis for Successive Plutonium Recycle - I. Results for Five Reactor Concepts"). In this document the July 1, 1961, uranium price schedule was used. In the current study, various uranium price schedules are used with use charge rates of 4.75, 7, 10, and 12.5 percent. The cases completed were mainly batch irradiation; graded cases were being run.

Salt Cycle Processing

The fourth and last material balance subroutine for the Salt Cycle economics code was completed and debugged. This subroutine provides for adding fresh UO_2 in varying amounts and enrichments to the Salt Cycle process for recovery as mixed PuO_2 - UO_2 fuel along with varying quantities of spent fuel in a plutonium recycle reactor concept. Certain process conditions are handled as input variables.

Work is proceeding on the economics program utilizing these subroutines to determine the economic optimum process concept. When completed it will be possible to compare the Salt Cycle economics relative to conventional reprocessing for various combinations of stated economic parameters.

Irradiation of Structural Materials

A review of the irradiation program of the Physical Metallurgy Operation disclosed that irradiation of structural materials in the PRTR to obtain in situ stress-strain relationships is of interest. If the access holes to the calandria can be used, the PRP will not be materially affected. If the regular process tubes are used, three fuel elements will be displaced causing a $3\frac{1}{2}$ percent change in the fuel element processing schedule. This program was still under study.

Thallium Irradiation

Although there is no present laboratory effort on the photochemical excitation process for pure mercury isotope preparation, study work on the application of such isotopes as reactor coolants is still under way. A research and development proposal for laboratory study of the process was submitted. If such a study is authorized, the availability of a few milligrams of certain isotopes, particularly mercury-204, would be very helpful. The stable isotopes which are listed as available are generally rather crude fractions of variable composition. For example, for mercury-204, the purest material listed is 80 percent at a price of \$10/milligram. In anticipation of further interest in reactor applications for purified, low cross section mercury isotopes, particularly mercury-204, a proposal has been made to produce a few hundred milligrams of mercury-204 by a unique process which is expected to yield the desired product essentially without contamination with other mercury isotopes. The proposed method involves the irradiation of a few kilograms of thallium (Tl-203 plus Tl-205 in the natural ratio). On aging the radionuclide, Tl-204, decays mostly to lead-204; however, about two percent of this decays to yield mercury-204 by electron capture.

Preliminary action was taken to scope the feasibility, problems and costs of such an irradiation program. In addition to the resulting high purity isotope there will also be produced a substantial fraction of thallium-204 which may be of interest as a moderately long lived, low intensity, radionuclide heat source (four year half-life - pure beta emitter - about 0.7 watt per gram of pure Tl-204). Other undefined research benefits may be realized from this proposed small program.

SPECIFIC FUEL CYCLES

Reduced U-238 Spatial Concentration

The effect on computed fuel costs of reducing the average U-238 spatial concentration of the fuel has been studied for two fueling systems in several generalized reactors whose relative moderating powers are different. The plutonium selected for this study contains 45 percent Pu-239, 40 percent Pu-240, 10 percent Pu-241, and 5 percent Pu-242, and was chosen as a typical high Pu-240 composition batch that has low plutonium values when used to enrich an "inherited" lattice which was optimized for U-235:U-238 enrichment. In some inherited lattices, this plutonium showed high values if the U-238 concentration was reduced. This study is designed to see if there is general benefit in reducing the U-238 spatial concentration if one can also change

the lattice spacing. For each fueling scheme, three reactors were analyzed. These had SDFV's⁽¹⁾ of 3, 5, and 8 which correspond to varying the moderator-to-fuel ratio over a wide range. Minimum fuel costs were obtained for the following concentration of U-238 in theoretically dense UC_2 : 33-1/3, 50, 66-2/3, 85, 100, and 133-1/3 percent.

The computed minimized fuel costs for combinations of SDFV and concentration are shown in Table I.

TABLE I
MINIMIZED FUEL COSTS

<u>Mils per KWH</u>						
<u>SDFV = 3</u>						
U-238 Concentration	<u>33-1/3%</u>	<u>50%</u>	<u>66-2/3%</u>	<u>85%</u>	<u>100%</u>	<u>133-1/3%</u>
U-235 in U-238	2.26	2.13	2.01	1.89	1.82	1.73
Pu in Tails	1.8	1.74	1.71	1.73	1.88	--
<u>SDFV = 5</u>						
U-235 in U-238	2.3	2.19	2.07	1.95	1.86	1.69
Pu in Tails	1.92	1.85	1.79	1.74	1.70	1.63
<u>SDFV = 8</u>						
U-235 in U-238	2.35	2.22	1.95	1.97	1.88	--
Pu in Tails	1.98	1.92	1.86	1.80	1.75	1.65

(1) Is explained in the following equation.

$$\text{SDFV} = \frac{\xi \sum_s V_m \phi_m}{V_f \phi_f}$$

where, V_m = moderator volume
 V_f = fuel volume
 ϕ_m = flux in moderator
 ϕ_f = flux in fuel
 ξ = logarithmic energy decrement
 $\sum_s$ = macroscopic scattering cross section of the moderator.

Inspection of the above tables shows that for a SDFV of 3, the fuel costs are less for plutonium enrichment when the U-238 spatial concentration is less than 100 percent. However, if the SDFV is 5 or 8, the fuel costs are less the higher the U-238 density. In the case of U-235 enrichment, the fuel costs are always less for the higher the U-238 density. This does indicate that if consideration is given to plutonium enrichment of a reactor with SDFV of 3, that the plutonium value is higher if the U-238 density is decreased. The general combined cycle study treats this in greater detail and the above conclusions should be labeled, "current observations."

An attempt has been made to normalize the necessary economic and physics parameters so that logical comparison can be made between cases run at different U-238 concentrations. Specifically, as the U-238 concentration was reduced, the specific power was raised so that the power per cubic centimeter of fuel was essentially constant.

The MELEAGER code parameter, SCA, which is a function of spatial concentration of the fuel and which enters into the formula for self-shielding was also adjusted according to the density. The amount of variation of the economic parameters necessary to normalize for varied spatial concentration is not as clear cut as with the physics parameters. It might, at first, seem mandatory that the Fuel Element Fabricating and Jacketing charges be kept constant per unit volume of fuel element. This has been the approach used so far in the study. However, it should be noted this neglects a possible virtue of reduced spatial concentration since low density U-238 fuel elements may make possible entirely new types of elements such as a metallic-plutonium matrix, which contains a large void volume for fission gas collection in UO_2 . The separations costs have appeared as an extremely important factor in the density study. In some cases the separations cost has run as high as 25 percent of the total fuel cost. The breakdown of the separations cost as it has been used is:

\$10/pound	plutonium and uranium separation cost.
5.60/kg	uranium conversion cost.
1.50/gram	plutonium conversion cost.

The portion of the fuel cost directly attributable to plutonium conversion cost has varied from 0.2 to 0.4 mill/kwh_e. It has been highest in the high density cases. The possibility of fuel handling methods such that the plutonium and uranium nitrate conversion steps are eliminated makes it mandatory that the economics evaluation include cases where conversion costs are eliminated and separations costs are based on \$/unit volume basis as an alternative to the \$/pound basis. This is the next project of the density study.

The second in a series of formal reports describing work on fuel cycle analysis was issued. This report, "QUICK - A Simplified Fuel Cost Code," HW-71812, describes the economic model and IBM-7090 computer code used to calculate fuel cost.

Combined Cycles

The major effort of the combined cycle analysis this month was concerned with computation of minimized fuel costs using two alternative fertile materials, U-238 and Th-232, each enriched with five alternatives (U-233, U-235, and three different isotopic compositions of plutonium) in various reactor lattices so as to span most thermal reactors under consideration today. Batch and graded cycles were computed for several different neutron leakage and neutron control allowance levels. MELEAGER computations which supply the basic physics results were about 90 percent completed for this study. The successive computations were extended to progressively lesser degrees, as indicated in Table II. The fuel cost change which results from varying the cross sections in MELEAGER is to be computed to coincide more nearly with that computed with the General Atomics Spectrum code which uses 100 energy groups. These results will be used to define the conservatism introduced by the MELEAGER code when used for closely spaced lattices with the plutonium enrichment series as listed in Table II.

OTHER ACTIVITIES

Hazards Analysis

The hazards analysis of the Hanford Test Pile (305 reactor) was completed and a memorandum issued to Fuels Preparation Department. The accident assumed involved a reactor excursion as a result of inadvertently charging enriched material in the form of unclad I and E elements. Failure of operators and instruments was assumed. The excursion would then be limited by the decrease in reactivity with increasing temperature. Under these conditions the release of radionuclides and the off-plant exposure are estimated to be insignificant. The estimated release rate of noble gases would be considerably less than the routine release rate of A⁴¹ from one of the production reactors. Greatest uncertainties in the estimate are believed to be the temperature achieved in the enriched fuel and the diffusion rate of fission products out of the graphite structure. Assumptions involved are believed to be conservative. This work was completed in conjunction with others in HLO and FPD.

TABLE II
IDENTIFICATION OF FEED COMPOSITIONS BY SERIES NUMBER

Assigned Series No.	Fuel Combinations	Plutonium Compositions Used for Fissile Enrichment					Degree of Completion		
							MELEAGER %	PROTEUS* %	QUICK and MINIMIZER %
		Pu-239 %	Pu-240 %	Pu-241 %	Pu-242 %				
1,000	U-235 in uranium		None				90	90	60
2,000	U-233 in depleted uranium		None				90	90	60
3,000	Plutonium in depleted uranium	70	18	11	1		90	90	60
4,000	Plutonium in depleted uranium	95	5				90	90	60
5,000	Plutonium in depleted uranium	31.3	33.6	25	10.1		90	90	60
6,000	U-235 in Th-232		None				75	75	20
7,000	U-233 in Th-232		None				75	75	20
8,000	Plutonium in Th-232	70	18	11	1		75	75	20
9,000	Plutonium in Th-232	95	5				75	75	50
10,000	Plutonium in Th-232	31.3	33.6	25	10.1		75	75	50

* Extends MELEAGER data.

Quarterly Report

Preparation was started on a quarterly report describing the work done at Hanford for the Division of Reactor Development. The report is to contain ten sections (exclusive of Introduction and Summary sections) - each section presents an area of work and may contain more than one DRD program. To expedite publication, each section of the report can be prepared separately as it becomes available. Submittals for eight sections were received during the month.

Hanford Science Colloquium

Hanford Science Colloquium speakers and subjects have been scheduled as follows:

April 3, 1962 - Professor Francis Birch, Harvard University, Geophysicist,
"The Physics of the Earth's Interior"

May 15, 1962 - Colonel John P. Stapp, USAF, MC, Assistant to the Commander
of Aerospace Medicine, "The Challenge of Manned Space
Flight".

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RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF FEBRUARY 1962

A. ORGANIZATION AND PERSONNEL

Effective February 5, Allen R. Maki was transferred from Radiation Monitoring to Environmental Studies and Evaluation, and Andrew L. Stinnett was transferred from Environmental Studies and Evaluation to Radiation Monitoring. Pattie C. Freed was reactivated into Composite Dose Studies and Records on February 6. Geraldine P. Staats resigned from the Company effective February 9. Frances S. Glick was deactivated due to personal illness effective February 11. Effective February 12, Melvin L. Brinkerhoff was transferred from the Fuels Preparation Department to Internal Dosimetry, and Paul E. Cox was transferred from Internal Dosimetry to the Irradiation Processing Department.

B. ACTIVITIES

Occupational Exposure Experience

No new cases of plutonium deposition were confirmed by bioassay analyses during the month. The total number of plutonium deposition cases that have occurred at Hanford remains at 283, of which 204 are currently employed.

An HLO technician received a plutonium contaminated abrasion to a knuckle on his hand at the 231-Z Building when his gloved hand brushed against a revolving polishing wheel. Contamination to 80,000 d/m at the injured area was reduced to about 20,000 d/m by radiation monitoring personnel. Further decontamination was accomplished by an Industrial Physician. An examination at the Whole Body Counter indicated about 1.3×10^{-3} μ c of plutonium remained after completion of the medical action. Bioassay sampling was initiated.

A CPD process operator was involved in two separate occurrences involving the spread of plutonium contamination at the 234-5 Building. Nasal contamination to about 300 d/m was detected after a glove rupture early in the month and special sampling was initiated. In the second incident, facial contamination to 40,000 d/m occurred during the removal of material from a hood through the plastic bag port. The employee was wearing respiratory protection but failed to make a personal survey before removing the mask. Nasal contamination to 4000 d/m was detected. This occurrence was discussed with an Industrial Physician, but medical treatment was not prescribed. A preliminary bioassay sample indicated that minor additional deposition occurred for the employee, who is a previous deposition case estimated as less than 10 percent of the permissible body burden.

Five separate occurrences involving the exposure of 10 Plutonium Metallurgy employees and 2 CPD maintenance people to plutonium air contamination occurred at the 231 Building. The maintenance employees were exposed when a plutonium contamination spread occurred while they were working on a vacuum welding hood. The other occurrences were the result of a glove rupture, inadvertent removal of a plastic bag from the hood port, and failure to obtain adequate seals on a plastic bag and can, each containing plutonium samples. Bioassay samples were requested for all personnel involved.

A regular whole body examination of an HLO technician revealed a body burden of 0.2 μ c cobalt-60 (~2% of the maximum permissible body burden). Investigation revealed that the intake had probably occurred the day previous to the examination while the employee was working with irradiated graphite boats at the 3730 Building. Smaller quantities of cobalt-60 were detected in the examinations of 5 other personnel similarly exposed.

High level contamination was detected in the Purex railroad tunnel following the removal and transfer of the F-11 concentrator to the tunnel burial extension. The contamination was confined to the tunnel area except for that existing on the flat car spacers used in the burial operation. These cars were removed for further decontamination at another site. Air contamination required fresh air respiratory protection in the railroad tunnel for several days after the burial. Examination of several Purex employees at the Whole Body Counter revealed trace quantities of radioactive ruthenium in a few people.

Regular processing of the film badge dosimeter for an EFD reactor operator indicated a dose of 1.8 rads beta and 0.9 r gamma. Neutron dosimeter results for the corresponding period showed 0.2 rems fast neutron dose. The exposure exceeded the Hanford operational controls which limit the skin dose to 2 rems and the penetrating dose to 1 rem in a 4-week period. Investigation failed to reveal a satisfactory explanation for the exposure.

An emission of a broad beam of X-radiation from the unshielded portion of an X-ray diffraction unit in room 16 of the 231 Building occurred. One of the seldom-used windows was opened while making adjustments to the X-ray head. Later, the X-ray machine was turned on for an overnight exposure and the operator left the room. After about three minutes of operation, the operator returned to the room. As he entered the room, he noticed the open window and shut off the machine, exposing himself for about two seconds. A film study is being conducted to precisely define the parameters and intensity of the beam. The employee's dosimeter film showed a total of less than 10 mr.

Decontamination of cells in the High Level Radiochemistry Facility was initiated. Contamination levels on the order of 60-70 rads/hour exist in the cell. Rigid bioassay schedules and whole body examinations were arranged for all personnel involved in the flushing, cleaning, and equipment removal operations.

Radiation monitoring services were requested on two occasions by outside interests. A survey performed at Goodwill Industries, Pasco, Washington, of a small chemistry set-type bottle labelled "radium" revealed no contamination. Surveys performed for the Kennewick Police Department regarding radioactive markers that had been removed from the city dump by boys revealed no spread of contamination at the home or the dump site. Approximately 35 markers were collected during the survey. Radiation measurements revealed dose rates of 150 mrad/hour, including 8 mr/hour (uncorrected for source size) emanating from each marker. Gamma scan established the contents of each marker as radium. Further analysis showed approximately 4 μ c of radium in each marker.

Year-end reports for 1961 external exposure of Hanford personnel were received from data processing. The annual summary of exposure experience was prepared for the AEC as required by AEC Manual Chapter 0523. The maximum penetrating dose

recorded in 1961 at Hanford was 4.5 rems, received by a CPD technician assigned to the 234-5 Building. There were 4 employees who exceeded the 3 r/year operational control, based on the film dosimeter system

Failure to receive the complete order of the radiation exposure report card forms delayed distribution of the annual report cards to employees. Inquiries were initiated by Data Processing personnel to speed delivery by the supplier and the cards were expected in time to permit distribution early in March.

Environmental Experience

The concentrations of fallout materials at various locations in the Pacific Northwest averaged $5.2 \mu\text{mc}/\text{m}^3$ during February. This is about 20 percent under that reported last month.

The average concentrations of radionuclides of HAPCO origin in the Pasco sanitary water was equivalent to < 2 percent of the MPC_{W-GI}. The trend continues downward.

A total of 93 fish, mainly whitefish and suckers, was taken from sampling locations on the Columbia River between Priest Rapids and Richland. Eighty-two produce samples were obtained for radiochemical analysis. These included milk from Ringold, Riverview, Benton City, Mesa, and Eltopia that totaled 82 gallons; two samples of oysters from Willapa Bay; and 51 sets of beef thyroid glands.

Studies and Improvements

The effluent of the stack of Uranium Scrap Burning Facility, Building 303-L, was investigated. The facility appears well designed and its operation should not result in any appreciable release of uranium to the environment.

The hazards evaluation of CPD facilities continued. The waste storage tank problem was judged an incredible operational accident.

One routine aerial flight was made over the project and adjacent lands and waters to collect background data and keep the emergency monitoring equipment in a state of readiness.

AEC approval was obtained for the use of their aircraft in aerial determination of fishing pressure in conjunction with the creel survey program. Forms sent in by the State Game Department indicated a total of 96 contacts with fishermen during the month.

Gamma energy analyses were made on a section of plutonium fuel element which contained several times the normal amount of Pu^{240} . Comparison of the energy analyses for the normal fuel elements and this element reveal that the gamma energy spectra are essentially the same. The significant difference between these fuel elements is the increase in the neutron dose rate from the Pu^{240} enriched material. The neutron dose rates from fuel elements of comparable geometry will increase slightly faster than the increase in Pu^{240} content.

Fabrication of the new personnel dosimeter processing machine continued in the 328 Technical Shops. A film packet holder was designed to improve the X-ray coding of radium-gamma and uranium-beta calibrated film. With this assembly, the routine processing time for identifying this film will be reduced. The X-ray identification markings will conform with the standard code that is used to identify all personnel dosimeter film.

Calibration studies of the new Hanford film badge dosimeter have indicated that its improved low energy gamma detection is sensitive enough to detect low energy gamma photons from the standard radium-gamma calibration source. For routine calibration purposes, these low energies will be shielded from the gamma calibration flux.

Fast neutron exposures to compare the NTA film response at various fast neutron energies were completed. Monoenergetic neutrons obtained by using the positive ion accelerator at various energies from 352 Kev to 5 Mev were used.

Final prints for the polyethylene double moderator for use with the new portable BF₃ neutron monitoring instruments were prepared. Calibrations Operation issued a purchase requisition to provide sufficient new double moderators for field requirements.

Studies were initiated to determine the effect of attenuation of fast neutrons through a human body on the Hanford neutron dosimeter. A phantom was used with the NTA neutron dosimeters to measure the fast neutron dose simultaneously at the chest and at the back of the body. Studies were also initiated to determine the effect of predosing in the evaluation of NTA film. Another experiment was started to determine the performance of the neutron dosimeter when shielded with various thicknesses of lucite plastic. At some optimum plastic thickness, it may be possible to improve the energy characteristics of this dosimeter.

Preliminary calibration of the Royco Particle Analyzer was studied using standard 0.557 and 1.17 micron latex particles. Calibration indicated that the instrument response was linear and that the particle size resolution is good. A Goetz aerosol spectrometer was considered for additional particle size studies. Available data indicates that this instrument will detect particles from 0.01 to 5 microns by applying centrifugal accelerator action to a continuous linear flow of the aerosol under study. The Royco Particle Analyzer detects particles from 0.32 to 10 microns.

Small, personally carried, air samplers were calibrated for air flow rates. A study was started to attain data to compare the breathing zone air activity with the air activity found by other air sampling techniques.

Design and electronic prints for new alpha, beta, and gamma radiation hand and shoe counters were reviewed and corrections were detailed for revision by Drafting. Three specifications for hand and shoe counters were studied: beta-gamma background compensated hand and shoe counters; alpha only hand and shoe counters; alpha-beta and gamma hand and shoe counters. Design prints and circuits for these counters are available but extensive review is required prior to standardization.

A study of the detection limit of the plutonium wound monitor at the 200 West First Aid Station was completed. With a confidence of 95%, a 1% body burden of plutonium can be detected through 1/8 inch polystyrene. Counter efficiency was determined as 0.785%; the maximum theoretical efficiency is 2%.

The Oregon and Washington State Boards of Health were each provided two Junos, two GM's and two Scintillation Portable Poppies as requested by the AEC. The instruments are to be a part of the Civil Defense program. Arrangements were completed for recalibration and any necessary maintenance.

Information from the Mound Laboratory indicates that neutron source M-199 was one of several that evidenced defective canning. A request was made to replace this source with a later model similar to the Mound Laboratory standard M-590. Arrangements were also completed to send the new source to the Bureau of Standards for calibration prior to its shipment back to Hanford.

During the month, 13 Emergency Monitoring Kits were exchanged and serviced. Each Emergency Monitoring Kit that is opened for any purpose is promptly exchanged and serviced. Brackets were fabricated and distributed to the field for mounting Emergency Monitoring Kits on the walls where they will not be a tripping hazard and where they will be plainly visible when needed.

An inventory was taken of all supplies and equipment that have been stockpiled for use as needed in the event of a Plant emergency. A complete listing of these will be issued in the near future.

The criteria for the placement of the Hanford criticality dosimeters were completed. Included are criteria that define the facilities requiring dosimeters and criteria that estimate the number of dosimeters that are needed at each facility.

Hand-held wind velocity meters were obtained for inclusion in emergency kits for environmental monitoring. Calibration of these simple instruments showed an accuracy of ± 10 percent which is quite adequate for the intended use. All equipment to be included in the emergency kits is now on hand and fabrication of the cases was started.

C. VISITORS AND VISITS

Four members of the Division of Operational Safety, USAEC, Washington, visited Hanford on an annual audit inspection. The visiting team included

John P. H. Kelley
Arthur Schoen
William Doran, M.D.
Alex Perge

Also visiting with members of the Radiation Protection Operation staff during the month was:

Ed Patterson - Office of Health and Safety, USAEC, Germantown, Maryland

D. RELATIONS

Eight suggestions were submitted by the personnel of the Radiation Protection Operation during the month of February. Four suggestions were adopted and nine were rejected. Ten suggestions submitted by RPO personnel are pending evaluation.

Training in the use of the 5000 r/hour dose rate meters and self-reading dosimeters was given to two shifts of firemen in 200 West Area, two shifts at the Central Fire Station, and two shifts of 300 Area firemen.

Radiation protection orientation included lectures to new 300 Area employees and new Biology Research employees.

A seminar on "Application of Atmospheric Dispersion Equations" was given to RPO personnel by E. C. Watson.

A seminar on "The Environmental Evaluations Program" was given to HLO personnel by R. F. Foster.

Safety meetings were held throughout the various components during the month with topics ranging from "Action in the Event of a Criticality" to safety in the office and home.

E. SIGNIFICANT REPORTS

HW-72200 - "Serious Radiation Event - Current Capabilities and Action Plan, Hanford Atomic Products Operation" by HAPO staff.

HW-72263 - "The Plutonium Recycle Test Reactor Shielding Evaluation" by L. D. Williams.

HW-72691 - "Summary of Radiological Data for the Month of January, 1962" by R. F. Foster.

HW-72854 - "Monthly Report - February, 1962, Radiation Monitoring Operation" by A. J. Stevens.

"Summary of Radiological Conditions in the Vicinity of Hanford for 1961" by I. C. Nelson.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDSExternal Exposure Above Permissible Limits

	<u>February</u>	<u>1962 to Date</u>
--	-----------------	---------------------

Whole Body Penetrating	0	0
Whole Body Skin	0	0
Extremity	0	0

Hanford Pocket Dosimeters

Dosimeters Processed	2,592	5,056
Paired Results - 100-280 mr	9	16
Paired Results - over 280 mr	0	1
Lost Results	0	0

Hanford Beta-Gamma Film Badge Dosimeters

Film Processed	8,752	18,473
Results - 100-300 mrad	137	1,003
Results - 300-500 mrad	19	113
Results - Over 500 mrad	4	40
Lost Results	10	30
Average Dose Per Film Packet - mrad (ow)	8.95	9.40
- mr (s)	27.51	24.39

Hanford Neutron Film Badge DosimetersSlow Neutron

Film Processed	712	2,824
Results - 50-100 mrem	1	1
Results - 100-300 mrem	1	1
Results - Over 300 mrem	0	0
Lost Results	0	0

Fast Neutron

Film Processed	196	717
Results - 50-100 mrem	6	103
Results - 100-300 mrem	17	131
Results - Over 300 mrem	0	1
Lost Results	0	10

Hand Checks

Checks Taken - Alpha	32,696	71,333
- Beta-Gamma	55,502	111,364

Skin Contamination

Plutonium	21	24
Fission Products	31	103
Uranium	1	4
Tritium	0	0

<u>Whole Body Counter</u>	<u>Male</u>	<u>Female</u>	<u>February</u>	<u>1962 to Date</u>
GE Employees				
Routine	19	1	20	45
Specials	29	0	29	36
Terminations	13	0	13	29
Non-routine	23	5	28	59
Non-employees	1	0	1	1
Pre-employment	0	0	0	0
	<u>85</u>	<u>6</u>	<u>91</u>	<u>170</u>

Bioassay

Confirmed Plutonium Deposition Cases	0	0*
Plutonium - Samples assayed	557	1,026
- Results above 2.2×10^{-8} $\mu\text{c/sample}$	19	39
Fission Products - Samples assayed	607	1,180
- Results above 3.1×10^{-5} $\mu\text{c/sample}$	1	5
Uranium - Samples assayed	301	470
Biological - Samples assayed	0	22

Uranium Analyses

<u>Sample Description</u>	<u>Following Exposure</u>			<u>Following Period</u>		
	<u>Units of 10^{-9} $\mu\text{c U/cc}$</u>			<u>of No Exposure</u>		
	<u>Number</u>			<u>Number</u>		
	<u>Maximum</u>	<u>Average</u>	<u>Samples</u>	<u>Maximum</u>	<u>Average</u>	<u>Samples</u>
Fuels Preparation	41.5	3.0	90	13.2	2.1	81
Fuels Preparation**	0	0	0	0	0	0
Hanford Laboratories	77.2	10.5	34	29.4	5.1	37
Hanford Laboratories**	0	0	0	0	0	0
Chemical Processing	25.5	5.1	30	13.1	3.8	29
Chemical Processing**	0	0	0	0	0	0
Special Incidents	0	0	0	0	0	0
Random	0	0	0	0	0	0

Iritium Samples

	<u>< 1 $\mu\text{c/l}$</u>	<u>> 1 but < 5</u>	<u>> 5 $\mu\text{c/l}$</u>	<u>Max. Value of</u>	<u>Total</u>
				<u>Sample > 5 $\mu\text{c/l}$</u>	
Urine Samples	89	145	38	25.6	272

	<u>Maximum</u>	<u>Number</u>
	<u>$\mu\text{c/ml}$</u>	<u>Analyzed</u>
D ₂ O Samples		
Moderator	533.0	10
Reflector	544.5	14
Primary Coolant	86.0	12

36

Other Water Samples

#2527 top and bottom shield	.0434	<u>100</u>
		<u>408</u>

* The total number of plutonium deposition cases which have occurred at Hanford is now 283, of which 204 are currently employed.

** Samples taken prior to and after a specific job during work week.

UNCLASSIFIED

CalibrationsNumber of Units Calibrated
February 1962 to Date

Portable Instruments

CP Meter	1,089	2,120
Juno	312	574
GM	592	1,146
Other	217	407
Audits	<u>110</u>	<u>207</u>
	2,320	4,454

Personnel Meters

Badge Film	1,344	3,060
Pencils	0	4,780
Other	<u>411</u>	<u>897</u>
	1,755	8,737

Miscellaneous Special Services

Total Number of Calibrations	645	967
	4,720	14,158

Manager
RADIATION PROTECTION

AR Keene:ljw

FINANCE AND ADMINISTRATIONI. FINANCEA. Cost Accounting

An additional authorization of \$150,000 was received by Hanford Laboratories from the University of California, Lawrence Radiation Laboratory in connection with the Project Whitney program.

Preparation of the Budget for FY 1964 and Revision of Budget for FY 1963 is under way and is proceeding on schedule. The major effort by Hanford Laboratories this month was the preparation of research and development proposals. Radiation Protection Operation and Finance and Administration Operation detailed cost budgets were reviewed and approved by the managers concerned.

Effective February 1, 1962, organization codes were established for the new section, Finance and Administration Operation, and the existing section codes for the Technical Administration Operation and Laboratory Auxiliaries Operation were cancelled. Concurrently, codes assigned to the Programming Operation and Test Reactor and Auxiliaries Operation were revised. Fiscal year-to-date operating costs of the above organizations were recast on the basis of the new organizations. FY 1962 operating budgets and equipment allocations were also revised to reflect the new organizations.

One new program code was established during the month, .28, entitled "HLO Weapons Research and Development".

One special request code was established during the month, .3B - Biology Symposium on Transuranic Elements. Reimbursable costs are not to exceed \$7,500. Transportation and per diem of participants to be reimbursed must be approved individually by the Commission.

Billing arrangements were made with the Commission relative to the use of certain tube testing equipment in the C-25 Building during February while personnel of Kaiser Engineers were not occupying the facility. Financial arrangements were made to accept \$226 for the monthly landlord rental and \$149 for utilization of specific equipment items.

B. General Accounting

Savannah River Operations Office provided a new price list for Heavy Water based on their December 31, 1961 inventory. The price per pound is now \$13.76 fund and \$9.87 nonfund for a total of \$23.63. The new price results in a reduction of \$.20 per pound fund cost. Price allowance for scrap returned has also been reduced which will result in an additional charge to operating costs of \$1,051 based on the February 22, 1962 PRTR inventory.

Heavy Water losses chargeable to FRTR operating cost for the month of February amounted to \$37,875 representing (1) D₂O BFID loss of 2500.16 pounds valued at \$34,902, (2) 888.32 pounds of scrap generated during February valued at \$1,921, and (3) an additional charge to cost of \$1,051 resulting from a revaluation of scrap based on the new price list received from Savannah River Operations Office.

At February 28, 1962, accumulated scrap for return to Savannah River totaled 18,061.31 pounds valued at \$218,489.

Platinum scrap (3,695 grams) valued at \$10,317 was shipped to the New York Operations Office, AEC for reclaiming and credit. Upon receipt of confirmation of shipment, the material will be billed to the AEC. In addition, 2,117 grams of contaminated platinum valued at \$5,075 were made available for off-site shipment. This material will be shipped upon receipt of shipping instructions.

The physical inventory of movable catalogued equipment in the custody of Chemical Research and Development Operation is progressing on schedule, and reconciliation of inventory findings with plant records is in progress. At the present, there are a total of 24 unlocated equipment items valued at \$9,319.

SS material holdings increased \$4.3 million during February, bringing the total to \$24.8 million. The increase consists primarily of plutonium for use on programs sponsored by the Division of Reactor Development. SS Material Consumed in Research for all of HAPC in the current fiscal year at January 31, 1962 is \$1.7 million of which \$1.3 million are applicable to Hanford Laboratories and \$.4 million to Fuels Preparation Department.

Seventy-five items valued at \$57,061 were received in the Laboratory Equipment and Material Pool during the month of February. Twenty-three items valued at \$15,110 were loaned or transferred in lieu of placement of requisitions, nine items valued at \$5,577 were temporarily withdrawn by custodians and eleven items valued at \$5,363 were withdrawn permanently by custodians. There are currently 984 equipment items valued at \$502,317 physically located in the storage pool of which 27 items valued at \$86,609 are held for the convenience of others. Reactor and Other Special Materials on hand at the pool at month end totaled \$190,317, and materials held for the convenience of others totaled \$214,384.

A special report entitled Annual Industrial Fire Program and Experience was prepared showing the replacement value of plant and equipment assigned to Hanford Laboratories at December 31, 1961. Excluded from the report are underground installations and open structures, valued at \$1,607,817. The report indicated the replacement value of plant and equipment to be \$74,620,265 as compared to \$66,281,460 actual recorded book value which includes \$1,130,379 EWIP and \$774,894 OWIP. The index used in reflecting replacement value was taken from the Building Cost Index published by Smith, Hinchman and Grylls.

J. A. Jones Construction Company agreed with General Electric to discontinue billing final cost variances on completed work when the variance was \$25 or less. J. A. Jones Construction Company will absorb this variance in their Contractor Indirect Expense. This change will materially decrease clerical effort, pertaining to final costs, in both J. A. Jones Construction Company and General Electric Company.

Final unitization reports were issued on the following projects during February:

CGH-935	Metal Storage Building - 328 Building Addition	\$ 22 291
CGH-907	Strontium 90 Interim Program	419 615
CA -681	Hanford Equipment in the ETR (Arco)	1 037 659

During the month, \$122,226 of equipment was transferred to classified plant accounts from Equipment Work in Progress.

Two renewal consultant agreement contracts were processed and are listed below:

CA-322	J. L. Powell
CA-323	P. E. Church

Travel activity in Hanford Laboratories this fiscal year is below the level established in the past two years as follows:

Trips Started - Year-to-Date

FY 1960	800
FY 1961	829
FY 1962	719

Following is a summary of the status of letters or agreements covering specific actions requiring AEC concurrence:

<u>No.</u>	<u>Title</u>	<u>Status</u>
AT-52	Expanded Use of Whole Body Counter	Approved 1-26-62
AT-104	Fission Products Dispersal Handbook	Still held by AEC
AT-198	Miniature Swine for Hammersmith Hospital	To Commission 2-15-62
AT-205	Sanitary Engineering Advisory Committee - Washington State University	Approved 1-15-62
AT-220	Delay in Disposition of Residence of Transferred Employee	Approved 2-6-62
AT-222	Delay in Disposition of Residence of Transferred Employee	Approved 2-6-62

<u>No.</u>	<u>Title</u>	<u>Status</u>
AT-223	University of Washington Primate Research Center - Executive Committee	In Process
AT-225	University of Washington Request for Data	To Commission 2-21-62
AT-227	Participation of Spencer H. Bush in National Academy of Sciences Committee on Metallic Materials	To Commission 2-21-62
AT-228	Processing of Film Demonstrating the AlSi Canning Process	To Commission 3-5-62
AT-229	Space Allocation for Universities	To Commission 2-23-62

C. Personnel Accounting

Hanford Laboratories' payroll statistics for February are as follows:

<u>Number of HLO Employees</u>	<u>Total</u>	<u>Exempt</u>	<u>Non-Exempt</u>
<u>Changes During Month</u>			
Employees on Payroll at Beginning of Month	1 440	674	766
Additions and Transfers In	15	4	11
Removals and Transfers Out	21	8	13
Employees on Payroll at End of Month	<u>1 434</u>	<u>670</u>	<u>764</u>

<u>Overtime Payments During Month</u>	<u>February</u>	<u>January</u>
Exempt	\$ 4 068	\$ 3 771
Nonexempt	16 771	15 548
Total	<u>\$ 20 839</u>	<u>\$ 19 319</u>

<u>Gross Payroll Paid During Month</u>	<u>February</u>	<u>January</u>
Exempt	\$ 620 495	\$ 629 761
Nonexempt	398 104	395 292
Total	<u>\$1 018 599</u>	<u>\$1 025 053</u>

<u>Participation in Employee Benefit Plans at Month End</u>	<u>February</u>		<u>January</u>	
	<u>Number</u>	<u>Per cent</u>	<u>Number</u>	<u>Per cent</u>
Pension Plan	1 283	99.4	1 280	99.4
Insurance Plan				
Personal Coverage	364		371	
Dependent Coverage	1 059	99.9	1 055	99.9
U.S. Savings Bonds				
Stock Bonus Plan	91	39.6	93	39.9
Savings Plan	77	5.4	77	5.4
Savings and Security Plan	1 078	89.7	1 081	89.8
Good Neighbor Fund	961	67.1	967	67.3

<u>Insurance Claims</u>	<u>February</u>		<u>January</u>	
	<u>Number</u>	<u>Amount</u>	<u>Number</u>	<u>Amount</u>
<u>Employee Benefits</u>				
Life Insurance	0	\$ 0	0	\$ 0
Weekly Sickness and Accident	13	1 240	16	987
Comprehensive Medical	63	4 761	90	5 833
<u>Dependent Benefits</u>				
Comprehensive Medical	<u>132</u>	<u>11 628</u>	<u>148</u>	<u>13 725</u>
 Total	 <u>208</u>	 <u>\$17 629</u>	 <u>254</u>	 <u>\$20 545</u>

II. AUDITS AND PROCEDURES

A report on the follow-up of the Company Traveling Auditors' recent recommendations was issued. Each of their audit suggestions has been adopted and completed, with exception of one interdepartmental procedural proposal which does not appear to be of benefit.

The report on Audit of Calendar Year 1961 Suggestion Awards conducted by C&AO Auditing Operation was reviewed and issued, together with annual certification of the Hanford Laboratories suggestion award program. The administrative improvements recommended in the audit report have either been fully adopted or implementation has been started.

A final review was made of the cost estimates contained in HW-68420, "Economic Evaluation of a 300-Mw(e) Experimental Pressure Power Reactor". The capital and power production cost calculations are accurately and objectively presented and conform with the AEC Nuclear Power Plants Cost Evaluation Handbook.

Contract and Accounting Operation suggested that Hanford Laboratories make a rent versus buy review of eight units of currently rented IBM punched card equipment. Seven of these, a verifier and six card punches, were recommended for purchase; we will continue renting the remaining card punch.

A joint HOO-GE appraisal of Hanford Laboratories records management practices was completed during the month. Messrs. E. W. Slusher (GE) and R. V. Button (HOO) carried out a series of interviews among Laboratories management to appraise our adherence to established records management procedures. A report of their findings is not yet available.

A seminar on Critical Path Scheduling was arranged for HAPO people interested in this topic by Business Systems Development Operation, and 15 Hanford Laboratories employees were scheduled to attend. The local IBM office is presenting the series, which is designed to impart an introductory knowledge of this relatively new managing tool.

Procedures studies in Laboratories by Business Systems Development Operation personnel were devoted to completion of a review of the Tuition Refund program, and preparation of the study report on administration of Capital Equipment. At this time the reports are being duplicated for issue.

III. TECHNICAL ADMINISTRATIONA. Employee Relations

Nineteen nonexempt employment requisitions were filled during February with fifteen remaining to be filled.

A presentation in proper telephone technique was given for 45 secretaries.

B. Professional Placement

Advanced Degree - Six Ph.D. applicants visited HAPD for employment interviews. Seven offers were extended; no acceptances and three rejections were received. Current open offers total four.

BS/MS - Ninety-six program offers and one direct placement offer were extended; one program offer was rejected. Offers open total 102. Fourteen schools were visited for recruiting purposes.

Technical Graduate Program - Ten Technical Graduates were placed on permanent assignment; one new member was added to the rolls. Current program members total fifty-four.

C. Technical Information

The annual inventory of Research and Development Reports was completed. For 1961, all copies were accounted for.

IV. FACILITIES ENGINEERINGA. Projects

At month's end there were 10 active projects having total authorized funds in the amount of \$2,653,600. The total estimated cost of these projects is \$7,467,000. Expenditures on them through January 31, 1962 were \$789,000.

The following summarizes the status of FEO project activity:

Number of authorized projects at month's end	10
Number of new projects authorized in February	0
Projects completed in February	1
CGH-937 Safety Improvements - 231-Z Building	
New project proposals submitted to AEC in February	1
CGH-955 Reactivation of the H-1 Loop - 105-H Building	
New Projects awaiting AEC authorization	3
CAH-917 Field Service Center - Atmospheric Physics	
CAH-932 300 Area Retention Waste System Expansion	
CGH-955 Reactivation of the H-1 Loop - 105-H Building	
Project proposals complete or nearing completion	4
CGH-957 Small Particle Technology Laboratory (sent to AEC 3-2-62)	
CGH-958 Plutonium Fuels Testing & Evaluation Laboratory - 308 Building (awaiting W. E. Johnson's approval)	

CAH-959 Graphite Machining Shop (awaiting W. E. Johnson's approval)
Low Level Radiochemistry Laboratory (in last stages of
preparation - DSJ)

B. Services

Engineering services during February were provided to the following activities with satisfactory progress shown on each:

Laboratory Modifications - 321-A and 3706 Buildings
306 Building - Salt Bath Furnace Alarm
314 Building - Induction Heating Equipment Relocation
325 Building Analytical Laboratory Layout Study
146 FR Equipment Heat Load Study
108-F Electrical Voltage/Load Study
144-F - Humidity Recorder
FRTR Filter Plant Control Cable Relocation
Critical Mass Split-Half Machine

Twenty requisitions were processed during the month with a value of \$34,000. Total value of equipment purchases in process is \$650,000.

Plant Engineering work was devoted to the major activities listed below:

300 Area Standard Warning System
3718 Building Modifications
Retention Basin Proportional Samplers
Whole Body Counter Cell Door Mechanism
PCTR Ventilation
306 Building Ventilation
Pressure Systems Code Work (FRTR Rupture Test Loop and 300 Area Safety Valves)
Hot Waste Disposal Facility

Plant Engineering services were completed on these assignments:

326 Building, Room 12-A Air Conditioning

C. Maintenance and Operation

Costs for January are \$154,400. Costs to date (7 months) are 99% of forecast. Improvement maintenance costs for January were \$15,700.

The following tabulation summarizes waste disposal operations:

	<u>January</u>	<u>December</u>
Concrete Barrels	12	12
Loadluggers-Hot Waste	3	4
Milk Pails (up to 2-20-62)	23	22
Gatling Gun	1	3
Crib Waste	320,000 gal.	260,000 gal.
Total Beta Conc.	271/uc/gal.	293/uc/gal.

D. Drafting

The equivalent of 155 drawings were completed during the month. Overtime was 115 hours which is 2.9% of available hours. Total work backlog is 160 man-hours.

Major jobs in progress are: Break-A-Way Corrosion Loop, Fission Product Packaging, TF-23 Test Loop, 280 Ton Extrusion Press Installation, Welding Apparatus for 8 Ft. Fuel Element, Automatic Densitometer "As Built", PRTR Shim Rods, Vibration Compaction, A-C-Columns - 321 Building.

The reproduction equipment was transferred to FPD and will temporarily be operated in its existing location.

E. Construction

There were 65 existing J. A. Jones Company orders at the beginning of the month with a total unexpended balance of \$133,017. One hundred nine new orders, three supplements and adjustments for underruns amounted to \$69,014. Expenditures during the month on HLO work were \$80,801 (includes C. O. cost). Total J. A. Jones backlog at month's end was \$121,230.

Summary

	<u>No.</u>	<u>HL Unexpended Balance</u>	<u>No.</u>	<u>CE&U Unexpended Balance</u>
Orders outstanding beginning of month	64	\$130 546	1	\$2 471
Issued during the month (Inc. Supp. & Adj.)	112	69 014		
J. A. Jones Expenditures during month (inc. C.O. Costs)		80 801		
Balance at Month's end	63	118 991	1	2 239
Orders closed during month	110	146 975		

Construction activities completed during February were:

231-Z Building - Installation of Equipment removed from 234-5 Building
306 Building - Special Alloy Storage Facility
326 Building - Install 20 Ton Air Conditioner, Room 12A.
327 Building - Move Dark Room and Install Ladies' Rest Room
300-N - Construct New Burial Ground

Manager
Finance and Administration

W Sale:whm

SEMI-MONTHLY PROJECT STATUS REPORT						HW- 72902		
GENERAL ELECTRIC CO. — Hanford Laboratories						DATE 2-28-62		
PROJ. NO. CAH-822		TITLE Pressurized Gas Cooled Facility				FUNDING 4141 Operating		
AUTHORIZED FUNDS		DESIGN \$ 43,000		AEC \$ 0		COST & COMM TO 1-28-62 \$ 1,102,170		
\$ 1,170,000		CONST. \$ 1,127,000		GE \$ 1,170,000		ESTIMATED TOTAL COST \$ 1,170,000		
STARTING DATES	DESIGN 8-19-59	DATE AUTHORIZED 2-2-62	EST'D. COMPL. DATES	DESIGN 4-29-60	PERCENT COMPLETE			
	CONST. 10-17-60	DIR. COMP. DATE 6-30-62		CONST. 6-30-62				
ENGINEER TR&AO-MEE0 - DP Schively MANPOWER FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT-ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING						WT'D.	SCHED.	ACTUAL
					DESIGN	100	100	100
					TITLE I	100	100	100
					GE-TIT. II			
					AE-TIT. II			
					CONST.	100	91	91
					PF	1.4	0	0
					CPFF	22.1	99	99
					FP	6.6	100	100
SCOPE, PURPOSE, STATUS & PROGRESS Project Proposal Revision No. 4 has been approved (HOO-AEC, Directive No. AEC-145, Mod. No. 6) and extends project completion date to 6-30-62 and increases authorized funds to \$1,170,000. Percentage completions have been changed to reflect the new schedule and current status. Construction work is complete except for installation of the new heater and blowers. Bristol-Siddeley has apparently stopped work on the blowers. Efforts are being made to re-start vendor efforts to enable closing out order on 4-15-62 at a price commensurate with blower capabilities.								

SEMI-MONTHLY PROJECT STATUS REPORT						HW-72902	
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 2-28-62	
PROJ. NO.	TITLE					FUNDING	
CAH-842	Critical Reactivity Measuring Facility					58-e-15	
AUTHORIZED FUNDS	DESIGN \$ 45,000	AEC \$	COST & COMM. TO 1-14-62	\$ 180,169 (GE)			
\$ 400,000	CONST. \$ 355,000	GE \$	ESTIMATED TOTAL COST		\$ 400,000		
STARTING DESIGN 11-17-59	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN 2-1-61	PERCENT COMPLETE			
DATES CONST. 10-3-60	DIR. COMP. DATE 3-15-62	DATES CONST. 3-15-62		WT'D.	SCHED.	ACTUAL	
ENGINEER			DESIGN 100 100 100				
TR&AO-MEEO - WS Kelly			TITLE I				
MANPOWER			GE-TIT. II				
FIXED PRICE			AE-TIT. II				
COST PLUS FIXED FEE							
PLANT FORCES			CONST. 100 98 98				
ARCHITECT-ENGINEER			PF				
DESIGN ENGINEERING OPERATION			CPFF 60 97 97				
GE FIELD ENGINEERING			FP 40 100 100				
SCOPE, PURPOSE, STATUS & PROGRESS							
<p>The revised project proposal requesting additional funds and an extension of the scheduled completion date has been approved by the AEC. A tentative schedule has been prepared which reflects the changes in percent complete.</p> <p>CPFF percentages includes Government Furnished equipment. Authorization to spend the funds required for the additional work were received by the CPFF field forces the week of January 7.</p>							

PROJ. NO.	TITLE					FUNDING	
CGH-857	Physical & Mechanical Properties Testing Cell - 327 Bldg.					0290	
AUTHORIZED FUNDS	DESIGN \$ 45,000	AEC \$ -	COST & COMM. TO 2-18-62	\$ 201,997			
\$ 460,000	CONST. \$ 415,000	GE \$ 460,000	ESTIMATED TOTAL COST		\$ 460,000		
STARTING DESIGN 11-2-59	DATE AUTHORIZED 9-22-61	EST'D. COMPL. DATES	DESIGN 3-15-61	PERCENT COMPLETE			
DATES CONST. 2-12-62	DIR. COMP. DATE 12-15-62	DATES CONST. 12-15-62		WT'D.	SCHED.	ACTUAL	
ENGINEER			DESIGN 100 100 100				
FEO - KA Clark			TITLE I				
MANPOWER			GE-TIT. II 100 100 100				
FIXED PRICE			AE-TIT. II				
COST PLUS FIXED FEE							
PLANT FORCES			CONST. 100 NS 5				
ARCHITECT - ENGINEER			PF				
DESIGN ENGINEERING OPERATION			CPFF				
GE FIELD ENGINEERING			FP				
SCOPE, PURPOSE, STATUS & PROGRESS							
<p>This project will provide facilities for determining physical and mechanical properties of irradiated materials, and involves the installation of a cell in the 327 Building.</p> <p>Pouring of footings for the foundation walls is complete. Forming of the walls is in progress.</p> <p>Equipment procurement has been progressing satisfactorily except that some costs have exceeded the estimates. However, savings on other equipment have off-set these added costs. The balance on equipment procurement to date is \$12,000 "in the black".</p>							

SEMI-MONTHLY PROJECT STATUS REPORT						HW- 72902	
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 2-28-62	
PROJ. NO.	TITLE					FUNDING	
CGH-858	High Level Utility Cell - 327 Building					0290	
AUTHORIZED FUNDS		DESIGN \$	50,000	AEC \$	- -	COST & COMM. TO 2-18-62 \$ 346,602	
\$ 400,000		CONST. \$	350,000	GE \$	400,000	ESTIMATED TOTAL COST \$ 400,000	
STARTING DATES	DESIGN 11-1-59	DATE AUTHORIZED	4-6-61	EST'D. COMPL. DATES	DESIGN 2-15-61	PERCENT COMPLETE	
	CONST. 5-15-61	DIR. COMP. DATE	2-28-62		CONST. 6-1-62	WT'D.	SCHED. ACTUAL
ENGINEER						DESIGN	100 100 100
FEO - KA Clark						TITLE I	
MANPOWER				AVERAGE	ACCUM MANDAYS	GE-TIT. II	95 100 100
FIXED PRICE				2	505	AE-TIT. II	
COST PLUS FIXED FEE						Vendor	5 100 100
PLANT FORCES						CONST.	100 100 50
ARCHITECT-ENGINEER					35	PF	
DESIGN ENGINEERING OPERATION					930	CPFF	100 100 50
GE FIELD ENGINEERING					22	FP	
SCOPE, PURPOSE, STATUS & PROGRESS							
<p>This project will provide facilities to prepare specimens from irradiated materials for use in determining their physical and mechanical properties and involves the installation of a cell in 327 Building.</p> <p>A revised project proposal is being submitted to HOO-AEC requesting an extension of the construction schedule to 6-1-62 because of the late delivery of the cell assembly. The first casting is expected on site by 3-1-62. Installation of cell and equipment will take six to eight weeks if all goes well. The additional four weeks extension is a contingency. Total project completion extension is a direct result of the delay in receipt of the cell assembly.</p>							

PROJ. NO.	TITLE					FUNDING	
AUTHORIZED FUNDS		DESIGN \$	AEC \$	COST & COMM. TO		\$	
\$		CONST. \$	GE \$	ESTIMATED TOTAL COST		\$	
STARTING DATES	DESIGN	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN	PERCENT COMPLETE		
	CONST.	DIR. COMP. DATE		CONST.	WT'D.	SCHED.	ACTUAL
ENGINEER					DESIGN	100	
					TITLE I		
MANPOWER				AVERAGE	ACCUM MANDAYS	GE-TIT. II	
FIXED PRICE						AE-TIT. II	
COST PLUS FIXED FEE							
PLANT FORCES						CONST.	100
ARCHITECT - ENGINEER						PF	
DESIGN ENGINEERING OPERATION						CPFF	
GE FIELD ENGINEERING						FP	
SCOPE, PURPOSE, STATUS & PROGRESS							

SEMI-MONTHLY PROJECT STATUS REPORT						HW- 72902	
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 2-28-62	
PROJ. NO. CAE-866		TITLE Shielded Analytical Laboratory - 325-B Building				FUNDING 61-a-1	
AUTHORIZED FUNDS \$ 700,000		DESIGN \$ 60,000	AEC \$ 546,500	COST & COMM TO 2-13-62		\$ 99,244 (GE)	
		CONST. \$ 640,000	GE \$ 153,500	ESTIMATED TOTAL COST		\$ 700,000	
STARTING DATES	DESIGN 9-5-59	DATE AUTHORIZED 5-31-60	EST'D. COMPL. DATES	DESIGN 11-14-60	PERCENT COMPLETE		
	CONST. 6-28-61	DIR. COMP. DATE 6-30-62		CONST. 6-30-62			
ENGINEER FEO - RW Descenzo				DESIGN	WT'D.	SCHED.	ACTUAL
MANPOWER				TITLE I			
FIXED PRICE				GE-TIT. II	10	100	100
COST PLUS FIXED FEE				AE-TIT. II	90	100	100
PLANT FORCES				CONST.	100	43*	29
ARCHITECT-ENGINEER				PF	3	1	1
DESIGN ENGINEERING OPERATION				CPFF	2	0	0
GE FIELD ENGINEERING				FP	95	97**	28
<p>SCOPE, PURPOSE, STATUS & PROGRESS</p> <p>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.</p> <p>The painter has nearly completed applying the protective paint to the insides of the exhaust system duct work.</p> <p>Acceptance testing of the piping has been started.</p> <p>The first two panels of the Meehanite wall sections were to be shipped on February 28, 1962.</p> <p>The five protection sprinkler system installation has been started.</p> <p>The electrical subcontractor has resumed work.</p> <p>*Official project construction schedule.</p> <p>**Contractor's construction schedule does not reflect actual conditions.</p>							

SEMI-MONTHLY PROJECT STATUS REPORT						HW- 72902																																									
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 2-28-62																																									
PROJ. NO. CAH-867	TITLE Fuel Element Rupture Test Loop					FUNDING 58-e-15																																									
AUTHORIZED FUNDS \$ 1,500,000	DESIGN \$ 130,000 CONST. \$ 1,370,000	AEC \$ 820,000 GE \$ 680,000	COST & COMM. TO 1-30-62 ESTIMATED TOTAL COST		\$ 525,581 (GE) \$ 1,500,000																																										
STARTING DATES DESIGN 8-1-60 CONST. 11-2-60	DATE AUTHORIZED DIR. COMP. DATE 6-30-62	EST'D. COMPL. DATES DESIGN 3-15-61 CONST. 6-30-62	PERCENT COMPLETE																																												
ENGINEER TR&AO-MEEO - PC Walkup			<table border="1"> <thead> <tr> <th></th> <th>WT'D.</th> <th>SCHED.</th> <th>ACTUAL</th> </tr> </thead> <tbody> <tr> <td>DESIGN</td> <td>100</td> <td>100</td> <td>100</td> </tr> <tr> <td>TITLE I</td> <td></td> <td></td> <td></td> </tr> <tr> <td>GE-TIT. II</td> <td>91</td> <td>100</td> <td>100</td> </tr> <tr> <td>AE-TIT. II</td> <td>9</td> <td>100</td> <td>100</td> </tr> <tr> <td>CONST.</td> <td>100</td> <td>88</td> <td>86</td> </tr> <tr> <td>PF</td> <td>2</td> <td>100</td> <td>50</td> </tr> <tr> <td>CPFF</td> <td>57</td> <td>100</td> <td>95</td> </tr> <tr> <td>FP (1)</td> <td>10</td> <td>100</td> <td>100</td> </tr> <tr> <td>(2)</td> <td>31</td> <td>63</td> <td>67</td> </tr> </tbody> </table>						WT'D.	SCHED.	ACTUAL	DESIGN	100	100	100	TITLE I				GE-TIT. II	91	100	100	AE-TIT. II	9	100	100	CONST.	100	88	86	PF	2	100	50	CPFF	57	100	95	FP (1)	10	100	100	(2)	31	63	67
	WT'D.	SCHED.	ACTUAL																																												
DESIGN	100	100	100																																												
TITLE I																																															
GE-TIT. II	91	100	100																																												
AE-TIT. II	9	100	100																																												
CONST.	100	88	86																																												
PF	2	100	50																																												
CPFF	57	100	95																																												
FP (1)	10	100	100																																												
(2)	31	63	67																																												
MANPOWER			AVERAGE	ACCUM MANDAYS																																											
FIXED PRICE			20	1840																																											
COST PLUS FIXED FEE			11	1780																																											
PLANT FORCES																																															
ARCHITECT-ENGINEER																																															
DESIGN ENGINEERING OPERATION																																															
GE FIELD ENGINEERING																																															
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 being procured by J. A. Jones Construction Company.																																															

PROJ. NO. CAH-888	TITLE Biology Laboratory Improvements					FUNDING 60-h-1																																					
AUTHORIZED FUNDS \$ 420,000	DESIGN \$ 44,000 CONST. \$ 376,000	AEC \$ 359,500 GE \$ 60,500	COST & COMM. TO 2-18-62 ESTIMATED TOTAL COST		\$ 51,479 (GE) \$ 420,000																																						
STARTING DATES DESIGN 8-8-60 CONST. 7-10-61	DATE AUTHORIZED DIR. COMP. DATE 3-31-62	EST'D. COMPL. DATES DESIGN 3-31-61 CONST. 6-15-62	PERCENT COMPLETE																																								
ENGINEER FEO - JT Lloyd			<table border="1"> <thead> <tr> <th></th> <th>WT'D.</th> <th>SCHED.</th> <th>ACTUAL</th> </tr> </thead> <tbody> <tr> <td>DESIGN</td> <td>100</td> <td>NS</td> <td>100</td> </tr> <tr> <td>TITLE I</td> <td></td> <td></td> <td></td> </tr> <tr> <td>GE-TIT. II</td> <td>17</td> <td>NS</td> <td>100</td> </tr> <tr> <td>AE-TIT. II</td> <td>83</td> <td>NS</td> <td>100</td> </tr> <tr> <td>CONST.</td> <td>100</td> <td>100</td> <td>75</td> </tr> <tr> <td>PF</td> <td>1</td> <td>60</td> <td>60</td> </tr> <tr> <td>CPFF</td> <td></td> <td></td> <td></td> </tr> <tr> <td>FP</td> <td>100*</td> <td>85*</td> <td>74*</td> </tr> </tbody> </table>						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	75	PF	1	60	60	CPFF				FP	100*	85*	74*
	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	75																																								
PF	1	60	60																																								
CPFF																																											
FP	100*	85*	74*																																								
MANPOWER			AVERAGE	ACCUM MANDAYS																																							
FIXED PRICE				2280																																							
COST PLUS FIXED FEE																																											
PLANT FORCES																																											
ARCHITECT - ENGINEER																																											
DESIGN ENGINEERING OPERATION																																											
GE FIELD ENGINEERING																																											
SCOPE, PURPOSE, STATUS & PROGRESS																																											
This project provides additional space for biological research supporting services, and involves an addition to the 108-F Building. Conduit for Fire Alarm System is complete. AEC is reviewing contractors estimate for cost of other revisions to this system. Taping of sheetrock and plastering is proceeding on second floor. Air lines, water lines and steam lines were tested in equipment room. Doors are being installed on first floor. Fabrication is nearing completion on Radiation Handling Facility by Allied Engineering.																																											
*Based on Revised Schedule Approved by AEC.																																											

SEMI-MONTHLY PROJECT STATUS REPORT						HW- 71902	
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 2-28-62	
PROJ. NO. CAH-916	TITLE Fuels Recycle Pilot Plant					FUNDING 4-62-d-3	
AUTHORIZED FUNDS \$ 385,000		DESIGN \$ 385,000	AEC \$	COST & COMM. TO 1-28-62		\$ 321,952	
		CONST. \$ -0-	GE \$ 385,000	ESTIMATED TOTAL COST		\$ 5,000,000	
STARTING DATES	DESIGN 3-1-61	DATE AUTHORIZED 10-27-61	EST'D. COMPL. DATES	DESIGN 6-15-62	PERCENT COMPLETE		
	CONST. 9-1-62	DIR. COMP. DATE --		CONST. 7-1-64	WT'D.	SCHED.	ACTUAL
ENGINEER FEO-RW Dascenzo					DESIGN	100	47
MANPOWER FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT-ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING					TITLE I	11	100
					GE-TIT. II	89	41
					AE-TIT. II		
					CONST.	100	NS
					PF		
					CPFF		
					FP		

SCOPE, PURPOSE, STATUS & PROGRESS

This project is to provide a facility to perform a full scope of engineering tests and pilot plant studies associated with fuel reprocessing concepts.

A revised design schedule has been sent to AEC extending the design completion date to August 15, 1962 from June 13, 1962.

The four large tanks, scoped as transferred equipment, for use in an exterior waste tank vault will probably not be available to the project. As a result, the detail design is proceeding using six smaller existing tanks which will be located in two shielded vaults inside the building. This will improve control of radioactive contamination, simplify operations and reduce cost.

As of 2-15-62 a total of 108 drawings were issued for comment and 9 for approval on project.

PROJ. NO. CAH-917	TITLE Field Service Center-Atmospheric Physics					FUNDING 61-1	
AUTHORIZED FUNDS \$		DESIGN \$	AEC \$	COST & COMM. TO		\$	
		CONST. \$	GE \$	ESTIMATED TOTAL COST		\$ 154,000	
STARTING DATES	DESIGN 4-15-62*	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN 8-15-62*	PERCENT COMPLETE		
	CONST. 10-1-62*	DIR. COMP. DATE		CONST. 6-15-63*	WT'D.	SCHED.	ACTUAL
ENGINEER FEO-JT Lloyd					DESIGN	100	
MANPOWER FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT - ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING					TITLE I		
					GE-TIT. II		
					AE-TIT. II		
					CONST.	100	
					PF		
					CPFF		
					FP		

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facilities necessary to conduct atmospheric physics research and development programs.

The project proposal was submitted to the AEC on January 23, 1961.

There has been no change in the status of this project for some time.

*Based on AEC authorization by 3-15-62.

SEMI-MONTHLY PROJECT STATUS REPORT						HW - 72902	
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 2-28-62	
PROJ. NO.	TITLE					FUNDING	
CAH-922	Burst Test Facility for Irradiated Zirconium Tubes					62-k	
AUTHORIZED FUNDS	DESIGN \$	29,600	AEC \$	COST & COMM. TO		2-18-62	\$ 29,600
\$ 29,600	CONST. \$		GE \$	ESTIMATED TOTAL COST			\$ 228,000
STARTING DATES	DESIGN	11-7-61	DATE AUTHORIZED	10-23-61	EST'D. COMPL. DATES	DESIGN	5-15-62
	CONST.	6-15-62	DIR. COMP. DATE			CONST.	12-1-62
ENGINEER						PERCENT COMPLETE	
FEO-KA Clark							
MANPOWER							
FIXED PRICE							
COST PLUS FIXED FEE							
PLANT FORCES							
ARCHITECT-ENGINEER							
DESIGN ENGINEERING OPERATION							
GE FIELD ENGINEERING							
AVERAGE						WT'D.	SCHED.
ACCU MANDAYS							ACTUAL
						DESIGN	100
						TITLE I	NS
						GE-TIT. II	38.5
						AE-TIT. II	61.5
						CONST.	100
						PF	
						CPFF	
						FP	
SCOPE, PURPOSE, STATUS & PROGRESS							
This project will provide facilities to permit deliberate destructive testing of irradiated zirconium tubing. This will provide operating and tube life data not now available because of the limited operating history of Zircaloy-2 pressure tubing in reactors.							
Design is restarting in accordance with the recently approved criteria.							
Information meetings are being held with CE&UO to expedite design information, communications and approvals.							
A new design schedule is being prepared by CE&UO.							

PROJ. NO.	TITLE					FUNDING	
CAH-924	200-KW Induction Heating System - 306 Building					0290	
AUTHORIZED FUNDS	DESIGN \$	3,200	AEC \$	24,650	COST & COMM. TO	2-18-62	\$ 24,383
\$ 31,000	CONST. \$	27,800	GE \$	6,350	ESTIMATED TOTAL COST		\$ 31,000
STARTING DATES	DESIGN	5-1-61	DATE AUTHORIZED	3-27-61	EST'D. COMPL. DATES	DESIGN	10-15-61
	CONST.	12-4-61	DIR. COMP. DATE	2-28-62		CONST.	2-28-62
ENGINEER						PERCENT COMPLETE	
FEO-LF Higginson							
MANPOWER							
FIXED PRICE							
COST PLUS FIXED FEE							
PLANT FORCES							
ARCHITECT - ENGINEER							
DESIGN ENGINEERING OPERATION							
GE FIELD ENGINEERING							
AVERAGE						WT'D.	SCHED.
ACCUM MANDAYS							ACTUAL
						DESIGN	100
						TITLE I	
						GE-TIT. II	NS
						AE-TIT. II	100
						CONST.	100
						PF	
						CPFF	
						FP	
SCOPE, PURPOSE, STATUS & PROGRESS							
This project will provide a source of power for induction heating for R&D work in the 306 Building.							
Construction installation is complete, with the exception of replacement of a cam in a rotary station selector switch. As soon as the part is installed, operating tests will be made. The drawings are being as-built.							

DATE 2-15-62

GENERAL ELECTRIC CO. — Hanford Laboratories

SCOPE, PURPOSE, STATUS & PROGRESS

An invitation-to-bid for the lump sum portion of the construction has been mailed to prospective bidders. Bid opening is scheduled for 3-9-62 at 2:00 p.m. in the J. A. Jones Co. offices, 1256 Building, 3000 Area.

SEMI-MONTHLY PROJECT STATUS REPORT						HW- 72902	
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 2-28-62	
PROJ. NO.	TITLE					FUNDING	
CAH-936	Coolant Systems Development Laboratory					62-k	
AUTHORIZED FUNDS		DESIGN \$ 12,000	AEC \$ 78,000	COST & COMM. TO 2-18-62		\$ 14,486	
\$ 93,000		CONST. \$ 81,000	GE \$ 15,000	ESTIMATED TOTAL COST		\$ 93,000	
STARTING DATES	DESIGN 9-8-61	DATE AUTHORIZED 8-9-61	EST'D. COMPL. DATES	DESIGN 1-1-62	PERCENT COMPLETE		
	CONST. 4-20-62	DIR. COMP. DATE 10-31-62		CONST. 10-31-62	WT'D.	SCHED.	ACTUAL
ENGINEER					DESIGN	100	100
FEO-KA Clark					TITLE I		
MANPOWER					GE-TIT. II	100	100
FIXED PRICE					AE-TIT. II		
COST PLUS FIXED FEE							
PLANT FORCES					CONST.	100	NS
ARCHITECT-ENGINEER					PF		
DESIGN ENGINEERING OPERATION					CPFF		
GE FIELD ENGINEERING					FP		
AVERAGE					166		
ACCUM MANDAYS							
SCOPE, PURPOSE, STATUS & PROGRESS							
<p>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.</p> <p>Bids will be opened 3-14-62 at 2:00 p.m. in the 703 Building for the lump-sum construction of this facility.</p>							

PROJ. NO.	TITLE					FUNDING	
CGH-937	Safety Improvements to 231-Z Building					61-j	
AUTHORIZED FUNDS		DESIGN \$ 5,500	AEC \$	COST & COMM. TO 2-18-62		\$ 44,220	
\$ 45,000		CONST. \$ 39,500	GE \$ 45,000	ESTIMATED TOTAL COST		\$ 45,000	
STARTING DATES	DESIGN 7-12-61	DATE AUTHORIZED 6-29-61	EST'D. COMPL. DATES	DESIGN 9-29-61	PERCENT COMPLETE		
	CONST. 7-10-61	DIR. COMP. DATE 5-15-62		CONST. 2-20-62	WT'D.	SCHED.	ACTUAL
ENGINEER					DESIGN	100	NS
FEO-JT Lloyd					TITLE I		
MANPOWER					GE-TIT. II	NS	100
FIXED PRICE					AE-TIT. II		
COST PLUS FIXED FEE							
PLANT FORCES					CONST.	100	100
ARCHITECT-ENGINEER					PF		
DESIGN ENGINEERING OPERATION					CPFF	100	100
GE FIELD ENGINEERING					FP		
AVERAGE					4		
ACCUM MANDAYS					460		
SCOPE, PURPOSE, STATUS & PROGRESS							
<p>This project provides supplemental ventilation and installation of a fire detection system in the 231-Z Building.</p> <p>Fire detection system is complete.</p> <p>Building ventilation is being balanced. Final inspection was made February 16, 1962. The building was accepted with minor exceptions.</p>							

SEMI - MONTHLY PROJECT STATUS REPORT						HW - 72902																																					
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 2-28-62																																					
PROJ. NO. CGH-951		TITLE A-C Column Facility - 321 Building				FUNDING 0290																																					
AUTHORIZED FUNDS \$ 55,000		DESIGN \$ 5,000	AEC \$ -0-	COST & COMM. TO 2-18-62		\$ 1,884																																					
		CONST. \$ 50,000	GE \$ 55,000	ESTIMATED TOTAL COST		\$ 55,000																																					
STARTING DATES	DESIGN 1-30-62	DATE AUTHORIZED 1-12-62	EST'D. COMPL. DATES	DESIGN 4-1-62	PERCENT COMPLETE																																						
	CONST. 3-15-62	DIR. COMP. DATE 10-31-62		CONST. 10-31-62																																							
ENGINEER FEO - OM Lyso				<table border="1" style="width:100%; border-collapse: collapse;"> <thead> <tr> <th></th> <th>WT'D.</th> <th>SCHED.</th> <th>ACTUAL</th> </tr> </thead> <tbody> <tr> <td>DESIGN</td> <td>100</td> <td></td> <td>65</td> </tr> <tr> <td>TITLE I</td> <td></td> <td></td> <td></td> </tr> <tr> <td>GE-TIT. II</td> <td>100</td> <td>NS</td> <td>65</td> </tr> <tr> <td>AE-TIT. II</td> <td></td> <td></td> <td></td> </tr> <tr> <td>CONST.</td> <td>100</td> <td></td> <td></td> </tr> <tr> <td>PF</td> <td></td> <td></td> <td></td> </tr> <tr> <td>CPFF</td> <td></td> <td></td> <td></td> </tr> <tr> <td>FP</td> <td></td> <td></td> <td></td> </tr> </tbody> </table>					WT'D.	SCHED.	ACTUAL	DESIGN	100		65	TITLE I				GE-TIT. II	100	NS	65	AE-TIT. II				CONST.	100			PF				CPFF				FP			
	WT'D.	SCHED.	ACTUAL																																								
DESIGN	100		65																																								
TITLE I																																											
GE-TIT. II	100	NS	65																																								
AE-TIT. II																																											
CONST.	100																																										
PF																																											
CPFF																																											
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MANPOWER FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT-ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING				AVERAGE	ACCUM MANDAYS																																						
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.																																											

PROJ. NO. CGH-955		TITLE Reactivation of the H-1 Loop - 105-H Building				FUNDING 0490																																					
AUTHORIZED FUNDS \$		DESIGN \$	AEC \$	COST & COMM. TO		\$																																					
		CONST. \$	GE \$	ESTIMATED TOTAL COST		\$ 105,000																																					
STARTING DATES	DESIGN 4-1-62*	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN 8-15-62*	PERCENT COMPLETE																																						
	CONST. 7-1-62*	DIR. COMP. DATE		CONST. 12-1-62*																																							
ENGINEER FEO - OM Lyso				<table border="1" style="width:100%; border-collapse: collapse;"> <thead> <tr> <th></th> <th>WT'D.</th> <th>SCHED.</th> <th>ACTUAL</th> </tr> </thead> <tbody> <tr> <td>DESIGN</td> <td>100</td> <td></td> <td></td> </tr> <tr> <td>TITLE I</td> <td></td> <td></td> <td></td> </tr> <tr> <td>GE-TIT. II</td> <td></td> <td></td> <td></td> </tr> <tr> <td>AE-TIT. II</td> <td></td> <td></td> <td></td> </tr> <tr> <td>CONST.</td> <td>100</td> <td></td> <td></td> </tr> <tr> <td>PF</td> <td></td> <td></td> <td></td> </tr> <tr> <td>CPFF</td> <td></td> <td></td> <td></td> </tr> <tr> <td>FP</td> <td></td> <td></td> <td></td> </tr> </tbody> </table>					WT'D.	SCHED.	ACTUAL	DESIGN	100			TITLE I				GE-TIT. II				AE-TIT. II				CONST.	100			PF				CPFF				FP			
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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. The project proposal was submitted to the AEC February 21, 1962. *Based on AEC Approval by March 15, 1962.																																											

TEST REACTOR AND AUXILIARIES
MONTHLY REPORT - FEBRUARY 1962

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output was 1013 MWD for a plant efficiency of 51.7% and a total experimental time efficiency of 54.4%. Accumulated exposure to date is 5612 MWD. Additional exposure to date information is as follows:

MWD (1019 February)	5612 to date
Maximum UO ₂ fuel element	86.8 MWD or 1710 MWD/T
Average UO ₂ fuel element	57.8 MWD or 1130 MWD/T
Maximum Pu-Al fuel element	75.3 MWD or 45% burnup
Average Pu-Al fuel element	37.6 MWD or 25% burnup

The seventh refueling was accomplished on February 7, 1962. One Pu-Al test element was discharged at the planned exposure and retained in the basin for future PRP-CF measurements. Three new Pu-Al elements were charged on the outside of the Pu-Al region and the shim-free critical level was 68 inches. There were two minimum outages during the month in which three and two new Pu-Al elements were charged.

Table I shows the status of the test fuel elements at month's end.

PRTR Test Number	Tube Location	Element Number	Description	Date Charged	Date Discharged	Accumulated MWD
5	1653	1501	UO ₂ -Tubular	11/3/61		36.8
10	1051	1082	UO ₂ -Hot Swaged	11/3/61		20.5
10	1857	1067	UO ₂ -Vipac	11/3/61		22.9
13	1253	5092	Pu-Al Instrumented	12/3/61		32.3
13	1544	5093	Pu-Al Instrumented	12/3/61		28.5
13	1556	5095	Pu-Al Instrumented	1/17/62		24.1
13	1247	5051	Pu-Al Instrumented	1/17/62	2/7/62	13.1
13	1847	5052	Pu-Al Instrumented	1/17/62		21.9

D₂O and Helium losses were 2500 pounds and 144,500 scf, respectively.

There were six reactor scrams during the month. Three of these were manual scrams from low power levels as part of planned tests, one was a spurious flux monitor trip at a sub-critical level, one was a pressurizer high pressure trip during heatup, and one was a manual scram from full power level due to a valve failure in the helium pressurization system.

There was one major outage during the month. Significant outage work included inspection of six process tubes and four fuel elements, and annual testing of the light water injection system.

There were three short outages of approximately minimum downtime duration. The reactor was refueled on two of these outages representing the first time this activity has been accomplished successfully on a minimum outage.

Equipment Experience

Primary Pump #2 was replaced because of seal failure. The pump had been in operation about six and one-half months.

A complete Class A overhaul (electrical) was completed on the fueling vehicle.

The fission chamber in position 2L was replaced.

The S-80 valve (main steam valve) and its actuator were disassembled and inspected. The valve failed to close completely during shutdown. No apparent cause of the malfunction could be found.

Twelve helium sample valves in the hot shop glove box were replaced with valves of improved design.

Preventive maintenance required 439 man-hours or 8.3% of total craft effort.

Improvement Work (Significant items)

Work Completed:

- Shield Pump Seals
- Biological Shield Door Interlock with Fueling Vehicle

Work Partially Completed:

- Reflector Pump Seals
- New Helium Unloading Station Valves
- Safety Circuit Ground and Low Voltage Detector
- Outlet Nozzle Cap Modification
- Automatic Condensate Drain from HX-7 (Direct return of D₂O condensate)
- RTD Terminal Board Modification

Design Work Completed:

- Rerouting of Primary System D₂O Injection
- Dual Valves on Helium Stack Lines
- Enlarge Chemical Feed System (Secondary system)
- Chemical Feed Lines Flushing and Heating
- Modification of Storage Basin Crane
- Deactivate RX7-K2 from Startup Channel - eliminate potential for operating error
- Interlock between Charge-Discharge Machine Shroud Seat and Discharge Hoist

Design Work Partially Completed:

- Decontamination Facility

Process Engineering and Reactor Physics

Work on PRTR tests during the month included:

- PRTR Test Number 17 (Primary Pump Vapor Binding) Supplement No. 1 was performed and an interim report prepared. Supplement No. 2 was prepared.
- PRTR Test Number 25 (Calandria Temperatures After a Reactor Scram) Performance was completed and the final report prepared.
- PRTR Test Number 29 (Flow Monitor Changes During Reactor Operation) Performance was completed.
- PRTR Test Number 30 (Functional Test of S-80 Valve) Performance was completed and final report prepared.
- PRTR Test Number 32 (Shim Worth Measurement) Performance was completed.

Procedures

Revised Operating Procedures issued	21
Revised Operating Standards issued	22
Temporary Deviations to Operating Standards issued	2
Revised Process Specifications accepted for use	2
Maintenance manuals issued	1
Maintenance Procedures issued	4

A total program for preventive maintenance has been established. The program status follows:

Mechanical

Total Inspection Sheets	195
Completed Inspection Sheets	56

Electrical

Total Inspection Sheets	50
Completed Inspection Sheets	6

Measurement and Control

This work is being planned in terms of "information channels". These total 125 to 150. Currently, over half of these have been integrated into the PM framework; however, preparation of inspection sheets is not over 10% complete. Items for which inspection sheets were issued during the month:

Potentiometers	14 units
Controllers	10 units
Pressure Switches	25 units

One additional program was started to provide annual testing of all safety relief valves.

Drawings approved for as-built corrections total 210 - an increase of 39 during February. Drawings in engineering for approval of as-built corrections total 45.

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Drawings in drafting total 176. A reduction of about 60 during February. Drawings requiring as-built review total about 400.

Personnel training was as follows:

Reactor operating qualification subjects	289 man-hours
Specifications, Standards and procedures	58
Unit Motion	17
Total	364 man-hours

Personnel Qualifications at month-end:

Qualified Reactor Engineers	9
Provisionally qualified Reactor Engineer	1
Qualified Technicians	6
Qualified Technologists	18

Project Status

Fuel Element Examination Facility - Replacement parts for the damaged gear box cracked during heat treating at the vendor's plant.

PRTR Shipping Cask - The changes requested by CPD have been detailed and a work request issued for performance of the work.

Second Communication System - Drawings were approved during the month.

PRTR Water Control Laboratory - Equipment installation drawings were approved.

Plutonium Recycle Program Critical Facility

Project Status (Project CAH-842)

The overall project is an estimated 98% complete. Control rods were installed and the drives tested satisfactorily. One safety rod assembly was installed. The vendor was unsuccessful in attempts to clad two additional rods.

Performance of design tests was halted pending completion of additional work. Design tests for the instrumentation are approximately 75% complete. All others (Electrical, Fuel handling, and Thimble Coolant System) are approximately 10% complete. Difficulties were experienced with handling of the thimble tube. Hoses interfered with lock door operation.

Operation

One additional engineer was assigned to assist with operational start-up work. A training guide was prepared and work was begun on data sheets, qualification questions, and start-up tests. A total of 88 man-hours were devoted to training.

Fuel Element Rupture Test FacilityProject Status (Project CAH-867)

The overall project is an estimated 86% complete compared to 88% scheduled. Design and development was started on the cooling assembly for discharge operations. Design was started for shielding of ex-reactor components. Design of the basin clean-up system is essentially complete. Water plant work is approximately 67% complete. Concrete placement is approximately 95% complete.

Operation

A master flow diagram was prepared. Other work underway included valve book handbook inserts, training aids, and Instrument manual preparation.

GAS COOLED POWER REACTOR PROGRAMGas Cooled LoopProject Status (Project CAH-822)

Directive Modification #6 was received which increased funds to a total of \$1,170,000. The project is 91% complete based on revised authorization. Negotiations continued for a heater. Progress on the primary blowers was nil. Bristol-Siddesley (England) was advised to indicate continuous operating capabilities of blowers at this time with intent to renegotiate contract and request delivery of reduced capacity blowers.

Operation

A new master flow diagram was prepared. Work began on instrument manual, PRTR handbook inserts, and electrical manual. All visual aids, except for the heater, were completed.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 21,094 hours. This includes 14,173 hours performed in the Technical Shops, 4,515 hours assigned to Minor Construction, and 2,606 hours assigned to off-site vendors. Total shop backlog is 18,930 hours, of which 60% is required in the current month with the remainder distributed over a three-month period. Overtime hours worked during the month was 4.2% (749) of the total available hours.

Distribution of time was as follows:

	<u>Man-Hours</u>	<u>% of Total</u>
Fuels Preparation Department	3,805	18.04%
Irradiation Processing Department	2,288	10.85%
Chemical Processing Department	339	1.61%
Hanford Laboratories Operation	14,644	69.42%
Construction Engineering & Utilities Operation	18	.08%

Requests for emergency service increased requiring a 4.2% overtime ratio compared to a 2.8% ratio for the previous period.

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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.

<u>INVENTOR</u>	<u>TITLE OF INVENTION OR DISCOVERY</u>
L. L. Ames, Jr.	Synthesis of an Acid-Resistant, Cesium-Selective Zeolite or Molecular Sieve
G. B. Barton	Glasses Containing Fission Product Cesium (HW-72774)
G. B. Barton	Glasses Containing Fission Product Strontium (HW-72775)
E. M. Sheen	Electronic Circuitry, A Cable-Driving Pulse Preamplifier (HWIR-1481)
J. F. Fletcher	Self-Control of Nuclear Reactors Through Thermally-Induced Variation of Reflector or Moderator Density
H. E. Hanthorn	Control of Nuclear Reactors

Manager, Hanford Laboratories

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END

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