HW--82001

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**HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT**

APRIL 1964

MAY 15, 1964

**HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON**

**GENERAL ELECTRIC
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HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT
APRIL 1964

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By Authority of CG-PR2

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Compiled by
Section Managers

May 15, 1964

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

PRELIMINARY REPORT

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Table I - Hanford Laboratories Force Report

Date: April 30, 1964

	<u>At Beginning of Month</u>		<u>At Close of Month</u>		<u>Total</u>
	<u>Exempt</u>	<u>Salaried</u>	<u>Exempt</u>	<u>Salaried</u>	
Chemical Laboratory	147	123	150	121	271
Reactor & Fuels Laboratory	210	195	210	194	404
Physics & Instruments Laboratory	109	81	111	81	192
Biology Laboratory	44	65	44	64	108
Applied Mathematics Operation	21	6	22	6	28
Radiation Protection Operation	44	94	44	95	139
Finance & Administration Operation	140	116	132	116	248
Programming Operation	21	5	22	5	27
Test Reactor & Auxiliaries Operation	63	309	63	309	372
General	<u>4</u>	<u>5</u>	<u>3</u>	<u>5</u>	<u>8</u>
TOTAL	<u>803</u>	<u>999</u>	<u>801</u>	<u>996</u>	<u>1797</u>

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BUDGET AND COST SUMMARY

April operating cost totaled \$2,905,000, an increase of \$404,000 over the previous month; fiscal year-to-date costs are \$26,914,000, or 80% of the \$33,509,000 control budget. Hanford Laboratories' research and development costs for April, compared with last month and the current control budget, are shown below:

(Dollars in thousands)	COST				
	Current Month	Previous Month	To Date	Budget	% Spent
HL Programs					
02	\$ 101	\$ 105	\$ 856	\$ 1 180	73
03	4	7	249	250	100
04	1 307	978	11 665	14 085	83
05	135	101	1 165	1 438	81
06	296	272	2 708	3 352	81
08	31	31	187	290	64
	<u>1 874</u>	<u>1 494</u>	<u>16 830</u>	<u>20 595</u>	<u>82</u>
Sponsored by					
NRD	105	149	1 538	1 798	86
IPD	23	18	420	490	86
CPD	<u>158</u>	<u>146</u>	<u>1 413</u>	<u>1 778</u>	<u>79</u>
Total	<u>\$2 160</u>	<u>\$1 807</u>	<u>\$20 201</u>	<u>\$24 661</u>	<u>82</u>

Authorized funds for the nonproduction research and development programs assigned to Hanford Laboratories were adjusted as follows:

	<u>Increase</u>
<u>04 Program</u>	
Plutonium Recycle Program	\$ 20 000
Fueled Graphite Studies	220 000
EBWR Program	<u>60 000</u>
Total	<u>\$300 000</u>

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

An N-Reactor outer fuel component that experienced 3.6 vol% fuel swelling at 3100 Mwd/ton exposure in KER Loop 3 is being examined in the

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Radiometallurgy facility. Microscopic examination shows the presence of small voids that appear to be formed by grain boundary tearing.

The measurements of prototypic N-Reactor outer fuel elements from two KER loop tests (K-3-18 and K-4-15) have been completed. The swelling of the elements calculated from these data are in good agreement with the swelling data obtained from direct density measurements. No severe or unexpected dimensional changes were observed.

Two N-Reactor single tube fuel elements with fluted outer surfaces have successfully accumulated exposures of approximately 2000 Mwd/ton during irradiation in the ETR M-3 loop. The volume increase is 1.6%.

Two charges of KSE-5 single tube fuel elements are being irradiated in KER Loops 1 and 2 to provide a comparison of swelling behavior of alloyed and unalloyed fuel over a range of exposures and fuel operating temperatures.

Postirradiation examination of target elements that were irradiated in KER Loop 2 has continued. Analysis shows a gas content of 26.5 cm^3 of gas (helium + tritium) per cm^3 of target core with no indication of tritium in the Zircaloy-4 can or annulus.

Experimental target core material has been successfully extruded from Mg-14 wt% Li-2 wt% Al alloy.

Zircaloy-2 clad rods, 0.6 in. diameter, have been fabricated from several experimental uranium alloys. Evaluation of these rods is under way, and additional alloy billets are being prepared.

On the basis of free energy measurements, several experimental uranium base alloys have been shown to have a low corrosion potential.

Autoclave and thermal cycling tests have been performed on resistance projection welded fuel elements. No evidence of damage was observed metallographically.

Approximately 100 tests have been performed to determine friction factors for various fuel elements-process tube combination. Analysis of the test data is under way.

Only slight corrosion (0 to 0.2 mil) was found in the crevices under N-Reactor fuel element supports welded to electrically heated Zircaloy-2 clad specimens after exposure to recirculating water at 314 C, pH 10, with NH_4OH for 56 and 120 days at a heat flux of 200,000 Btu/(hr)(ft²).

Following decontamination, no rise in cladding temperature of either thermocouple element in KER-1 was noted during 43 days of operating at pH 10 with NH_4OH . A red-brown crud deposit was found on the discharged elements.

A corrosion rate of 0.3 mil/year was measured for carbon steel samples in pH 10 ammoniated water at 300 C.

Transient experiments investigating loss-of-coolant accidents for N-Reactor process tubes were continued. In tests simulating an inlet connector rupturing during the primary system heatup, preliminary results showed no significant temperature increases occurred in the simulated fuel element during the transient.

An experiment was performed on an electrically heated, simulated fuel column to investigate steam formation when cold water is valved into an uncooled fuel column heated to 1500 F, as might occur in N-Reactor upon reestablishment of coolant flow following a loss of coolant. An unstable situation occurred upon flow reestablishment with large oscillations in both flow rate and surface temperature. It was concluded that similar oscillations could be expected in N-Reactor following such an incident.

Experiments were performed to investigate the pressure surges associated with the hydraulic seating of the fuel elements in a present production reactor. The results indicated, that under certain conditions, surges can be produced during the hydraulic seating processes that would be sufficient to cause a Panellit pressure trip.

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Recent tests have shown that KS graphite monitors exposed simultaneously with TSGBF monitors have a sevenfold higher weight loss as compared to a two to threefold difference determined in laboratory experiments.

F-Reactor monitors showed acceptably low burnout rates.

The first results have been obtained for large graphite blocks exposed to a water oxidizing atmosphere in the simulated N-Reactor loop. Surface oxidation rates were tenfold greater than rates observed at positions greater than 3/8 in. below the surface.

The dual purpose tritium evolution and zirconium-gas atmosphere capsule was charged and has been in operation since April 5, 1964. Temperature coefficients for tritium evolution will be determined. Encapsulated lithium ceramics are being irradiated for evaluation. The zirconium-gas atmosphere capsule is operating satisfactorily at design temperature.

The rate of oxidation of an irradiated lithium-aluminum alloy in high temperature water was measured. The gas retention in the oxide is being determined.

The zirconium-graphite compatibility loop completed 1 month of satisfactory operation. Graphite oxidation data were obtained by GRAD at 1000 ppm water in helium. Confirming inhibition of gas phase hydriding, Zircaloy-2 samples, exposed simultaneously, oxidized at the expected rates and absorbed only hydrogen from the water corrosion reaction.

Oxidation rates of Zircaloy-2 and crystal bar zirconium in CO_2 at 350 and 450 C (662 and 842 F) have been measured for 143 days. These data show the oxidation rate in CO_2 to be considerably less than in dilute H_2O vapor. Hydrogen pickup data for zirconium indicate that a major fraction of weight gain may be due to traces of water in the CO_2 .

Burst tests have been made on slotted pressure tube specimens instrumented with strain gages. These show a stress intensification factor

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of 2.4 at the tip of the slot and an as-yet unexplained compressive hoop stress in the area of the tube immediately adjacent to the midpoint of the slot length.

Long term irradiation of N-Reactor graphite continues satisfactorily.

Revision of the "Foil" Fortran program is complete and is reported in HW-81367.

Swaged UO_2 fuel elements exceeded 5840 Mwd/ton in the PRTR. Incrementally loaded UO_2 -0.48 wt% PuO_2 exceeded 5680 Mwd/ton. Elements containing impacted UO_2 -1 wt% PuO_2 exceeded 2280 Mwd/ton.

A seven-tube water tube (inverted cluster) fuel element was constructed for irradiation in PRTR.

Sixty-two percent of the EBWR Zircaloy-2 cladding was rejected.

Eight EBWR prototypic fuel rod capsules were discharged from MTR-ETR at exposures of 1.6×10^{20} fissions/cm³.

Three UO_2 -1 wt% PuO_2 PRTR fuel elements were completed. Three 19-rod clusters for physics tests also were completed.

The Dynapak machine operated routinely without a misfire or broken can.

Fabrication of fuel elements for PRCF (Plutonium Recycle Critical Facility) is progressing satisfactorily.

Fabrication of PCTR (Physics Constants Testing Reactor) fuel elements is complete.

A flow test was performed on a model of a proposed PRTR "inverted cluster" fuel element (Mk IXB) to determine the pressure loss characteristics and flow distribution. The results of the experiment confirmed earlier theoretical predictions that an imbalance exists between heat-transfer surface area and coolant flow rate in this particular design.

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The results of recent analyses of the Doppler delay problem in mixed oxide fuel were presented to General Electric Technological Hazards Council. On the basis of these analyses, GETHC has approved the use of this fuel in EBWR.

Two decontaminating solutions (EDTA-ammonia-hydrazine and oxalic-peroxide-peracetic) were found to be effective in dissolving PuO_2 in laboratory tests.

A dilute APAC decontamination process was tested and found to be ineffective in removal of rupture debris from a metallic uranium fuel element.

Testing of the second generation shim rod in the environmental shim rod test facility continued through April without manifestations of rod sticking or other problems which have plagued the present PRTR shim rods.

Additional transducers installed on three PRTR inlet jumpers indicate maximum peak-to-peak amplitude of jumpers of 4.0 to 4.5 mils, compared to previously measured amplitudes of 7 to 8 mils which occurred during the period that a primary pump motor bearing failed.

A specimen from PRTR tube 6079 was burst tested at room temperature. The ultimate hoop strength of this specimen was approximately 119,000 psi. A pressure of 11,400 psi was required to burst the specimen. A significant amount of plastic deformation was produced before the burst; the burst origin was in this region of plastic deformation.

PuO_2 was explosively melted by passing a high current pulse through a direct resistance heated sample.

Irradiated specimens of PuO_2 and PuN are being examined after exposures of about 20×10^{20} fissions/cm³.

A high temperature vacuum pushrod dilatometer was put in operation.

Apparatus for measuring the thermionic work function and emission constant using a contact potential difference technique was completed.

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Radiation self-damage of PuC seems to be saturating after 975 days.

A computer code for determining optimum particle size distributions for maximum compacted density in cylindrical and annular configurations has been developed in cooperation with Applied Mathematics.

Less than one-half the total fission gas was released from an insulated UO_2 capsule that operated, with all the fuel molten, at 2700 C cladding surface temperature.

Pneumatically impacted ThO_2 ($5.5 \text{ m}^2/\text{g}$) released 0.53 cc gas per gram at 2×10^{20} nvt. Sol-Gel ThO_2 ($0.068 \text{ m}^2/\text{g}$) released only 0.018 cc/g at 2×10^{20} nvt.

Central fuel temperature of an instrumented ThO_2 fuel element increased from 480 C to 1050 C during a 30-day irradiation in the MTR.

Pneumatically impacted UN was irradiated in the MTR.

Highly loaded UO_2 -stainless steel cermet cores were clad with stainless steel tubing and fabricated into tubular sections using the "one shot" pneumatic impaction process.

Tungsten coated spherical UO_2 and UO_2 -tungsten powder blends were fabricated into tungsten clad UO_2 -tungsten cermets with powdered tungsten end caps.

UO_2 -niobium cermet cores were clad with niobium tubing and sealed with niobium powder end caps using the pneumatic impaction process.

A good end closure weld was achieved with one of four French beryllium cladding tubes now being studied.

The high temperature Brew furnace was successfully heated to 2800 C in flowing hydrogen atmosphere.

Single crystals of UO_2 were shipped to several sites for basic studies, and high surface area UO_2 was received through EURATOM.

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The melting points and microhardness of Nd_2O_3 and Sm_2O_3 have been determined.

The absorption spectra for single crystal ThO_2 was measured between 0.3 and 2.5 μ , showing a characteristic absorption edge at 3.9 ev.

Solid fission products were transferred from an irradiated sample to an unirradiated one along a potential gradient.

Additional information on hydrogen pickup for Zircaloy-2 samples irradiated in the ETR G7 loop confirm results previously reported. Again, no consistent relation between fraction pickup and flux or cold work was observed.

Oxidation rates of arc cast molybdenum in the fully annealed and stress relieved conditions were similar in 1 μ oxygen partial pressure at 1000 C. Rates were linear, about 0.1 mil/day.

Four Inconel X-750 tensile and stress-rupture specimens were discharged from the ETR and will be scheduled for testing. The specimens had been given an experimental heat treatment to provide finely dispersed in-gran γ' precipitate and reduced-grain-boundary carbide precipitate.

To provide a high resistance to buckling, specimens for low cycle fatigue studies were designed with an hourglass shape rather than a cylindrical one.

Tensile specimens of Hastelloy X-280, Inconel 625, Incoloy 800, and Inconel 600, irradiated at 50 C (122 F) to exposures of 9.8×10^{19} , 4.61×10^{20} , and 7.64×10^{20} nvt ($E > 1$ Mev) were tested at room temperature. Results showed an increase in strength with small decreases in uniform elongation.

Three general swelling capsules containing high purity uranium samples having U^{235} contents of natural, twice natural, and four times natural enrichment were opened in Radiometallurgy and the specimens were successfully recovered. No obvious effect of burnup rate was evident. Preliminary metallographic examination of the specimens revealed crystallographically aligned porosity typical of the higher irradiation temperature, and grain boundary tearing typical of the lower irradiation temperature.

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Irradiation of H_2 -CO and CH_4 -CO₂ mixtures at ambient temperatures showed smaller yields of CH_4 than irradiation at 550 C. This finding is attributed to the stability of carbon suboxide polymers that at the higher temperature decomposes to carbon which can then react with hydrogen to form CH_4 .

Radiation of carbon with N_2O at 30 C produces N_2 , CO, CO₂, and NO. The extent of reaction increases with surface area of the carbon.

Long term irradiations of EGCR graphite continues satisfactorily. Revision of the method of calculating exposure has resulted in exposure values that are 60% of those formerly reported.

Using the high contraction rate of certain carbons to produce a tensile stress in graphite specimens has resulted in failure of the specimens at a strain of 0.25 to 0.3% at 625 C.

The thermal expansion behavior of a series of monitored graphites was found to be relatively insensitive to the temperature of graphitization.

CSF graphite irradiated at 600 C and annealed to 2500 C contracted at the very much lower rate in a subsequent irradiation than did an unannealed control specimen. The behavior is opposite of that observed with lampblack-type graphites.

Analysis of the dimensional changes in boronated graphites irradiated to 10^{20} nvt $E > 0.18$ Mev disclosed larger changes than would be observed in nuclear graphite. The behavior of 7 wt% gray boronated graphite was anomalous, showing expansion in the direction parallel to the extrusion axis. The compressive strength of unirradiated black boronated graphite is about 4500 psi; of gray, about 2500 psi.

Irradiation of three tubular Zircaloy-2 clad, Th-2.5 wt% U fuel elements continued in the ETR P7 loop. Exposure is currently 6200 Mwd/ton with a maximum measured fuel volume increase of 0.8%.

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Irradiation of metallic uranium fuel containing a fine dispersion of uranium carbide continued in the ETR. The fuel sample, that is operating at a maximum temperature of 550 C, has reached an exposure of 0.20 at. % burnup.

Ten Th-U base alloys have been fabricated into Zircaloy-2 clad coextruded rod stock 0.525-in. diameter. Specimens of these alloys for defect testing are being prepared.

Four dispersion hardened thorium-base alloys have been fabricated into alloy sheet 0.048-in. thick. Fabrication was achieved by the combined forging and rolling of stainless steel jacketed specimens.

Little corrosion was evident on aluminum-clad fuel elements discharged from C-1 Loop after 25 days of exposure at 260 C, neutral pH. Another test at pH 4.5 has been started.

An experimental program was started to determine the difference in the boiling burnout characteristics of a 19-rod bundle fuel element operated in the horizontal and in the vertical positions. Where identical coolant flow rate, outlet pressure, and inlet pressure existed, preliminary data comparisons revealed that up to 26% higher boiling burnout heat fluxes were obtained for the vertical model.

Molybdenum single crystals deformed in tension after irradiation have been oriented and sectioned parallel to a primary slip plane for electron microscope examination.

Twelve single crystals of molybdenum that will provide data on length changes and lattice parameter changes as a function of irradiation dose have been encapsulated and submitted for irradiation. The specimens were divided into three groups for goal exposures of 1×10^{19} , 5×10^{19} , and 1×10^{20} nvt.

Lattice parameters have been measured for as-grown rhenium single crystals. Values obtained are $A = 2.7604A$, $C = 4.4599A$, and $C/a = 1.6157$. They are in agreement with published values.

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The pole figure goniometer was installed on the XRD-5 diffractometer in a glove box for use in the study of deformation textures in plutonium. The initial activity has been concerned with establishing X-ray absorption corrections.

Installation of the plutonium research glovebox in Room 46 in the 231-Z Building has been completed.

It was experimentally demonstrated that press fitted zirconium caps adequately sealed PRTR fuel rod sections whose end closures had been removed. The capped sections are suitable for shipment without further containment.

Procedures were established for polishing pure beryllium, for etching uranium-zirconium alloys, and for polishing low level irradiated samples with the Buehler Automet polisher.

Material testing in support of the HTLTR continued. Fuel cladding, control rod, thermocouple, and structural materials were exposed to a simulated reactor environment of nitrogen gas-graphite at 1200 C for 840 hr.

EBWR rods are being compacted satisfactorily by the resonant plate vibration technique.

An approximately 1-in. cube of 1/8-in. hexagonal honeycomb grid having 9/32-in. solid tungsten end blocks for electrical contacts was fabricated by pneumatic impaction.

A tungsten circular grid consisting of eight concentric rings connected by radial webs was fabricated by pneumatic impaction.

A small test grid utilizing molybdenum mandrels to more closely match thermal expansion coefficients was fabricated. This grid was crack-free by visual examination. A large cermet grid that used alumina-coated mandrels was also impacted and was crack-free by visual examination.

Special aligning electrodes were designed and fabricated in preparation for cermet welding development to provide continuous clad at welded joints.

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Thermal cycling studies were conducted at high pressure on cermet containing 20 vol% UO_2 in tungsten.

Tungsten- UO_2 dispersions have been successfully produced by coprecipitation from ammonium-paratungstate and uranyl nitrate solutions followed by calcination and hydrogen reduction. Dispersions containing 10, 20, and 30 vol% UO_2 are being prepared for evaluation studies.

The direct extrusion of tungsten and molybdenum powders show promise as a means of minimizing the rough innerface problem encountered during coextrusion of tungsten honeycomb using wrought components. Studies are under way to establish extrusion conditions that will yield an acceptable product directly from powders.

Materials and equipment required in the fabrication of UO_2 fueled tungsten plates are being ordered.

Discharge of 1 ton of vibrationally compacted irradiated thoria on April 1, 1964, marked successful completion of the first large-scale thoria exposure at Hanford. Two additional tons were discharged on April 15, 1964, and replaced with the fourth ton, which will complete the "pilot" run of 4 tons.

2. Physics and Instruments

The "hot" physics test irradiation in the fully loaded N-Reactor has been completed. The special fuels have been disassembled and the activity of the detector pins has been counted. Data reduction is under way.

A simplified model of the N-Reactor plant has been simulated on the new analog computer. The simulation includes a single node reactor kinetics model, and models for primary pumps, steam generator, dump condenser, and surge tank. Automatic control systems include control for the primary coolant flow, steam generator water level, steam header pressure, surge tank level, and condensate header pressure. The simulation can operate in real time or at speeds ten times faster.

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Operational improvement was achieved for the N-Reactor low-level neutron monitoring system with the development and inclusion of specific pulse discrimination and amplifying circuits. Individual tests were satisfactorily completed on the 24 gamma energy spectrometers for the N-Reactor fuel rupture monitoring system.

Experimental investigations at C-Reactor storage basin on the use of gamma spectrometry techniques to determine the cooling time of irradiated fuels continued to produce satisfactory results.

Modifications and improvements are being made to the ultrasonic inner probe test for characterizing extrusion billets for the N-Reactor fuels process. Eddy current billet tests have been discontinued.

Development of a cap-to-core bond test for the wedge or "Chevron" end closure on experimental N-Reactor fuels is proceeding. A low frequency angled ultrasonic beam is used and lack of return signal from the end correlated with lack of bonding. Further investigation is under way to try to check the inner bond region from the outside, thus eliminating a probe.

Evaluation tests on the N-Reactor fuel enrichment tester have shown that the instrument can differentiate 0.95% from 1.25% U^{235} enrichment in billets very reliably even with "green" uranium. HTR reactivity tests are being discontinued.

The X-ray fluorescence test equipment for detecting uranium contamination on fuel ends has been installed in the coextrusion shop and is being tested and evaluated. Modifications in aperture design have reduced background variations significantly. Minor line voltage variations were determined to be the cause of drift and have been overcome by precise regulation. Preliminary tests have shown a considerable portion of 50 test pieces to have uranium contamination. A production test to evaluate the results and measure throughput and reliability is being designed to start immediately.

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Laboratory measurements on a range of hydrided Zircaloy test samples have confirmed the effects observed on the hydride tube test. The ultrasonic longitudinal wave velocity increase with hydride is almost ten times the transverse velocity increase. This establishes the test basis for ultrasonic hydride detection and an invention disclosure is being written on this application. An ultrasonic rotating probe mechanism is being fabricated and is about 30% complete.

Poor reproducibility in bond testing irradiated AlSi fuels was determined to be caused by transducer and transducer alignment variations. These are being reduced. Surface roughness has caused a high background noise level and made detection of small voids impractical. After completion of core cracking tests and a further evaluation test the equipment development will be completed.

Improved electronic gating circuitry to discriminate against variations in cores or in cladding thickness was developed for the AlSi fuel bond tester and evaluated with satisfactory results. Reproducibility tests, however, indicate an amplifier drift is causing $\pm 10\%$ change in overall response. Prototype circuitry development is proceeding.

Criticality experiments were begun with an 11.5-in. diameter stainless steel sphere with provisions for either a water or concrete reflector. Pu nitrate solutions under study are in the concentration range of 200 to 400 g Pu/l. Neutron flux distributions were determined from foil irradiations for evaluating extrapolation lengths and material bucklings of the water and concrete reflected critical assemblies.

Pulsed neutron source experiments and reactor noise measurements were performed concurrently with the above critical experiments for determining neutron lifetime and k_{eff} .

A series of Monte Carlo and multigroup diffusion calculations were made to evaluate the nuclear safety of the proposed design of the HAPO-V

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shipping cask for slightly enriched irradiated fuel elements. The calculations show that for uranium enrichments ≤ 1.18 wt%, the k_{eff} of the proposed shipping cask will be less than about 0.9.

The continued existence of discrepancies in the slow neutron scattering law for 95 C H₂O indicates a subtle difficulty with a portion of these results.

Major improvements have been made in the computing programs that process fast-neutron total cross section data and results.

A preliminary value of the Pu²⁴⁰/Pu²³⁹ atom ratio has been obtained and reported for a NBS-Pu Isotopic Standard sample.

Work continued on critical experiments in the PRCF using EBWR fuel. Rod calibrations were obtained, yielding 13.1 mk per rod (3) and 4.6 mk per control sheet. Reactor noise recordings were made. A report, "Experiments with PuO₂-UO₂ Fuel Elements in Light Water," was prepared for the Physics Research Quarterly Report.

Reactor physics computer code development continued at a high level during the month. Significant progress was made in documentation of RBU, modifications of PLOTCH to plot cross sections from the RBU tape, review and updating of the RBU Basic Library, debugging of parts of BARNS II, and the release of ZODIAC (burnup code) for production use.

High temperature materials tests for the HTLTR continued. A graphite heating element achieved 600 hr of heating at 1400 C with no indication of failure. The fifth of a series of high temperature tests was completed at 840 hr at 1200 C in a nitrogen atmosphere; significant data on material instability and compatibility at high temperatures were obtained.

Calculations of the k_{∞} to be expected in the proposed Phoenix fuel-PCTR experiments were completed for several possible conditions under which the experiments could be run.

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Multiplier phototube gain shifts were measured on six detector probes for use in the PRTR fuel element rupture monitor. Further testing was planned for two more tubes of a different type to determine if stability can be achieved. Progress was achieved on engineering improvements to the PRTR exhaust monitor; the required detection level for Cs^{138} , Rb^{88} , and I^{131} was determined to be $5 \times 10^{-6} \mu\text{Ci/cc}$.

Chemical separation and mass spectrometer analyses were completed on all plutonium regenerating and uranium flux monitoring detectors for the extended lifetime neutron flux monitor program. Cross section information appears to limit correlation accuracy; however, acceptable agreement was obtained between experimental data and calculated results. A summary of the regenerating detector investigations was prepared for use at the forthcoming Geneva conference. Development work was accelerated to provide a mechanically stable beta current in-core flux monitor.

Initial test results, at KW-Reactor, appeared promising for the microwave neutron flux monitor that employed a nitrogen-filled resonant cavity for the test. Results indicate a frequency shift occurred with neutron induced ionization of the gas in the cavity as expected, although the results were not conclusive because of dimensional changes of the cavity that resulted from heating.

Multiparameter eddy current test principles are being applied to the nondestructive testing of tubing with special emphasis being placed on the detection of hydride in N-Reactor process tubes. A technique for using a tuned air core transformer with a rotating secondary winding for coupling energy to a helical scanning test coil was demonstrated. A magnetic saturation technique was applied in the inspection of sheathed thermocouples to reduce the undesired signals resulting from the magnetic alumel wire.

Test standards of various metals have been fabricated for use in ultrasonic attenuation studies. Ultrasonic diffraction effects are being studied by measuring the diffraction field from transducers and from small dissimilar

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objects in prepared samples. Information from these tests will be used to augment the analytical study to be carried out on the effect of diffraction and attenuation on pulse shape and propagation.

The effect of rough surfaces on the Zircaloy hydride detector was evaluated. These tests indicated that roughness of 125 μ in. and gouges of 0.005 in. gave approximately the same signal as 400 ppm of hydride. It is not expected that the N-Reactor process tubes, within which this detector is to be used, will have roughness or gouges of this magnitude.

Approximately 3000 fuel sheath tubes have been inspected to date with the prototype ultrasonic tester. This large number of inspections has provided experience with the operation of the unit and has pointed out potential wear points that occur in the mechanical system. With the exception of "O" rings, which are purposely chosen to wear, the tester should be maintenance free over long periods of operation.

A laboratory mockup of an animal physiological function and dose telemetry system is being developed. It will include thyroid (I^{131}) dose measurement, respiration rate, and temperature determination channels. Development testing progress also was achieved on solid state circuitry to automatically control air and smoke inhalation for animal biological studies. Most of the required circuitry was completed. The alpha energy analysis air monitor, developed for Biology Laboratory radionuclide inhalation experiments, was completed and final laboratory testing began.

Several experimental scintillation detector probes were completed for field studies of contamination problems with low energy beta emitters, such as Pm^{147} . The probes can be used directly with HAPO solid state radiological monitoring instruments.

Results of the whole body counting at Anaktuvuk Pass, Alaska, in March showed the average Cs^{137} body burden of the Eskimos had increased 28% since January. This confirmed the increase shown by urinalysis results.

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Radiological Physics, Radiological Chemistry, and Inhalation Toxicology made a study of the inhalation of dust and radioactivity at two uranium mines in Colorado with the Hanford Mobile Whole Body Counter.

Calorimetric measurements of the half-life of Pm^{147} were completed. The result was in excellent agreement with that found by the Hanford Chemical Laboratory.

Analysis of turbulence data for atmospheric dispersion studies will be facilitated by a program established on the IBM 7090 computer for determination of the power spectral density distribution of wind velocity fluctuations.

Design and fabrication was performed on a special low resistance measurement instrument used to measure beryllium preweld resistance. Correlation was being sought on weld quality versus the resistance.

A U^{235} fuel enrichment monitor successfully completed all laboratory tests and was delivered for routine use in metal development laboratories.

An ultrasonic test has been developed to inspect both the tubing and the small ribs on 3 in. diameter Zircaloy tubing for a new fuel element. The new test can be applied at any stage during manufacture and also after irradiation.

Strain gages have been applied on critical areas on the PRTR rupture loop to measure both strains and vibration during subsequent phases of operation. A weld failure recently occurred in this area.

Instrumentation was provided for a successfully completed program that measured the electrical resistivity and the density simultaneously during phase changes of alpha plutonium.

A novel pressure gage that measures both positive and negative pressures within waste storage tanks has been developed. The gage, which does not compromise tank integrity, is simple in design.

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Deterioration of the wood in utility line poles is being measured as a change in the restoring force of the pole. Data, accumulated in the field, offer encouraging results that may lead to a practical nondestructive test to detect poles prone to failure.

3. Chemistry

Programmed heating and quenching tests of fully irradiated N-Reactor fuel elements yielded encouraging evidence that cladding integrity is high enough to prevent exposure of the fuel during a postulated 5 min delay in operation of the back-up coolant system.

No significant changes were evident in the areal extent and concentrations of radiocontaminants in the ground water during the past month.

Zeolites may be useful in reactions involving the neutron irradiation of adsorbed cations. This belief is based on the lack of a detectable change in cation selectivities, capacities, and structures of two synthetic zeolites (with silica-to-alumina ratios of 2.8 and 10) irradiated at 3.16×10^{18} nvt and 3.14×10^{19} nvt, respectively.

Laboratory experiments indicate that the addition of one part of a solution of cesium phosphotungstate dissolved in 2M NaOH to 11 parts of Purex alkaline supernate is readily accommodated in the Linde AW-500 cesium recovery operation without significant effect.

Recent laboratory studies have pointed the way to a significant advance in the technology for plating nickel onto uranium metal. A 40% reduction in necessary plate thickness has been achieved using an $\text{HCl-H}_2\text{SO}_4$ etching solution in the uranium surface preparation procedure, rather than the standard $\text{HCl-H}_3\text{PO}_4$ etch.

Further laboratory-scale experiments to evaluate possible ion exchange resin hazards have been performed. Gas analyses have shown that the gas produced by nitric acid degradation of the resin is nonflammable. Experiments

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on the degradation of Permutit SK in nitric acid above 100 C have given a variety of results, from complete dissolution of the resin (at 200 C) to an explosively rapid reaction (above 170 C) when 40 ml of resin covered with Redox feed was heated rapidly in an 80 ml bomb.

Investigations of the chemical-thermal stability of anion exchange resins have been started in plant-scale equipment. Early experiments indicate runaway conditions can occur at temperatures as low as 130 C.

A pyrochemical processing scheme for 3% UO_2 -97% ThO_2 fuels has been tested on a 1 kg scale. The scheme involves dissolution in a chloride melt, precipitation of the thorium as ThO_2 , and recovery of the uranium as a solid solution of UO_2 in ThO_2 by electrodeposition.

Started this month were theoretical studies of the gamma dose rates associated with Pu^{238} , the probable contribution from Pu^{236} impurity, and the possibility of minimizing the Pu^{236} content by special irradiation procedures. The calculations show that the gamma activity from 1 ppm of initial Pu^{236} would equal that from the Pu^{238} itself after about 15 months, and at saturation (about 10 years) would be about 20 mr/hr at 1 meter from a 1000 w source.

The anion exchange purification step of the kilogram-scale technetium purification run was completed, and denitration and concentration of the 120 liters of product is in process.

Studies are in progress in A-Cell of 325-A to pilot with actual sludge the leaching and dissolution procedure proposed for the treatment of Purex tank sludge in the Hanford Waste Management Program. Significant observations to date are that the sludge solids remaining after the acid dissolution and caustic metathesis steps are very voluminous and that their settling rates are very poor.

Rapid-scanning polarography is being used to study the mechanism of reduction of Pu^{3+} in fused chlorides. The results indicate the existence of a divalent plutonium intermediate.

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The pot from an ORNL-type pot calcination of Purex full-level waste, run in A-Cell, has been sectioned. It was found that, contrary to earlier results with simulated feed, the solids had not melted but had pulled away from the wall and draft tube.

The Hanford Mobile Whole Body Counter was used to make uranium ore inhalation studies at two mines in Colorado.

Some modifications were made in a Los Alamos rapid procedure for plutonium bioassay. The procedure provides results in about 2 hr.

More experiments were completed which provide information on the mechanism of radiation damage to human red blood cells and the abilities of chemical additives to reduce this damage. Three protective agents tested gave protection that paralleled the protective indices determined by the dye competition method.

The computer code described last month, for calculating radioisotope generation and decay has been revised to obtain output in atoms, grams, curies, beta watts, and kilowatt-hours.

Major efforts on the development of heat source technology were devoted to the preparation of materials for trial pneumatic impaction and to impaction of two new can variations.

Essentially all (93%) of a pneumatically impacted and irradiated thorium sample was dissolved in 3 hr in a nitric acid-trace hydrofluoric acid-trace aluminum nitrate solution; the residue was readily dissolved in a small volume of fresh acid.

Thorium distribution ratios for a 1-5 dimethyloxypentane extractant were about five times higher than for hexone under similar conditions.

Solvent extraction testing of cesium recovery by the SCREX process in the presence of gamma radiation indicates a marked reduction in cesium extraction.

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A sample of Penberthy 3.8 density glass discharged after irradiation to 8×10^6 r at a rate of about 15,000 r/hr.

Samples of an irradiated, incrementally loaded UO_2 plus PuO_2 PRTR element did not completely dissolve in 2 hr in boiling, concentrated nitric acid.

The C-Cell Salt Cycle process vessel has been equipped with a graphite partial liner. A codeposition (Pu and U) experiment in this vessel was unsuccessful because of the presence of carbon particles.

A modified Astrosonics atomizing nozzle has operated satisfactorily for six runs in the cold semiworks spray calciner.

4. Biology

Reinfection of trout with columnaris continues to occur as long as contact is maintained with other infected fish. One of 10 fish under observation completed two cycles of infection within a 2 month period and at no time cleared itself completely of the organisms.

Field sampling of fish continues to give a negative response with respect to columnaris associated with the fish gills.

Following skin exposure of miniature swine with I^{131} vapor, maximum thyroid concentration occurred on the fifth day and amounted to from 0.2 to 0.5% of the initial skin deposition. This finding suggests that I^{131} movement from skin to blood is slow and very limited in amount.

Thyroid uptake of injected I^{131} in day-old lambs was reduced by 60% in those animals born to mothers that had been administered 100 mg of stable iodine 2 or 3 days before parturition.

Desferrioxamine B (DFAB), a chelating agent that has been effectively employed in combating iron disorders in humans, was found to be very effective in preventing plutonium deposition in bone of rats. Total skeletal deposition was reduced from 61% of injected plutonium in untreated controls to 1.5%

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in DFAB treated animals. This is probably the most effective plutonium removal yet obtained in prompt treatment of rats.

Further results from studies of the metabolism of lung tissue indicate that fatty acid synthesis occurs only in the mitochondria fraction.

Experiments were initiated to determine whether cigarette smoke inhaled by rats altered the rate of clearance of inhaled radioiron particles from the lungs. Preliminary results showed no gross effects following acute administration of cigarette smoke.

Tissue respirometry studies of temperature acclimation in trout indicate active use of the pentose shunt in carbohydrate metabolism at low water temperatures (8 C). Further work is continuing to determine whether the pentose shunt is maintained in this highly active state when water temperatures are increased.

Leaf/soil ratios for the uptake of W^{185} varied over 50-fold in different soils. Soil type appears to be of greater influence on the uptake of this anion than with most cations previously studied.

Body burdens of Cs^{137} in adult Eskimos from Anaktuvuk Pass increased about 40% over measurements made in mid-January. Present levels are about the same as those of last June. A maximum value of 1250 nc was measured in an adult male.

Gross autoradiograms of alpha radiation can be obtained with less than 1/500 of the exposure time normally required by NTB plates. This reduction in time is obtained by using ZnS as a scintillator and a fast film to record the scintillation.

5. Programming

Recent revisions to Hanford fuel cycle and economics codes have included:

- Updating of cross-section libraries

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- Modification of QUICK to handle multiple reactor zones with plutonium of different vintage (i.e., different isotopic composition) in each zone
- Revision of MINIMIZER to determine the minimum fuel cost point directly as a function of fuel exposure, and also to tabulate fuel cost as a function of exposure

The AEC has approved a Safeguards Analysis extending the range of fuels and lattices for light-water moderated experiments in the Plutonium Recycle Critical Facility (PRCF).

TECHNICAL AND OTHER SERVICES

There was one new plutonium case confirmed by special bioassay analysis during the month. It resulted from inhalation following a glove rupture in the Weapons Manufacturing Building (234-5) and was estimated to be 4% of the maximum permissible body burden (MPBB, plutonium with bone as reference, is 0.04 μCi). The total number of individuals who have received internal plutonium depositions at Hanford is 343, of which 247 are currently employed.

An IPD maintenance foreman received an exposure of 1.5 rems whole body penetrating radiation during April as measured by his beta gamma film badge dosimeter. This exceeds the Hanford Operational Control limit of 1 rem per badge period. The employee's accumulated exposure, including the above, does not exceed any permissible limit.

A spread of contamination occurred when four HL employees were performing an experiment with sodium iodide (I^{131}) in the Cold Semiworks Facility (321-A). Examination in the Whole Body Counter indicated depositions of I^{131} for the four ranging from 7 to 14% of the maximum permissible body burden (MPBB, I^{131} based on the thyroid = 700 μCi).

Planning is well under way on a new system to process and analyze multidimensional gamma ray spectrometer data.

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A closed-form solution was obtained for steady state heat diffusion in a long cylindrical body possessing nonuniform and nonlinear thermal sources. This problem arose in connection with the investigation of possible causes of the recent Redox fire.

A program was completed which computes the Fourier expansion of a function specified by a set of equally spaced data. The data are then transformed to yield their spectral distribution.

Using weight percentages and screen sizes calculated from theoretical particle packing studies, a satisfactory "inverted cluster" fuel element was produced by vibratory compaction on the first attempt.

The statistical analysis to investigate the effect of incubation temperature and rearing temperature on the growth of cichlids was completed.

The computer program for the quantitative analysis of metallographic data collected on the Ziess particle size analyzer has been completely debugged and is now being documented. A companion program to unfold an observed two-dimensional particle size distribution into its three-dimensional counterpart is in progress.

Studies of the dimensional behavior of production fuel elements resulting from out-of-reactor thermal effects indicated that there will be considerable difficulty in duplicating in-reactor effects by these methods.

The statistical analysis of bond integrity measurements from an integral spire core test on production fuel elements strongly indicated the superiority of a particular vibrator.

A report on a systems approach to economic-biological interactions under postnuclear attack conditions was prepared. This report formed the basis for discussions with the Technical Analysis Branch of the Division of Biology and Medicine, and an expanded version is being prepared for discussions with the RAND Corporation in June.

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SUPPORTING FUNCTIONS

PRTR output for April was a record 1851 Mwd, for an experimental time efficiency of 91% and a plant efficiency of 88%. There were five operating periods during the month, four of which were terminated manually and one extended through month-end. A new record was set for the longest sustained critical period. A summary of the fuel irradiation program as of April 30, 1964, follows:

	Al-Pu		UO ₂		PuO ₂ -UO ₂		Other		Program Totals	
	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd
In-Core	0		7	1 468.2	78	12 136.9			85	13 605.1
Maximum				294.6		288.5				
Average				209.7		155.6				
In Basin	7	569.9	25	2 720.1	33	2 426.7			65	5 716.7
Buried							1	7.3	1	7.3
Chemical Processing	68	5 465.8	35	1 965.8					103	7 431.6
Program Totals	75	6 035.7	67	6 154.1	111	14 563.6	1	7.3	254	26 760.7

Note: (Mwd/Element) x 20 = ~Mwd/ton_U for UO₂ and PuO₂-UO₂.

Heavy water and indicated helium losses for April were 945 lb and 114,411 scf, respectively.

Experiments with EBWR fuel continued in the Plutonium Recycle Critical Facility. Data were taken for rod and sheet calibrations, moderator level sensitivity, horizontal and vertical traverses, and reactor noise analysis. Void and substitution measurements are in progress at month-end.

The Fuel Element Rupture Test Facility was not operated during April. The test section had been removed from the reactor to permit a thorough inspection of the piping and required maintenance.

The chemical processing of 35 spent PRTR fuel elements of Al-Pu was completed at Redox. Plutonium remaining (0.126 kg) from the first campaign

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was added to 6.147 kg from this (third) campaign, for a total 6.273 kg. One batch of 2.263 kg had a Pu^{240} content of 22.44% and the remaining batch of 4.01 kg had a Pu^{240} content of 27.34%. A shipment of 3.7 kg was made to the 308 Building.

Technical Shops Operation total productive time for the period was 26,703 hr. Distribution of time was as follows:

	<u>Man Hours</u>	<u>% of Total</u>
N-Reactor Department	2 366	8.8
Irradiation Processing Department	4 638	17.4
Chemical Processing Department	763	2.9
Hanford Laboratories	18 936	70.9
Hanford Utilities and Purchasing Department	0	0

Laboratory Maintenance Operation total productive time was 23,000 hr of 25,200 hr potentially available. Of the total productive time, 85% was expended in support of Hanford Laboratories components, with the remaining 15% directed toward providing service for other HAPO organizations. Man-power utilization (in hour) for April was as follows:

A. Shop Work		2 500
B. Maintenance		8 700
1. Preventive Maintenance	2 600	
2. Emergency or Unscheduled Maintenance	2 000	
3. Normal Scheduled Maintenance	4 100	
C. R&D Assistance		11 800

Two additional research and development authorizations were received from off-site sources as follows:

1. Evaluation of Tungsten-Technetium Alloys for NASA - \$60,000.
2. Advance Test Reactor Heat Transfer Burnout Studies for Phillips Petroleum, contractor at IOO-AEC - \$150,000.

The heavy water inventory at the end of April 1964 showed a loss of 945 lb valued at \$13,070 for the PRTR. Heavy water scrap generated during the month amounted to 602 lb, resulting in a \$967 charge to operating cost. Heavy water accumulated at April 30, for return to SROO, amounted to 17,961 lb valued at \$224,060.

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Cumulative data of Hanford visitations:

	<u>Number of Visitors</u>	
	<u>In April</u>	<u>Since June 13, 1962</u>
Visitors Center	2 060	66 029
Plant Tours	440	---

HAPO professional recruiting activity for April:

	<u>Plant Visits</u>	<u>Offers Extended</u>	<u>Offers Accepted</u>	<u>Offers Rejected</u>	<u>Offers Open</u>
Ph. D	10	3	1	5	2
BS/MS (Direct Placement)	-	1	1	1	3
BS/MS (Program)	-	11	11	28	19

Eight Technical Graduates were placed on permanent assignment. There were no additions, leaving a total of 50 program members at month's end.

Authorized funds for eight active projects total \$7,543,500. The total estimated cost of these projects is \$10,550,000. Expenditures on them through March 31, 1964 were \$2,914,000.

Manager, Hanford Laboratories

HM Parker:JEB:dh

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REACTOR AND FUELS LABORATORY MONTHLY REPORT

APRIL 1964

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - O2 PROGRAM

1. Metallic Fuel Development

N-Reactor Fuel Evaluation. Radiometallurgical examination of an N-Reactor fuel element which was irradiated to high exposure in a KER loop is under way. The outer component attained 3070 Mwd/ton and experienced 3.64 vol% fuel swelling; the inner component attained 2440 Mwd/ton and experienced 1.17 vol% fuel swelling. Superficially, the elements appear to be in good condition with no visible indications of warp, clad bumping or rippling, crud deposits, corrosion, or inner bore distortion. The closures and heat-affected zones show no evidence of unusual performance. Although metallographic examination of the fuel in the outer component at magnifications to 1000X reveals no evidence of spherical porosity such as might be generated by gas agglomeration, the swirled or distorted structure contains small irregular shaped voids. These voids appear to be characteristic of "grain boundary tearing". Electron microscopy will be used to further study the microstructure and its relation to the measured volume increase.

Outer and inner N-fuel components that have been irradiated to about 2000 Mwd/ton in the KER loops have been induction heated over their full 24-inch length to 980 C and water-quenched. This work was done in the Radiometallurgy facility by Particulate Gaseous Waste Research (HL) in support of N-Reactor hazards studies. After quenching, the outer clad surface of the components appear bumpy but showed no evidence of failure. Bulk densities of the components revealed an average increase in volume for two outers of 7.1%, and for two inners of 4.8%. Examination of cross-sections of the tested components show that the outer claddings had separated from the fuel by failure of the bond and that the fuel had cracked severely.

Dimensional Changes in Irradiated N-Fuel. The measurements of prototype N-outer fuel elements from two KER loop tests (K-3-18 and K-4-15) have been completed. The swelling of the elements as calculated from these data are in good agreement with the swelling data obtained by density measurements. In addition, swelling gradients can be observed

in the fuel elements which are in agreement with the swelling versus reactor position curves generated from the density data.

No severe or unexpected dimensional changes have been observed in the fuel elements. An analysis of the data, though incomplete, indicates that the ID of the fuel is decreasing, and the OD is increasing as a result of the swelling. In most cases, the ID decrease is approximately one-half of the magnitude of the OD increase. A maximum of 0.016 inch (single-throw) warp over 18.3 inches of fuel was observed in the irradiated fuel. The maximum increase in the average fuel thickness was measured as 0.009 inch which corresponds to 3.6% swelling.

Fluted Fuel Element Irradiation. The two N-single tube fluted fuel elements undergoing irradiation testing in the ETR under prototypic N-Reactor operating conditions have completed their tenth cycle of irradiation. The maximum accumulated exposure is 2000 Mwd/ton and the volume increase incurred by the fuel is 1.6%.

KER Irradiation of Alloyed Uranium Fuel (British composition). Two charges of KSE-5 single tube fuel elements are being irradiated in KER Loops 1 and 2 to provide a comparison of swelling behavior of alloyed and unalloyed fuel over a range of exposures and fuel operating temperatures. The charges consist of alternately loaded elements containing U with Fe and Al additions (simulated British fuel). One of the charges is being monitored with thermocouples located to detect buckling or collapse of the inner bore (as was observed with KSE-3 elements at 3600 Mwd/ton) by monitoring changes in coolant temperatures.

Target Element Development. Post-irradiation examination of the target elements which were irradiated in KER Loop 2 has continued. Vacuum fusion analysis of the target core of element No. 25 yielded a gas content of 26.5 cm³ of gas (He + T₂) per cm³ of target core.

Gas samples from the annulus between the Al can and the Zircaloy-4 can of elements Nos. 25 and 28 have also been analyzed. No tritium was detectable within the resolution of the mass spectrometer. Samples of the Zircaloy-4 can wall and the end caps from the same elements were vacuum outgassed. Again, no tritium was detectable in the resulting gas samples. However, molecular species which could be interpreted as HD or He³, and D₂ or HT were detected in the gas. There is reason to believe that these molecular species are HD and D₂. This will be verified by beta-counting the gas.

Four additional target elements designed to evaluate the performance of two-phase lithium-aluminum alloys have been discharged from KER

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Loop 2. No evaluation of these elements is planned during the next two months

Extrusion of Mg-14 Li-2 Al Alloy. Two billets were extruded to evaluate the feasibility of using the Mg-14Li-2Al alloy for target material. The billets were both preheated to 250 C and one billet extruded into 0.692 inch rod at a reduction of 34/1. The other billet was extruded at 28/1 into a 0.692 OD x 0.230 ID tube.

The same lubrication and extrusion technique was used as was developed for the aluminum clad Li-Al alloys and appears satisfactory. The extrusion constants were approximately 6 TSI which compares closely with 1100 grade aluminum extruded at 400 to 450 C.

Long life fuel fouling detectors for use in N-Reactor are being fabricated using a Th - 1 1/2 wt% (fully enriched) U - 1 wt% Zr alloy fuel. Two 65-pound ingots have been double vacuum arc melted and are currently being prepared for coextrusion. Final melting of the two ingots was performed using 4000 amps and 25-26 volts. Excellent sidewall quality was noted and final billet dimensions were achieved at 100% cleanup.

Alternate Uranium Composition. Studies are in progress to determine the effects of altered fuel compositions upon fuel element fabrication, corrosion behavior and irradiation swelling resistance.

Samples of a number of 1.47% enriched uranium base fuel alloys that had been fabricated to 0.600-inch diameter Zircaloy-2 clad rods were heat treated and are currently being examined for structure evaluation.

Uranium Alloy Development. A series of 14 uranium alloys containing niobium and zirconium additions were investigated with respect to corrosion properties. On the basis of free energy measurements on the corrosion cell, several alloys appear very promising. A gamma heat treated ternary alloy of 2% Nb + 2% Zr had a corrosion cell free energy of -10 K cal as compared to a delta F of -142 K cal for alpha annealed ingot uranium. A 4% Nb alpha heat treated alloy and a 2% Nb beta heat treated alloy each had a free energy of -20 K cal. These low free energy values indicate a low corrosion potential and provide a reasonable screening test of alloys for further investigation.

Hot Headed Closure Studies. Work on a nondestructive testing evaluation study has continued. Twenty-six welded elements have been ultrasonically tested. One element contained known defects and several

other elements were welded with variations in post-heat and pressure. The test results for each element were recorded so that the elements can be "mapped" and cross-sectioned for examination.

A tensile testing fixture was designed and constructed. This fixture will be used to study the effect of process variation on joint strength.

Welding. Eight of the fifteen elements welded by resistance projection welding have been autoclaved. The last three have been given a 400 C, 1500 psi steam autoclave treatment for 72 hours. Two of the autoclaved elements have been given a thermal shock test. These were quenched from 600 C into water ten times. Metallographic examination of these pieces does not show any evidence of damage. Further welding and testing will be carried on as scheduling permits.

Braze Alloy Preforms for N-Reactor. Work is in progress to produce Be-Zr preform wire. A small solid cylinder of this alloy has been encapsulated, under vacuum, in 304 stainless steel. This capsule will be swaged at 800 C to 1/8" diameter rod and then wire drawn at this temperature to 25 mils in diameter.

Attempts will be made to produce another preform by sintering powdered Be-Zr alloy into rings. A zirconia mold will be used to contain the powdered metal compact.

Charging Force Studies. Approximately 100 tests have been performed to determine friction factors for various fuel element-process tube combinations. The major inducement for determining this information are new support concepts (split weld tee section, and split weld arch section) for the K5 fuel element. Forty-eight of the tests were comparisons between existing K5 fuel supports, and the two new support designs. Twenty of the tests were directed toward settling the question about the effect of charging lubricant in N-Reactor charge-discharge operations. The balance of the tests were supplemental to work done earlier in this area. The data taken have not yet been interpreted.

2. Corrosion and Water Quality Studies

Corrosion of Zirconium in Heated Crevices. A corrosion test of a pair of electrically-heated specimens clad in Zircaloy-2 to which N-suitcase-handle fuel element supports had been welded was completed. The specimens were exposed for 56 and 120 days, respectively, in TF-3

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at a heat flux of about 200,000 Btu/hr-ft², 314 C bulk water temperature, 2000 psia pressure, and pH 10 with NH₄OH. Metallographic examination of 30 heated crevices under the supports revealed only slight corrosion (0 to 0.2 mil penetration). The temperature and exposure in this test duplicated that in previous tests with LiOH where severe corrosion (up to 7 mils) was encountered. This test confirms the conclusion from previous tests with NH₄OH at shorter exposure and lower temperature that this type of corrosion under N-fuel supports can be avoided by using NH₄OH rather than LiOH. Two additional samples exposed in NH₄OH were discharged from TF-2 but have not yet been examined.

Testing of NH₄OH in KER Loops. The thermocouple elements were discharged from KER-1 after 43 days of operation at pH 10 with NH₄OH following a decontamination of this carbon steel loop. Throughout the test the thermocouple readings indicated no rise in cladding temperature due to crud deposition. Crud concentration in the coolant was low, averaging 25 ppb. The discharged elements were visually examined in the KE viewing basin. The four NIE-1 heater elements, and also the Zr-2 sleeves surrounding them, were covered with a uniform, adherent red brown crud deposit, presumably ferric oxide. The film was thin enough on two of the elements that dark ZrO₂ could be seen underneath; the film completely covered the surface of the other two. A similar red brown film was present on the two Zr-clad thermocouple elements and on two similar elements (without thermocouples) clad in stainless steel. The film appeared thinner on the downstream thermocouple element than on the one upstream. This test indicates that this degree of crud deposition, which seems typical of KER operation with NH₄OH, does not produce significant increases in cladding temperature.

Corrosion of Carbon Steel in Ammoniated Coolant. The corrosion rate of carbon steel in ammoniated water is a factor of about two higher than the long-term rate normally encountered in lithiated coolants at pH 10 and 300 C. The rates in the present tests may decrease further as additional exposure is accumulated. Only about half as much crud was released in the TF-1 test as is normally encountered in pH 10 lithiated coolants. The negative crud release rate in TF-4 indicates that deposition of corrosion products released by the rapidly corroding defilmed loop piping surfaces is occurring more rapidly than metal is being removed from the filmed test specimens. The samples in these tests were exposed during only the periods of normal high temperature operation in cyclic decontamination tests. The loop piping, but not the test coupons, was chemically cleaned weekly. This resulted in release of corrosion product from the piping and deposition on the coupons.

N Steam Generator Tubing. A sample of N-Reactor steam generator tubing which had severe pre-exposure intergranular attack was examined by eddy current techniques after a three-month exposure in 530-580 F, pH 10 (LiOH) water. There was no change in the amount of corrosion during the exposure period.

A sample of N-Reactor steam generator tubing which was ground on the OD until an area of intergranular corrosion was intersected, was removed from TF-7 after 2½ months of exposure to pH 10 (LiOH), 530-580 F water. The lithiated water leaking through the intergranularly attacked area to the atmosphere had caused a large amount of white precipitate (LiOH or Li_2CO_3) to form over the leak area. Metallographic examination did not reveal any wire drawing due to leakage or caustic cracking due to the deposit formation.

Evaluation of Cleaning Solution for N Reactor. The corrosion data from the Alk-14 test (as reported last month) has been evaluated. (Alk-14 is a proprietary cleaning solution containing ammonia, EDTA, and inhibitors.) Penetrations were nil on AISI 304 stainless steel, Inconel-600, and Zircaloy-2 after a 17-hour exposure to 2.4% Alk-14 at 121 C. Penetrations on A-245 carbon steel were 0.1 mil during the same exposure. Corrosion rates were normal during a one-week operating period following the cleaning. There was no stress-corrosion cracking on 304 stainless steel or A-212 carbon steel.

Chemonuclear Studies. The in-reactor test section of the single-pass chemonuclear facility was installed in KW Reactor. Assembly of the out-of-reactor package is complete except for the relief valve. The development test for the ammonia-hydrazine tests was approved.

Operation of Dual Purpose In-Reactor Capsule. The dual purpose tritium evolution and zirconium-gas atmosphere capsule was charged at the last outage at KE Reactor and has been in operation since April 5, 1964.

The aluminum-lithium target capsule has operated at 360 C (680 F) since April 10, 1964. Between zero and four GVR (Gas Volume Ratio, i.e., cc T_2 + He per cc of metal), the average tritium evolution rate was 0.14% per day at 360 C. The capsule temperature is higher than anticipated, but the present outgassing rate indicates that tritium retention will be sufficiently high to permit a study of the temperature coefficient for tritium evolution near the end of the planned 40-day exposure. During the exposure tritium equilibrium pressures and tritium evolution rates will be measured.

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Small samples ($\sim \frac{1}{2}$ inch diameter by $\frac{1}{2}$ inch long) of lithium aluminate and an aluminum-lithium aluminate cermet, encapsulated in aluminum, are also undergoing irradiation in the capsule envelope to provide information on the behavior of these materials under irradiation. A thermocouple on the lithium aluminate cladding indicates a temperature of 87-91 C (125-132 F); the temperature of the cermet capsule is not monitored; it is undoubtedly lower than the temperature of the aluminate capsule, since the rate of heat generation is lower.

A zirconium-gas atmosphere capsule is operating satisfactorily at 400 C (752 F) in a simulated reactor atmosphere.

Tritium Escape from Oxidized Aluminum-Lithium Alloy. The degree of tritium retention in catastrophically oxidized aluminum-lithium alloy is being studied. A 0.12 gram sample of irradiated aluminum - 0.84 wt% lithium alloy was induction heated in deionized water to 300 C (572 F) in an autoclave. The GVR of the alloy was approximately

$$14 \frac{\text{cc}(T_2 + \text{He})}{\text{cc alloy}}$$

The course of the catastrophic failure was followed by observing pressure rise in the autoclave from hydrogen generated by the oxidation reaction. The first indication of catastrophic failure occurred after the sample had been at temperature for approximately 80 minutes; the oxidation was complete after 300 minutes at 300 C.

The autoclave water was found to contain 1.35 millicurie of tritium per ml, accounting for about one-third of the tritium estimated to be in the sample. A sample of the gas above the water phase was taken, but the analysis is not yet available. If this analysis indicates that significant tritium was retained in the oxide, the oxide will be outgassed at temperatures to 1400 C (2552 F).

3. Gas Atmosphere Studies

Zirconium-Graphite Compatibility Loop. The loop completed one month of satisfactory operation. Graphite oxidation data were obtained by GRAD at 1000 ppm water in helium. Zircaloy-2 samples exposed simultaneously oxidized at the rate expected and absorbed only hydrogen from the water corrosion reaction. Inhibition of gas phase hydriding was confirmed.

Oxidation of Zircaloy-2 in CO₂. Oxidation rates of Zircaloy-2 and crystal bar zirconium in CO₂ at 350 and 450 C (662 and 842 F) have been measured for 143 days. These data show the oxidation rate in

CO₂ to be considerably less than in dilute water vapor. Hydrogen pickup data for zirconium indicate that a major fraction of the weight gain at 350 C may be due to traces of H₂O still remaining in the gas. The CO₂ contained about 50 ppm H₂O. A new experiment is planned with drier gas.

Thick Film Effects on Hydriding of Zircaloy-2. Zircaloy-2 coupons with 0, 50, 100 and 500 mg/dm² autoclave films were exposed for 14 days to 1000 psi H₂ - 15 psi H₂O at 400 C (752 F). All of the pre-filmed samples showed evidence of gas phase hydriding. The samples without a prefilm did not. The data confirm the observation reported previously that inhibition of gas phase hydriding breaks down after thick films develop.

Graphite-Burnout Monitoring. During the summer of 1963, well characterized TSGBF graphite was established as the standard material for burnout monitors; prior to this time monitor graphites were those used in construction of the respective reactors. The oxidative characteristics of all monitor materials were experimentally determined in the laboratory and used to correlate the newly acquired data with earlier results. A recent test in which alternately positioned TSGBF and KS monitors were exposed in channel 1475 at F-Reactor disclosed a seven-fold higher weight loss for the latter material as compared to a two- to three-fold higher rate determined from the laboratory experiments. Weight increases of some of the down-stream monitors and rusty spots were also observed, the latter possibly due to water entering the channel during sample removal.

Additional monitors exposed from November 20, 1963 to April 1, 1964, in channels 3580 and 3890 disclosed burnout rates (adjusted by a factor of seven) which did not exceed two percent per thousand operating days.

Graphite-Zirconium Compatibility Loop Experiment. A recirculating gas loop has been constructed by Chemical Metallurgy Operation which simulates a two-foot section of graphite in the N-Reactor. The loop consists of three main furnaces, an oxidation furnace, a diffusion-thimble furnace and a zirconium-sample furnace. All gas reaching the zirconium samples must diffuse across a graphite block of similar dimension to a tube block of the reactor.

The water-vapor content of the gas in the loop is controlled by an electrolytic hygrometer, and a gas chromatograph is available for analysis of H₂, CO, CO₂, CH₄, O₂, A, and N₂ in the gas stream.

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The oxidation furnace was charged with a bar of graphite 24 x 4 x 5.8 inches. A number of sample holes were placed in the bar to permit measurement of the depth of oxidation into the bar as well as along the length of the bar.

The first oxidation results have been obtained. The temperature of the bar was: upstream end, 785 C; center, 820 C; downstream end, 810 C. The water content was nominally 1000 ppm. All samples greater than 3/8" in from the surface showed very low oxidation rates, $\sim 3 \times 10^{-7}$ g/g/hr. The rates of the surface samples were a factor of 10 higher than the interior samples. The rates at the upstream end of the bar were also higher than downstream even though the temperature was lower. The lower rates along the length of the bar were possibly due to depletion of oxidant, build up of inhibitors or a combination of these causes. The oxidation rates of the surface samples were lower than those for laboratory samples, which have been pre-oxidized, and exposed to 1000 ppm of water vapor at comparable temperatures. No pretreatment was given the loop samples.

4. Process Tube Development

Fracture Studies. A burst test was made on a hydrided (275 ppm) N-Reactor pressure tube defected with a $2\frac{1}{4}$ inch milled slot. Strain gages in the hoop direction showed a stress intensification factor of 3.5 at the slot tip and 1.5 at a distance of $\frac{1}{2}$ inch from the slot tip. A hoop strain gage adjacent to the midpoint of the slot length showed an unexplained compressive stress equivalent to 1.5 times the expected tensile stress for an undefected tube.

To confirm the compressive stress noted above, a specimen of KER tubing defected with a two-inch slot was tested with hoop strain gages at 0, 45, 90, and 180° from the midpoint of the slot. At the failure stress of 33,000 psi the gages recorded the following strains: 0°, 1545 micro-in/in negative; 45°, 270 negative; 90°, 808 positive; 180°, 1560 positive. The expected strain in an undefected tube at this stress level would be 1200-1500 micro-in/in.

Electronic instrumentation on the hydrided N-Reactor tube indicated a crack speed of 4580 ft/sec. This compares with a speed of 2750 ft/sec for a similar unhydrided specimen.

Project CAH-922 Irradiated Burst Test Facility. The first two sections of irradiated material were burst tested in this new facility. All equipment has performed very well.

5. Thermal Hydraulic Studies

Transient Heat Transfer Experiments for N-Reactor. Transient experiments investigating loss-of-coolant accidents for N-Reactor fuel tubes were continued. In recent tests the flow, pressure, and temperature transients associated with the occurrence of an inlet connector rupture during heatup of the primary system were investigated. The experiments were performed on an electrically heated, full-scale model of the downstream half of an N-Reactor fuel column with a prototypic N-Reactor pressure tube and connector fittings. The following initial steady-flow conditions were established in the experimental mock-up prior to the transient:

Flow	41,000 lb/hr
Outlet Riser Pressure	350 psig
Outlet Water Temperature	400 F
Inlet Water Temperature	337 F
Total Fuel Column Power	2500 kw.

The transient was initiated by actuating quick-acting valves which simulated the rupture by opening the connector to atmosphere. Heated and pressurized water was supplied to the test section from a 1000-gallon vessel during the transient. Simultaneous with the valve actuation, the electrical energy supplied to the fuel model was reduced in such a manner to simulate a 1500 in-hour reactor scram with a two-second delay.

Preliminary examination of the oscillographic recordings of the test data indicated that no significant temperature increases occurred in the simulated fuel column during the transient. Peak rear-to-front flow rates as high as 300 gpm were noted.

N-Reactor Boilout Tests. One of the important hazards problems associated with N-Reactor is the case where coolant flow is lost due to a ruptured header and the reactor boils dry. The fuel would rise to very high temperatures, possibly in excess of 1500 F. Upon re-establishment of the coolant flow, large quantities of steam would be released. This steam must be contained in the reactor building since it may contain fission products from fuel elements failed by the high temperatures. The possibility exists that the quantity of steam produced in such an incident would be in excess of the containment capabilities of the reactor building.

To determine the quantities of steam which could be produced, an experiment was performed in which an electrically heated fuel element model was heated to 1500 F without coolant. When this temperature

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was reached, the coolant flow was re-established and inlet pressure was controlled at 300 psi. Inlet flow, inlet and outlet pressures, heat transfer surface temperatures, and outlet steam or steam-water flow were continuously recorded during the experiment. It was found that an unstable situation occurred following flow re-establishment with large oscillations in both flow rate and surface temperatures taking place. Inlet pressure oscillations with a peak-to-peak amplitude of 200 psi and a period of about two seconds were observed. Inlet coolant flow oscillated from 0 to 30 gpm. Outlet pressure remained at a relatively constant value.

This instability existed until surface temperatures were reduced sufficiently to permit a change from film to nucleate boiling. It was concluded that similar oscillations could be expected in the N-Reactor following such an incident.

Although the apparatus used in this experiment was not an exact replica of an N-Reactor fuel channel, the results will allow a valid check on the analytical methods being used to investigate the event for the reactor.

Hydraulic Tests for Present Production Reactors. Tests were performed to determine whether pressure surges resulting from hydraulic seating of a fuel charge can cause a Panellit pressure trip and shut down the reactor. The results showed that a trip can occur, generally, when the fuel column is separated from the rear nozzle cap by a distance of two inches or more. In these experiments 0 II N-fuel elements in a straight C-Reactor process tube were considered. Vertical and horizontal bends in the process tube as well as tube-to-fuel contact conditions can be expected to change the coolant flow required to seat the fuel charge, and, therefore, will affect this criterion somewhat.

6. Shielding Studies

N-Reactor Shield Evaluation. The "zero power" N-1 hot shield plug irradiation was conducted on April 2 for a period of eight hours. The reactor power level was maintained during this period at an almost constant level. Ion chamber readings, a neutron spectrometer count and thermocouple readings were obtained during this period. The Li^6 neutron detector and an ion chamber were positioned in the bottom of the shield well. This test included irradiation of 36 gold foils, and four each nickel, iron, and titanium wires in the reflector thimble. All foils installed within the shield plug and shield flux openings received optimum activation and the data are currently being

analyzed. Additional data taken included reactor coolant temperatures, complete reactor shield temperature readings and shield coolant flow rates. Absolute gamma dose and absolute neutron flux normalized to an arbitrarily selected value of one megawatt reactor power are being calculated.

7. Graphite Studies

N-Reactor Irradiations. The long-term irradiation of N-Reactor graphite continues to progress satisfactorily. The last of the third-generation capsules, H-6-3, was removed from the GETR after very successfully completing five cycles (115.8 effective days) of reactor operation. Preliminary length-change data confirm data from previous capsules.

The second of the fourth-generation capsules, H-5-4, was installed in the GETR and has operated satisfactorily during the past month. The H-4-4 capsule also continues to operate satisfactorily.

Neutron Flux Monitoring. The "Foil" program, a FORTRAN Program for the IBM-7090, is used to compute neutron exposures and fluxes from foil activation data. The third revision of this program has been prepared; this revision is to provide for cases where the irradiation time is less than 15 days. A report HW-81367, on the program and its use, has been prepared and will be issued within the next few weeks.

Oxygen Bomb Calorimeter. A copy of the "CHONS" data reduction program for the IBM-7090, which computes the energy of formation from "raw" heat-of-combustion data, has been received from the Naval Ordnance Test Station. Two subroutines have been rewritten to make them compatible with the HAPO FORTRAN II monitor, and test programs have been run successfully.

Several minor problems have prolonged the equipment calibration procedure, but the actual standardization runs should commence within the next month.

B. WEAPONS - O3 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.

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C. REACTOR DEVELOPMENT - 04 PROGRAM1. Plutonium Recycle ProgramFuels Development

PRTR Fuel Exposures. The present PRTR loading comprises seven natural UO₂ elements, 59 UO₂-0.48 wt% PuO₂ elements, and 19 UO₂-1.0 wt% PuO₂ elements. PRTR fuel element exposures on April 23, 1964, were:

Maximum UO ₂ Exposure/Element (Swaged)	5840 Mwd/ton
Average UO ₂ Exposure/Element	4020 "
Maximum UO ₂ -0.48 wt% PuO ₂ Exposure/Element (Incrementally loaded, Vipac FE-5118)	5680 "
Maximum UO ₂ -0.48 wt% PuO ₂ Exposure/Element (Incrementally loaded, Swaged FE-5142)	5540 "
Average UO ₂ -0.48 wt% PuO ₂ Exposure/Element	3620 "
Maximum UO ₂ -1.0 wt% PuO ₂ Exposure/Element	2280 "
Average UO ₂ -1.0 wt% PuO ₂ Exposure/Element	1300 "

PRTR Fuel Fabrication. One vibrationally compacted and two swage compacted fuel elements were delivered to the PRTR during April. Three fuel elements were fabricated for zero power physics tests. They contain UO₂-0.5% PuO₂ with various Pu-240 concentrations.

Reliability of operation of the Dynapak machine improved. Only two "off-cycles" occurred; they were caused by tooling alignment problems. A total of 652 kilograms of mixed oxide was densified by April 24 without a misfire or a broken can. The can breaking problem seems to have been solved by changing lubrication techniques.

PRCF Fuel Fabrication. Three hundred kilograms of UO₂-2 wt% PuO₂ mixed oxide (16.5% Pu-240) were impacted for 250 three-foot long rods for the PRCF. Ninety percent of the material has been decanned and is ready for crushing and screening. A total of 1000 tubes have been machined and four hundred have had the first closure completed.

PCTR Fuel Fabrication. Fabrication of 234 elements for the PCTR is complete. Impacted UO_2 -0.9% PuO_2 was vibrationally compacted into 164 rods, and sintered pellets were used for the other 70 rods. The plutonium contained 8% Pu-240. Rods are 36 inches long and about $\frac{1}{2}$ inch in diameter.

Water-tube (Inverted Cluster) Fuel Elements. Two water-tube fuel elements, each with seven coolant tubes, were designed for irradiation testing in PRTR, one with replaceable spike-enrichment fuel rods located in the channels. This offers possibilities for more efficient plutonium recycle in thermal reactors by permitting easy separation of "new" and "old" plutonium and, perhaps, extended reactivity advantages through an anticipated Phoenix action in the replaceable spike enrichment fuel rods.

Initial irradiation of these elements will be conducted in the FERTF. The calculated power generation and maximum fuel temperature for the Mark IX-A element operating in the FERTF position with no fuel in the replaceable spike enrichment rods are 180 Kw and 1350 to 1510 C, respectively. Calculated power generation and maximum fuel temperature for the Mark IX-B element are 680 Kw and 1800 to 2075 C, respectively.

A Mark IX-B element was completed.

Ultrasonic testing techniques were used to inspect Zr-2 cladding for the water tube (inverted cluster) PRTR fuel elements, Mark IX-A and Mark IX-B. Special techniques were developed by PTO to inspect the 3.070" OD tubing having eight equally spaced ribs electron-beam welded to the outer surface. Several areas of incomplete welds were removed to prevent LiOH concentration and subsequent corrosion.

Hot Wire Enrichment. Fabrication of a PRTR rod containing an axial Pu-15 wt% zirconium enrichment wire was started. This method of enrichment offers potential advantages in simplified fabrication of plutonium-containing fuel elements.

Irradiation Testing of EBWR Prototypic Fuel Rods. Thirty-five capsules containing short sections of vibrationally compacted EBWR fuel rods are being irradiated. All contain impacted U^{235}O_2 - PuO_2 fuel. Seven have prototypic fission gas plenums. Four 2.5 wt% PuO_2 specimens and four of the 1.5 wt% PuO_2 specimens (two with plenums) were discharged at exposures to 1.6×10^{20} fissions/cm³ (5600 Mwd/ton UO_2). One of the latter specimens that presumably generated 1080 w/cm (33 kw/ft) and a heat flux of 315 w/cm² (10^5 But/hr-ft²) developed columnar grain

growth and solid solution formation over two-thirds of the transverse cross sectional area. No central void was observed in the examined section. Exposure is estimated to be 0.6×10^{20} fissions/cm³ (2000 Mwd/ton UO₂).

Corrosion and Water Quality Studies

Irradiation Effects on the Corrosion of PRTR Zircaloy-2 Process Tubes. Metallographic determination of corrosion of two sections of a discharged PRTR Zircaloy-2 process tube is in progress. The process tube was in service from startup of PRTR until April 4, 1963. During this period of time, the tube was operating at a temperature of 400 F (204 C) or more for approximately 200 days. The two sections being examined have seen different flux densities, approximately 3.86×10^{12} nv (> 1 Mev) and 2.02×10^{13} nv (> 1 Mev).

Preliminary measurements of the oxide thickness on metallographic cross sections are equivalent to weight gains of 41 mg/dm² for both samples.

Assuming an original autoclave film on the process tube equivalent to a weight gain of 16 mg/dm², the in-reactor weight gain would be approximately 25 mg/dm². This weight gain is approximately a factor of six higher than would normally be expected for Zircaloy-2 exposed ex-reactor to the same temperature and water conditions. The data from these two samples are not sufficient to establish that the accelerated corrosion is due exclusively to irradiation. Other corrosion accelerating factors may have contributed to the corrosion of this process tube. Two possibilities are: the fluoride contamination incident and the plutonium fuel element rupture with subsequent decontamination. Before definite conclusions can be reached, additional samples must be examined.

An estimate of the amount of corrosion product hydrogen absorbed by the process tube during reactor exposure has been calculated. If the process tube corroded the same amount on both the ID and OD, and if 30% of the corrosion product hydrogen was absorbed, then the hydrogen content would have increased by 7 ppm.

Use of Chemical Shim in PRTR Moderator Coolant. Problems associated with the use of ion exchange equipment to remove boric acid poison from the PRTR moderator coolant were reviewed, and initial design recommendations were prepared and submitted. Additional anion exchange units will be required for boron control during the Phase I (low boron concentration) tests. The present moderator cleanup ion

exchanger can probably be used for both boron control and coolant purification during the Phase II (high boron concentration) tests after the anion portion has been converted to the borate form.

The use of a continuous wet chemical analyzer is feasible for monitoring the boron concentration during Phase I tests. A colorimetric procedure employing carminic acid as the indicator is satisfactory. This procedure is recommended in place of manual analyses to determine changes in boron concentration, particularly during the Phase I tests.

Plutonium Oxide Dissolution. Laboratory tests of two solutions were conducted to determine their effectiveness in dissolving powdered PuO_2 . Both solutions were tested for one hour without agitation, using 50 ml of solution and 5 mg PuO_2 . The solutions were:

- (1) 0.25 M EDTA, 0.75 M ammonia, 0.5 M hydrazine at 95-100 C.
- (2) Oxalic acid (2.3 g/l), H_2O_2 (45 ml/l), sodium oxalate (32 g/l), and peracetic acid (11 ml/l), at 85-90 C.

Cursory visual examination indicated that both solutions successfully dissolved the PuO_2 sample.

Decontamination with Dilute Solutions. A dilute APAC decontamination process was tested in the IRP Loop to clean piping surfaces contaminated with rupture debris from an irradiated metallic uranium fuel element. This process, which was recently used to decontaminate Shippingport, would permit removal of wastes by ion exchange rather than flushing. The first step was 10 g/l NaOH and 10 g/l KMnO_4 recirculated for 24 hours at 150 C. There was no change in activity levels after this step. The second step was 10 g/l dibasic ammonium citrate, recirculated 24 hours at 150 C. The filter activity was reduced from 430 to 300 mr/hr, but the test section activity increased from 38 to 45 mr/hr. A dark red-brown deposit was found on the stainless steel coupons; none of the coupons appeared to have been cleaned.

Reactor Components Development

Second Generation Mechanical Shim Rod for PRTR. Assembly of the new driving head, utilizing both heat sinks, was completed. This unit was installed with the zirconium lead screw lower assembly in the environmental test facility. The combined assembly has operated

continuously during the month under the following conditions:

- (1) Top face temperature: 450 F to 550 F.
- (2) Top shield temperature: 165 F to 180 F.
- (3) Moderator water temperature: 160 F.
- (4) Position readout equipment and limit switches energized continuously.
- (5) Operation of both A and B rods through at least two cycles each working day.

No faults have been revealed during the hot testing program with the exception of one thermocouple which is reading erroneously.

Fretting Corrosion Investigation. Tube inlet jumper vibration measuring systems were installed on PRTR channels 1354 and 1449 to augment the previous system on channel 1653. Measurements obtained during this report period show the following:

- (1) The peak-to-peak vibration amplitude at about 30 cps frequency are 3 to 3.5 mils for channel 1354 and 4 to 4.5 mils for channel 1653. These values have remained essentially constant throughout the current operating period. (Tube 1449 reading has been about 1.7 mils. However, the recorder system behavior has been erratic suggesting faulty signal transmission.) The 4 to 4.5 mil vibration on tube 1653 compares favorably with the level recorded in the prior operating period.
- (2) Prior to reactor startup on April 7, vibration measurements with pump 2 only, pump 3 only, and pumps 2 and 3 combined were obtained with primary coolant system at 1070 psig and 120 F. Recordings show only 0.1 to 0.2 mil differences from the base jumper vibration of 2 mils, although flow rates varied from 7400 gal/min to 10,300 gal/min with single pump and double pump operation, respectively.
- (3) During reactor startup on April 8, the inlet jumper vibration level changes resulting from increases in moderator level, primary coolant temperature, and reactor power were observed. Vibration amplitudes of

both inlet jumpers increase slightly (0.5 to 0.7 mil) when the moderator level was raised from 24 to 92 in. During this change the primary coolant temperature remained nearly constant at about 122 F and the reactor was subcritical. With reactor heatup from 122 F to 440 F, the vibration amplitude of tube 1354 increased about 2 mils (from 1.6 to 3.7 mils) while tube 1653 vibration increased only about 0.8 mil (from 2.0 to 2.8). However, a peak increase of 1.7 mils was recorded at a temperature of about 345 F. Reactor power increase from 7 to 70 Mw caused only 0.2 to 0.3 mil vibration increase on tube 1354 but 1.8 mils rise on tube 1653. One tube appears predominantly power sensitive and the other coolant temperature sensitive for reasons not determined as yet. It is of interest that the combined effects total 2.3 mils for tube 1354 and 2.6 mils for tube 1653, so that the over-all vibration level effects from cold to full power operating reactor appear comparable.

Another possibly significant variation has been noted in the preliminary observation of PRTR monitor records. A valve in the secondary coolant system stuck closed and caused a rise in HX-1 pressure. There was a 20 psi increase in HX-1 before the valve responded. The vibration monitor located on the inlet jumper of tube 1653 indicated an increase in amplitude of approximately 0.5 mil and was the only channel of those monitored which responded. During a subsequent routine test, wherein cold water is sprayed on the tube bundle, a similar response was obtained. Consideration of previously acquired data does not, however, support the possibility of a dominant secondary system temperature effect. It has also been observed that a reactor scram is followed by a ten-fold reduction in amplitude of vibration of a given inlet jumper.

The instrumented fuel element is still undergoing out-of-loop tests. Preliminary tests have established the characteristics of the eddy-current coils and do not indicate an inherent drift problem. The lead wires have shown erratic results this far. Tests are continuing on other types of lead wire.

A vibration transducer of the type currently used in the fretting corrosion study has been tested in a neutron

and gamma field. The purpose of this test was to evaluate the possible transient radiation effects on the transducers as caused by fast neutrons and gamma photons. The test was conducted utilizing the positive ion accelerator. The complete test consisted of six individual runs. Three of the runs utilized a tritium target and the remaining three runs were conducted with a straight beryllium target.

Gasket Test Station. The gasket test station, connected to the EDEL-I piping so as to be prototypical of the PRTR pressure-tube nozzle installation, has undergone an initial period of operation. A problem was encountered in that the nozzles and connecting lines were not insulated which required a longer (about double) heatup time and full heater power. The gasket test station is being insulated at the present time to eliminate this problem and to eliminate the hazard of hot piping.

A hollow, stainless steel "O" ring without a backing ring was tested in each of the three nozzle cap locations. Set-screw torques were established at 60, 80, and 100 lb-ft, but in no case did the gaskets hold. A special silicone rubber and metal combination gasket for the nozzle-to-tube connection has held after a one-week operating run.

Fuel Re-use. Developmental plans and drawings of the liquid sodium facility for use in making engineering evaluations of the fuel re-use concept are approximately 70% complete. Details for the sodium loop proper are essentially complete. This work includes electrical and instrumentation drawings and a redesign of the piping layout to provide acceptable piping thermal stresses to 1500 F should the need arise to upgrade the maximum loop operating temperature from 1200 F. Efforts are now concentrated on completing the details of the loop enclosure and the fuel removal and cleanup station.

Construction of the liquid sodium facility is scheduled to start in July and to be completed in September. Component procurement corresponds to this schedule.

As a result of a recent case interpretation, under the ASME Boiler and Pressure Vessel Code, it now appears possible to fully code the liquid sodium loop. This was not previously possible and necessary changes are in progress to assure code compliance under the latest interpretations.

Pressure Tube Monitoring. The pressure tubes in nine process channels were examined this month. Three of these process channels (1653, 1449, and 1354) are now examined during each scheduled outage in an attempt to correlate fretting behavior with measured vibrations. These three channels have in the past shown consistently poorer fretting behavior than the bulk of the process channels in the PRTR.

The inspections disclosed no new severe fretting since the last inspection. This inspection disclosed what appears to be a small blister in process channel 1651.

Design Analysis

Nuclear Safety of Mixed PuO₂-UO₂ Fuel in EBWR. Final results of the study carried out to evaluate the effect of heterogeneities on the Doppler effect in mixed PuO₂-UO₂ fuels were presented to the General Electric Technological Hazards Council. It was concluded that even with present uncertainties in internal transient heat conduction in EBWR fuel, the Doppler effect should be delayed a negligible amount. Significant interest was expressed by the Council in extending the analysis to include fast reactor fuels and excursions. A general treatment of this problem is being carried out for publication and will include both thermal and fast reactors.

In addition, a brief analysis of the effect of a change in PuO₂ particle self-shielding during melting was completed. It was concluded that for the particle sizes employed in EBWR fuel, the total effect is of the order of 1 or 2 milli-k and thus is nearly negligible. GETHC has approved the use of this fuel in EBWR.

Thermal Hydraulic Studies

Flow Test of New Fuel Design. A flow test was performed on a PRTR Mk IXB "inverted cluster" fuel element to determine the pressure loss characteristics and flow distribution. The results of the test demonstrated, among other things, that approximately 28.5% of the total coolant flow channeled through the annulus between the fuel element and the pressure tube. Earlier theoretical calculations predicted 30%. This annulus contains almost 50% of the total heat transfer surface. Therefore, it is possible that this element would have to be run at a low power level to avoid selective boiling in the cooling channels.

Heat Transfer Analysis of Proposed Capsule Experiment. Calculations were performed to determine maximum PRTR shroud tube temperatures during irradiation of a creep test capsule. The capsule was planned to be installed in the PRTR without a pressure tube and would be uncooled except for heat losses to the moderator across the gas gap between the capsule and shroud tube. During shutdown of the reactor it will be necessary to maintain the specimen at certain minimum temperatures, and electric heaters will be used to do this. Concern was expressed regarding the temperature of the shroud tube during reactor shutdown since the moderator will be absent from around the shroud tube. The only cooling will result from conduction, radiation, and convection to the surrounding shroud tube. The results of the calculations showed that for specimen temperatures of 400 and 900 C, maximum shroud tube temperatures of 175 and 540 C, respectively, can be expected.

2. Plutonium Ceramic Fuels Research

Plutonium Ceramics Irradiations. PuO_2 , PuC , PuN , $\beta\text{-Pu}_2\text{O}_3$, PuN-W cermet, and PuN-Pu cermet samples are under irradiation in the MTR. Specimens of PuO_2 and PuN are being examined after exposures of about 20×10^{20} fissions/cm³.

The PuO_2 appeared to have developed numerous cracks but did not fall apart when the capsule was disassembled. Metallography revealed a very porous structure with an unidentified light-colored phase within the pores.

The PuN exhibited a few cracks which finally caused the sample to break out of the mounting material during metallographic polishing. The PuN x-ray diffraction pattern indicated structural damage; however, low intensity peaks prohibited lattice parameter determination.

For both the PuO_2 and the PuN , the autoradiographs indicated a higher concentration of radioactive fission products at the edges.

High Temperature Vacuum Pushrod Dilatometer. A high temperature vacuum pushrod dilatometer was put into operation. Maximum temperature capability at present is 1650 C with the present alumina sample holder and pushrod.

Thermionic Emission Measurements. The contact potential difference apparatus for measuring work functions and emission constants was completed. A vacuum of $< 2 \times 10^{-10}$ Torr can be routinely obtained at room temperature.

Explosive Melting of PuO₂. PuO₂ was explosively melted by passing a high current pulse through a direct resistance heated sample. Drops of PuO₂, quenched on a cold wall in the furnace enclosure, are being examined by x-ray diffraction.

Radiation Self-Damage. The increase in the lattice constants of PuC and Pu₂C₃ due to self α -damage are being measured as a function of time at room temperature. The lattice spacing for PuC appears to be saturating after 975 days while for Pu₂C₃ it appears to be still increasing after 500 days.

3. Ceramic (Uranium) Fuels Research

Fuel Element Development. An insulated UO₂ capsule (GEH-14-428) that operated at 2700 C cladding surface temperature released 0.300 cc of a total of 0.70 cc (STP) fission gas generated. The tungsten cladding had cracked during irradiation. Four other high temperature, tungsten insulated UO₂ specimens were irradiated to 0.5×10^{20} fissions/cm³ (1600 Mwd/ton).

ThO₂ Irradiation Studies. Pneumatically impacted ThO₂ having an original specific surface of 5.50 m²/gram released 0.185 and 0.53 cc gas per gram of fuel during irradiation to 1×10^{20} nvt and 2×10^{20} nvt, respectively. A similar element containing Sol-Gel process ThO₂ with a surface area of 0.068 m²/gm released 0.018 cc/gram after an exposure of 2×10^{20} nvt.

No internal corrosion was observed on the X8001 aluminum cladding from four irradiated thorium elements. The elements appeared to have operated sufficiently hot that partial annealing of the cladding occurred on the lower exposure element, and complete annealing on the higher exposure element.

Instrumented ThO₂ Element Irradiation. The central fuel temperature of an instrumented ThO₂ fuel element increased from 480 C to 1050 C during a 30-day irradiation period in the MTR.

Uranium Nitride Irradiation. Two stainless steel clad capsules containing vibrationally compacted, pneumatically impacted uranium mononitride (9.96 and 10.26 g/cm³) were irradiated in the MTR. The UN has a particle density of 14.25 grams/cc. The 1.4 cm OD capsules operated with surface heat fluxes of 134 W/cm² (425,000 Btu/hr-ft²) and 210 W/cm² (656,000 Btu/hr-ft²).

EL-4 Beryllium. A good end closure weld was achieved with a French beryllium cladding tube. This closure has satisfactory grain size in the heat affected zone and is crack free. Arc-cast bar material used for end plugs produced less cracking in the upset weld area than did sintered material.

Stainless Steel Clad UO₂-SS Cermets. Highly loaded UO₂-stainless steel cermet cores were clad with stainless steel tubing and fabricated into tubular sections using the "one-shot" pneumatic impaction process. End caps were formed from stainless steel powder loaded into the tubes prior to impaction. A finish swaging pass over hardened, polished steel mandrels provided uniform OD dimensions, but failed to eliminate some ID non-uniformity.

Tungsten Clad UO₂-Tungsten Cermets. Tungsten coated spherical UO₂ and UO₂-tungsten powder blends were fabricated into tungsten clad UO₂-tungsten cermets with tungsten end caps. Intimate contact and apparent bonding of the cap, core, and clad was achieved. Electrolytic cleaning of the tungsten tubing prior to pneumatic impaction and a post-impaction heat treatment were used to enhance bonding.

Niobium Clad UO₂-Nb Cermets. Highly loaded UO₂-niobium cermet cores were clad with niobium tubing and sealed with niobium powder end caps using the pneumatic impaction process. Good densification of the core and clad, cap, and core bonding were achieved. Electrolytic etching of the niobium tubing and post-impaction heat treatment were used to enhance bonding.

Vibrational Compaction Studies. A computer code for determining optimum particle size distributions for maximum compacted density in cylindrical and annular configurations was developed in co-operation with Applied Mathematics.

Electrical Resistivity of Molten UO₂. Equipment is presently being tested in preparation for an experiment to determine the electrical resistivity of molten UO₂. High inert gas pressure will be used to suppress vaporization during melting. Encouraging results were obtained during preliminary heating and melting runs at low pressure.

High Temperature Testing. The high temperature Brew furnace was successfully operated at 2800 C with a hydrogen atmosphere flowing at the rate of 6200 liter/hr (220 ft³/hr). Redesign of the heating elements eliminated the arcing problem encountered earlier.

Materials and Information Exchange. Four single crystals of UO_2 were sent to General Atomic for study of the irradiation contribution to thermal conductivity. Four UO_2 single crystal specimens were sent to ORNL for high temperature compression studies. About thirteen kilograms of high surface area UO_2 powder of Italian origin was received through EURATOM.

Properties of Sm_2O_3 and Nd_2O_3 . The melting points of pneumatically impacted Sm_2O_3 were measured. Sm_2O_3 melts at 2270 and 2215 C in 8% H_2 -Ar and argon, respectively. Preliminary data show Nd_2O_3 melts at 2165 and 2250 C in 8% H_2 -Ar and argon, respectively. The Knoop hardnesses of Sm_2O_3 and Nd_2O_3 were 448 and 317 KHN, respectively.

Optical Absorption of Single Crystal ThO_2 . The optical absorption spectra for single crystal ThO_2 was measured between 0.3 and 2.5 μ . Transmission is high, from 2.5 μ to 20.4 μ with very weak absorption bands at 0.85 and 2.17 μ . The characteristic absorption edge was measured at 3.9 eV.

Fission Product Relocation. Solid fission products were transferred from an irradiated sample to an unirradiated one along a potential gradient. A gradient of about 1 v/cm was applied during the experiment, while the specimens were held at 1500 C for 24 hrs. The distribution of isotopes after the experiment indicate that fission products were transferred by an ionic conduction mechanism, but that Pu moved by thermal diffusion only.

4. Basic Swelling Studies

Irradiation Program

A general swelling capsule, containing high purity uranium specimens of tubular and rod geometry, of varying diameter and section size, was charged into the reactor. Due to the larger than normal quantity of fissionable material in this capsule, it cannot be controlled at a temperature in the high alpha during reactor operation. Power will be supplied to the capsule to maintain it at a controlled temperature of 700 C (1292 F) which is well into the beta.

A general swelling capsule containing enriched uranium specimens was modified to improve its heat transfer capability. Bench testing had indicated that the capsule, which was fitted with more than the usual number of cooling fins, would not be capable of operating at the prescribed temperature level due to insufficient heat transfer. The end cap was removed and the region between the

cooling fins was filled with stainless steel pellets. Subsequent bench testing showed that the over-all heat transfer coefficient had been approximately doubled. The capsule has been scheduled for irradiation.

Machined components have been received for the first three high pressure irradiation test capsules. Assembly is progressing on the prototype capsule, Pl. This capsule is being assembled for high temperature, high pressure test only and will not undergo irradiation. Testing of the prototype capsule is scheduled for May 1964, and charging of the first in-reactor test is scheduled for June 1964.

Post-Irradiation Examination

Three general swelling capsules containing high purity uranium samples having respective U-235 contents of natural (0.72%), twice natural (1.44%), and four times natural (2.88%) enrichment were opened in Radiometallurgy, and the specimens were successfully recovered and their densities measured. The visual examinations indicated that the condition of the respective specimens was about as expected from a consideration of their respective irradiation histories. Samples in the as-extruded state and in the beta heat treated condition were included in these capsules. There was the expected effect of heat treatment but no obvious effect of burnup rate.

The natural uranium specimens were irradiated to about 0.02 at% B.U. at 525 C (977 F) and for an additional 0.01 at% B.U. at about 425 C (797 F). The density measurements indicated about a 40% volume increase in the as-extruded samples and about 16% in the beta-treated specimen. This latter value may be low due to internal wetting of cracks and tears during the density measurement. Preliminary metallographic examination of an as-extruded specimen revealed crystallographically aligned microtearing and extensive grain boundary cracking and tearing typical of the 525 C (977 F) and 425 C (797 F) irradiation behavior of high purity uranium.

The specimens enriched to 1.44 wt% U-235 were irradiated to 0.16 at% B.U. in a capsule controlled at 625 C (1157 F). The end specimens operated at 575-600 C (1067-1112 F). The as-extruded specimen irradiated at 625 C exhibited a volume increase of 14% while the similar sample irradiated at 575-600 C revealed an increase of about 20%. The beta treated specimen had a volume increase of about 18%. These values tend to be slightly higher than previous data obtained with natural uranium.

The specimens enriched to 2.88 wt% U-235 were irradiated to 0.31 at% B.U. in a capsule nominally controlled at 625 C. Difficulty was experienced in maintaining close control on this capsule so that several excursions into the beta occurred. These excursions coupled with the relatively high burnup in these samples resulted in extreme damage to the specimens. The density measurements indicated 60 to 85% volume increases.

A fourth general swelling capsule containing eight natural uranium specimens representing three compositions in several metallurgical states was also opened in Radiometallurgy, the specimens successfully recovered, and their densities determined. Four high purity uranium specimens, two specimens alloyed with 400 ppm Fe, 640 ppm Al and 500 ppm C, and two specimens alloyed with 140 ppm Fe, 95 ppm Si, and 400 ppm C were included. The capsule was irradiated to about 0.11 at% B.U. in the specimens at a control temperature of 575 C. Due to longitudinal temperature gradients in the capsule the centrally located, four, high purity specimens were irradiated at 575 C while the four alloy specimens located at either end of the capsule were irradiated at about 535 C (995 F). Visual examination revealed that the behavior of the alloy specimens was greatly superior to that of the high purity specimens. There were also positive indications that the Fe-Al alloy behaved better than did the Fe-Si alloy. Furthermore, there appears to be a distinct difference in the behavior of any one material, depending on the metallurgical state.

The density measurements revealed that the high purity uranium samples increased in volume by 20 - 35%, whereas the alloy specimens showed increases varying from 0 to 13 wt%. An Fe-Al alloy specimen and an Fe-Si alloy specimen [vacuum annealed at 800 C (1472 F) for two hours, furnace cooled to room temperature and reheated to 730 C (1346 F) for 15 minutes and quenched into silicone oil prior to irradiation] both of which were irradiated at the same position in the capsule showed respective volume increases of 0 and 8.5% (actually the Fe-Al alloy specimen showed a volume decrease of 0.4%). Calculations of the volume changes of the other pair of alloy specimens, which had received a standard beta heat treatment in the form of a large diameter rod prior even to specimen machining, showed that the alloy with Fe-Al increased in volume by about 3% while that with Fe-Si increased by about 13%.

All eight of these specimens were irradiated in the temperature range where severe fission event and defect recovery damage have been observed in high purity uranium. The small alloying additions

have significantly reduced this type of damage inherent in pure uranium in the 525-575 C temperature range. Further, the greater amount of alloying addition in the Fe-Al alloy results in still greater improvement. Detailed studies of the microstructures of these specimens are in progress.

5. Irradiation Damage to Reactor Metals

Alloy Selection

Corrosion and oxidation tests of ten nickel base alloys are continuing as a part of a program to evaluate the applicability of these alloys as nuclear structural materials. Oxidation tests were completed by Chemical Metallurgy personnel on specimens exposed for 100 and 200 hours to 15 Torr water vapor in helium at 1500 F (816 C). Specimens are presently being exposed to this environment for 300 hours. Upon completion, the data will be compared with similar data for specimens exposed at 1700 F (927 C) and 1900 F (1038 C). Additional specimens are also being tested in 650 C (1200 F) steam at 3000 psi. Weight gain data for exposures to 1000 hours have been obtained.

Tensile specimens and oxidation coupons of TZM, Cb-752, Cb, Ta, Ta-10 W, and Cb-1 Zr were exposed to atmospheric air and dry air at a temperature of 260 C (500 F) for 14 days. These studies were made to determine whether or not any significant changes in mechanical properties might result if such specimens irradiated in the ATR gas loop facility were exposed to dry air. The weight change data indicate that small amounts of oxidation have occurred. Weight gains for these materials varied from about 0.004 mg/dm² for TZM to 0.03 mg/cm² for tantalum. The scatter in data make it impossible to distinguish between specimens exposed to dry and to atmospheric air. The determinations of mechanical properties and metallographic examination will be made during the next month.

Tensile specimens of Hastelloy X-280, Inconel 625, Incoloy 800, and Inconel 600 irradiated at 50 C (122 F) to exposures of 9.8×10^{19} , 4.61×10^{20} , and 7.64×10^{20} nvt ($E > 1$ Mev) were tested at room temperature during the past month. These data showed an increase in strength with small decreases in uniform elongation. The most significant changes were in the Inconel 600 alloy irradiated in the solution annealed condition. This alloy showed an increase in yield strength of 150% of the original value after an exposure of 9.8×10^{19} nvt. An increase of yield strength to 210% of the original value resulted from an exposure of 7.61×10^{20} nvt. The

uniform elongation was decreased from 42% to 1% after an exposure of 7.61×10^{20} nvt. The total elongation at this exposure was about 9%.

In-Reactor Measurement of Mechanical Properties

Two in-reactor creep tests on annealed 304 stainless steel are in progress. The stress for both tests is 20,000 psi. Test temperatures are 500 C (932 F) and 550 C (1022 F). The test conducted at 500 C is deforming so slowly that more testing time will be required before a creep rate can be reported. The 550 C test has accumulated more than 550 hours and the specimen is deforming at a rate of 2.3×10^{-6} /hr.

The creep curve for an ex-reactor control test conducted at 20,000 psi stress and 550 C is similar in shape to the in-reactor curve. The ex-reactor creep rate at 300 hours is 2×10^{-6} /hr.

A third in-reactor creep test on 304 stainless steel, to be conducted at 20,000 psi and 600 C (1112 F), is ready for reactor insertion.

Irradiation Effects in Structural Materials

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

Several tensile tests were performed at 300 C (572 F) on specimens of AISI 304, 348, 410, and Am-350 stainless steels, which were irradiated in the G-7 hot water loop at approximately this same temperature. A high temperature extensometer was used in these tests to measure strain to a point beyond the 0.2% yield strength and crosshead motion was used to measure the subsequent extension to fracture. Additional tests were performed at 300 C on unirradiated materials in the as-fabricated condition. Comparison of 300 C and room temperature tests on both the irradiated and unirradiated materials shows that reduced strength and ductility are characteristic of tests at elevated temperatures.

Room temperature tests of the AISI 304 and 348 irradiated at 50 C (122 F) and at 750 C (1382 F) show that the maximum radiation-induced change in tensile properties of these alloys occurs at

some intermediate irradiation temperature. Additional elevated temperature tests and irradiations are being performed to determine under what combination of irradiation and environment these materials exhibit minimum strength and ductility.

The effort to modify microstructures of nickel-base alloys to reduce their adverse reaction to irradiation is continuing. Structural changes are made by preirradiation cold work and heat treatments. Tensile and stress-to-rupture tests are being used to screen these treatments. Four specimens of Inconel X-750, a precipitation-hardened alloy, were discharged from the ETR during the month. They had been given an experimental heat treatment to provide a finely dispersed in-grain γ' precipitate and a reduced grain boundary precipitate. Four controls were also fabricated and heat treated in like manner.

Results were presented last month on the effect of various aging treatments on the room temperature tensile properties of control and irradiated specimens of Hastelloy X-280, a solution-hardened alloy. Comparison of these results with those obtained at NMPO has led to a hypothesis that irradiation of Hastelloy X-280 in the solution-treated condition causes carbides to precipitate at irradiation damage sites within the matrix, leading to increased strength and satisfactory ductility. At the same temperature ex-reactor, these carbides would precipitate in a continuous grain boundary phase causing a reduced ductility. To test this hypothesis, tensile specimens were fabricated in several combinations of 30% cold work (to introduce areas of high strain energy) and aging treatments (to allow precipitation to occur). R_A hardness measurements showed increases of 11 to 14 points over controls aged without cold work at the same temperatures.

An important factor in nuclear power plant design is the resistance of construction materials to stresses and strains caused by rapid temperature changes resulting from response to power demands. These effects may be severe, leading to fatigue damage in a relatively few cycles. An investigation of the effects of pre-test treatment, test temperature, and irradiation on the low cycle fatigue life of nickel-base alloys has been initiated. To provide a high resistance to buckling, specimens were designed with an hourglass shape rather than a cylindrical one. Since axial strain measurement has little significance, a diametrical strain gage is being fabricated. It has been designed to remain aligned with the minimum section of the specimen as it moves axially during the compression-tension cycle.

Environmental Effects

Hydrogen analyses were completed during the report period on all specimens of Zr-2 irradiated in Quadrants 73, 74, 85 and 86. Fractional absorption of theoretical corrosion hydrogen ranged from 13 to 18% in Quadrant 73; 13 to 22% in Quadrant 74; 23 to 55% in Quadrant 85; and 22 to 41% in Quadrant 86. Weight gain and exposure data for these quadrants were previously reported in the August and November 1963 monthly reports.

These results are within the range previously reported for 12 other quadrants of irradiated material. Again, no consistent variation of hydrogen pickup with either neutron flux or cold work was observed. The surface color of the oxide film on specimens exposed in Quadrant 85 and 86 was more brown than dark grey or blue-black as observed for all other irradiated Zr-2 examined to date.

Both fully annealed and stress relieved molybdenum sheet made from arc cast material oxidized at the same rate in 1×10^{-3} mm Hg partial pressure of oxygen at 1000 C (1832 F). The metal oxidized linearly due to the formation of a volatile oxide, presumably MoO_3 . The metal penetration at 4.38 days was 0.6 mil.

ATR Gas Loop Studies

Mark I Test Section. Fabrication of the model gas loop Mark I test section has been completed. General features incorporated into this test section, which also apply to the ATR test section, are the same diameters and materials, possible tube closures, and types of insulation compatible with attaining and maintaining ultrapure helium in the coolant system.

Castable Fiberfrax Insulation Fabrication. A report⁽¹⁾ was written describing the techniques developed for fabricating the castable Fiberfrax insulation used in the model gas loop. In both the model and ATR gas loops, coaxial piping is used for gas temperatures which exceed 871 C (1600 F). Ultrapure helium coolant (< 1 vpm total impurities) at temperatures to 1149 C (2100 F) is contained in the inner pipe of Haynes 25 superalloy. The outer pipe of stainless steel is pressurized with ordinary purity helium to within 10 psi of the nominal 350 psi helium in the inner pipe. Accordingly, the pressure stresses on the hot inner pipe are reduced to those caused by this small differential pressure. Insulation is required in the annulus between the two pipes to maintain the outer pipe below its

(1) Doman, D.R., "Fabrication Techniques: Cast Fiberfrax Insulation for the Model High Temperature Gas Loop," HW-81075, 3/2/64.

maximum code allowable temperature. This insulation also supports and centers the inner pipe.

Criteria established for selecting insulating material for this application indicated castable Fiberfrax would most nearly meet the requirements. Techniques were developed for fabricating the Fiberfrax insulation into the shapes of reducers, bends, and straight pieces. Basically the insulation is made by tamping castable Fiberfrax into a metal or plaster mold of the desired shape with wire or mesh reinforcement added during tamping. Removal of the inherent moisture by heating produced the required cast piece.

6. Gas-Cooled Reactor Studies

Radiation-Induced Reactions of H₂-CO and CH₄-CO₂

In-reactor experiments at approximately 550 C in capsules containing H₂-CO mixtures have been described earlier in HW-78948. The results showed CH₄ and CO₂ as products of the reaction.

Investigation of this reaction was continued at approximately 30 C using a Co-60 facility. The reaction was studied from both the product and the reactant side of the equilibrium. Equimolar mixtures of H₂ and CO and CH₄ and CO₂ were irradiated in sealed capsules at pressures between 0.5 and 0.8 atm. The exposures ranged from 4.9×10^8 to 6.9×10^8 R for the H₂-CO capsules and from 1.2×10^8 to 6.0×10^8 R for the CH₄-CO₂ capsules.

For the H₂-CO in-reactor experiments, the ratio of formation CH₄:CO₂ was approximately 1.5:1.0; however, for the experiments in the gamma facility, the ratio was only 0.18:1.0. In the earlier experiments it was suggested that CO produced carbon suboxide polymers which subsequently decomposed into CO₂ and a carbon film on the capsule walls and that the CH₄ was then formed by the reaction of H₂ with the carbon film. In contrast, the carbon suboxides formed in the low temperature, gamma-induced reactions are apparently more stable and therefore do not furnish a carbon film with which the H₂ can react to form CH₄.

In the CH₄-CO₂ capsules there was a significant decrease of CH₄, ranging from 8 to 23%. A smaller decrease of CO₂ was observed, ranging from 2 to 4%. The products of the reaction were H₂ and CO and smaller concentrations of C₂H₆ and C₃H₈. From the limited number of capsules, it would be difficult to propose a reaction

mechanism, although one may speculate on some of the possible steps. It is possible that the following reactions may occur: (1) the radiation-induced reaction of CH_4 to form higher hydrocarbons which, in turn, decompose into H_2 and a carbon film; and (2) the radiation-induced reaction of CO_2 with the carbon film to produce CO by the reaction, $\text{C} + \text{CO}_2 = 2\text{CO}$. As support for step (2), there was substantially greater loss of CO_2 (45% of the initial quantity) in one of the CH_4 - CO_2 capsules that contained a large sample of graphite.

Radiation-Induced Reactions of Nitrous Oxide. Nitrous oxide resembles CO_2 and H_2O in its reactions with carbon at high temperatures. In contrast, however, at 400 C (a temperature too low for the CO_2 -C and H_2O -C reactions) N_2O reacts with C to form CO_2 and N_2 . This is believed to occur by a catalyzed reaction on the carbon surface.

Since the reactions, $2\text{N}_2\text{O} + \text{C} = \text{CO}_2 + 2\text{N}_2$ and $\text{N}_2\text{O} + \text{CO} = \text{CO}_2 + \text{N}_2$, have a large negative free energy change even at room temperature, it was anticipated that gamma radiation would initiate these reactions. Therefore, experiments using N_2O -carbon and N_2O -CO mixtures were performed. Two capsules containing carbon respectively as graphite or activated coconut charcoal were filled with N_2O at a pressure of approximately 0.7 atm and were exposed at approximately 30 C to a dose of 2.4×10^8 R. Analysis of the gases after irradiation indicated that reaction had occurred and that N_2 , CO , CO_2 , NO and possibly carbon suboxide polymer were reaction products.

The N_2O -CO experiments involved irradiation of equimolar mixtures to 4.5×10^8 R at 30 C in the presence of graphite or charcoal. The results showed CO_2 , N_2 and NO as products with greater extent of reaction associated with the high-surface-area charcoal.

EGCR Graphite Irradiation

Satisfactory progress is continuing in the series of long-term irradiations of EGCR graphite, as the eighth capsule in the series, H-3-8, was installed in the GETR and is operating successfully. The modification to the top capsule position appears to have been quite successful with temperatures running $300 \text{ C} \pm 15 \text{ C}$. All other sample temperatures are in good agreement with former capsules.

Preliminary information is now available on the determination of sample exposures based on activation measurements of monitors of nickel, iron, and titanium located in each of the irradiation

capsules. Sample exposures based on these activation rates have been calculated through use of neutron spectral distributions recently computed using 18 energy groups (of which 16 are greater than 0.18), more accurate cross-section data obtained in the last year, and resolution of problems of counting of monitor activity. These exposures are more accurate than the previously reported exposures (which were based on group one, $E > 0.18$ Mev, of the three-group computation made each reactor cycle by the GETR physics group) since they are based on measurements of the neutrons in the high energy portion of the spectrum, the region of primary interest relative to irradiation effects in graphite. Based on data from the flux monitors contained in the first six capsules in the program, the neutron exposures are only approximately 60% of those previously reported. Consequently, all sample exposures in HW-71500A and subsequent data from the H-3 irradiation program should be reduced to 60% of the former values.

Effect of Irradiation on Tensile Strain Limits

Previous reports have shown that the high contraction rates of certain carbons can be used in a suitable assembly to produce a tensile strain in graphite specimens. Irradiations have been carried out at two different strain rates, 0.07 and 0.15% per 10^{20} n/cm². Recent inspection of four assemblies that have accumulated approximately 1.8×10^{20} n/cm² ($E > 0.18$ Mev) at 625 C has shown that the two higher-strain-rate specimens have failed at a strain between 0.25 and 0.3% at 625 C. These limits are the sum of the strain measured on unfailed specimens at room temperature and the strain due to the thermal expansion of the strain assembly at 625 C. Work is in progress to refine this estimate.

The information to date indicates that this type of design will generate predicted tensile strains within reasonable limits and thus provide a simple method of producing high strains at 625 C in a reactor environment. These irradiations will continue at lower strain rates and to higher exposures.

7. Graphite Radiation-Damage Studies

Thermal Expansion Behavior of Molded Graphite

A series of molded graphite bars manufactured by Battelle Memorial Institute⁽¹⁾ which differed only in their graphitization temperatures

(1) Hedden, W.A., L.D. Loch, J.A. Slyh, W.H. Duckworth, BMI-962 (1954) 38.

were investigated to determine their thermal expansion behavior. Six samples were cut from each 1 x 3 x 6-inch graphite bar so that two were parallel to the molding-pressure direction; two were transverse to the molding pressure direction and parallel with bar length; and two transverse to the molding pressure direction and transverse to the bar length. The thermal-expansion measurements were made with a Brinkmann 25-1600 C dilatometer.

The average orientation of the graphite crystallites was determined from equations which relate bulk thermal expansion, crystal lattice thermal expansion, accommodation coefficient and crystallite orientation.

The results showed that the accommodation coefficient and degree of orientation were both relatively insensitive to graphitization temperature and that if microcracks form during cooling from graphitization temperatures, they must probably form below 1800 C.

The calculated orientation factors agreed well with orientation ratios determined by Battelle Memorial Institute from x-ray data.

Annealing of High Temperature Radiation-Induced Contraction

Annealing an irradiated graphite sample has been found to cause a drastic change in the effect induced in that sample during a subsequent irradiation. A series of six transverse and six parallel CSF samples are being irradiated in "hot capsules" to check the effect of annealing to 2500 C on the damage rate in subsequent irradiations. The second irradiation has been completed at a temperature estimated to be 600 C and incremental exposure of 4 to 7 x 10²⁰ nvt. The contraction in the unannealed control samples during the second irradiation was an average of 5.6 ± 1.0 times as great as the samples that had been annealed to 2500 C after the first irradiation. No significant orientation difference was noted. After the second irradiation the samples were again annealed to 2500 C, and a further contraction was noted in both orientations in the previously annealed samples causing the above ratio to fall to 3.6 ± 0.9 .

The behavior of the CSF samples is the opposite of that noted for lampblack-based samples during similar experiments. Thus it seems that for CSF the effect of annealing is to remove damage centers which participate in the contraction mechanism. As a result, during the second irradiation, the annealed samples appeared to retrace the curve of unirradiated samples, which display a slight expansion or a

low contraction rate during the early stages of irradiation. The control samples on the other hand followed the normal contraction behavior and contracted more rapidly during the second irradiation than the first.

These samples have been returned for further irradiation at the ETR.

8. Boronated-Graphite Studies

The second capsule of the long-term irradiation in the 2C test hole at KW Reactor is performing satisfactorily with all temperatures within specification and approximately the same as in the first irradiation. This capsule contains 22 samples from the first capsule. The static gas pressure in the capsule has increased during irradiation indicating a buildup of helium from the (n, α) reaction. The measurement of dimensional changes of specimens exposed to about 10^{20} nvt, $E > 0.18$ Mev in the first run has been completed.

Internal consistency of the measurements was very good on replicate samples under the same experimental condition. However, the variation of results with material and experimental parameters was not sufficient to permit unequivocal identification of behavior with material type or irradiation temperature at this time. The experimental results can be summarized qualitatively as follows:

1. With the exception of one material, the 7 wt% "grey", all samples contracted in the direction parallel to the extrusion direction at a rate far greater than CSF graphite. Half the specimens contracted 0.20% or more.
2. Four out of five of the 7 wt% "grey" material specimens showed expansion. This behavior may be attributable to the highly recrystallized nature of this material which is unique among the samples irradiated in this study.
3. In the direction transverse to the extrusion direction, expansion occurred in all specimens irradiated at 370 C and in the majority of specimens irradiated at 480 C. About half of the transverse samples expanded more than 0.05%.

An irradiation of 11 samples at 340 C (650 F) to 10^{19} nvt ($E > 0.18$ Mev) was successfully concluded. This test should provide an intermediate exposure point for the long-term irradiation.

Compressive strength of unirradiated "black" 5 and 7 wt% samples is about 4500 psi for 0.450-inch diameter by 0.750-inch long samples; for "grey" material, it is about 2500 psi. The strength appears to be independent of boron concentration and orientation.

The strain energy to failure (in compression), defined by the area of the stress-strain diagram, is independent of the concentration of boron and the type of material for longitudinal samples, averaging about 60 in.-lb/cu in. For transverse samples, the compressive strain energy to failure was slightly higher, more scattered and tended to show some effect of the type of boronated graphite, i.e., black or grey. High strain energy was associated with samples of higher ultimate strength.

9. Metallic Fuel Development

Irradiation of Thorium-Uranium Fuel Elements. The irradiation of three tubular Zircaloy-2 clad thorium - 2.5 wt% uranium fuel elements continued in the ETR P7 loop. The fuel elements have achieved an integrated exposure of 2.1×10^{20} fissions/cm³ (6200 Mwd/ton). The fuel elements are operating at a maximum temperature of 520 C, a surface heat flux of 62 cal/sec-cm² (8.3×10^5 Btu/hr-ft²) and a specific power of 57 watts/gm (174 kw/ft). Weight measurements made at an integrated exposure of 2.1×10^{20} fissions/cm³ (6200 Mwd/ton) indicated a maximum volume increase of 0.8% which is the same as that measured at an exposure of 1.8×10^{20} fissions/cm³ (5200 Mwd/ton). The volume expansion of the fuel continues to show no evidence of swelling due to agglomeration of fission gas.

Irradiation of Fine Carbide Uranium Fuel. In a test to investigate the ability of a submicron dispersion of uranium carbide to reduce uranium swelling, three NaK-filled capsules will be irradiated in the ETR to evaluate the performance of fuel rods produced from chill-cast uranium shot. Two of these capsules each containing two fuel rods identical in uranium composition, but with one having a uranium carbide size of 2-5 microns produced from arc-melted uranium and the other a carbide size of less than 0.5 micron produced from the uranium shot. The third capsule contains two fuel rods with the fine carbide. One capsule, GEH-14-609, which has completed its third cycle of irradiation, has accumulated an exposure of 0.20 at% burnup. The capsule is operating at a maximum core temperature of about 550 C. Goal exposure for this capsule is 0.3 at% burnup.

Thorium Defect Testing. Ten new thorium alloys have been fabricated into Zircaloy-2 clad, coextruded rod stock 0.525 inch in diameter.

Nominal alloy compositions are Th - 2.5 U and Th - 5.0 U plus tertiary alloys containing zirconium of 1, 2, 5, and 10%. All alloys were coextruded at 760 C with a reduction ratio of 17.4 to 1. Wafers from the front, middle, and rear of each extrusion have been taken for metallography. Specimens for defect testing in a windowed autoclave are being prepared.

Dispersed Phase Thorium Alloys. One phase of the metallic fuel development program is the study of quaternary additions to thorium base, solid solution strengthened, Th-U-Zr ternary alloys for development of finely dispersed intermetallic compounds in the structure. The dispersed phases are expected to contribute to additional irradiation swelling resistance in these high strength, corrosion resistant alloys. Initial alloys being studied are aluminum or beryllium additions (500 to 2000 ppm) to Th - 2.5 or 5 wt% U - 2 or 5 wt% Zr alloys. Both series of alloys were prepared as 100-gram arc melted buttons. Attempts to hot roll the alloys resulted in severe edge cracking when heating was done in neutral salt at either 850 or 900 C. Four alloy buttons containing 2.5 or 5.0 wt% U and 2.0 or 5.0 wt% Zr with 1000 ppm Al were machined to fit the $1\frac{1}{4}$ inch diameter x $\frac{3}{8}$ inch high cavity of a stainless steel compact and were sealed by electron beam welding. The compacts were heated to 775 C and hammer forged to approximately $\frac{2}{3}$ reduction in height. Further rolling was done from 750 C preheat for a 70% reduction. After removal of the sheath, the alloy sheet was 0.048 inch thick. No edge cracking was noted and the surface appeared excellent.

The success of this fabrication technique is attributed to breakup of the cast button structure by hot forging and the restraint afforded by the stiff stainless steel compact. The balance of the alloys are being prepared for fabrication and subsequent heat treatment.

10. Aluminum Corrosion and Alloy Development

The second charge of aluminum-clad fuel elements undergoing corrosion tests in C-1 Loop was discharged after a 25-day exposure at 260 C outlet temperature and neutral pH. A visual examination in the C-basin revealed no serious corrosion or other abnormalities. Only surface roughening was present with no nonuniform corrosion. Red-brown crud deposits were seen only on the unheated ends of the fuel elements and on the upstream half of the thermocouple element (the upstream element).

A third test was started at pH 4.5. After two days the loop dumped to process water. Since this exposure to low purity water may affect corrosion, the loop was recharged at the next outage.

The fourth charge is being exposed at 260 C outlet temperature, pH 4.5 with phosphoric acid. Little rise in the cladding temperature of the upstream fuel element has occurred after 10 days.

11. USAEC-AECL Cooperative Program on Development of Heavy Water Moderated Power Reactors

Effect of Orientation on Heat Transfer Characteristics. An experimental program was started to determine the difference in the boiling burnout characteristics of 19-rod bundle fuel elements operating in horizontal and vertical positions. These experiments are being performed on a full-scale, electrically heated model of a 19-rod bundle element with 0.587 inch OD rods, 6.3 feet long, installed in a 3.25-inch ID pressure tube. Rod spacing of 0.050 inch is maintained by wire-wrap with a nine-inch pitch on 12 of the rods. This model, in the horizontal position, was used in an earlier investigation of boiling burnout, and these tests are now being re-run with the model in the vertical position for comparison.

Two boiling burnout conditions have been determined which could be compared with data gathered earlier on the horizontal model. Both of these were obtained with a coolant flow rate of 0.5×10^6 lb/hr-ft² and an outlet pressure of 1200 psig and are summarized in the following table:

<u>Inlet Temperature</u>	<u>Boiling Burnout Heat Flux (B/hr-ft²)</u>	
	<u>Horizontal</u>	<u>Vertical</u>
232 F	276,000	305,000
510 F	174,000	220,000

12. Advanced Reactor Concept Studies

Fast Supercritical Pressure Power Reactor. A paper describing this reactor was given at the Ninth Nuclear Superheat Meeting in San Jose, California, on April 21.

Clearance for unrestricted publication of the report on this reactor was received from the Chicago Patent Office, and the report was given category distribution.

Segmented Fast Reactor (SFR). Two cores were selected for nuclear analysis from the preliminary parametric study reported last month. A preliminary data summary for these cores is shown in Table I. These cores have a higher coolant volume fraction than the two previously selected. Detailed mechanical calculations will not be made until the results of the nuclear analysis are available to give more accurate power distribution and flux peaking factors.

TABLE I

1000 Mwe SFR - PRELIMINARY DATA SUMMARY

	<u>Core III</u>	<u>Core IV</u>
Core Diameter (ft)	7.95	6.49
Core Length (ft)	3.25	4.86
Core L/D	0.409	0.749
Core and Axial Blanket Power (Mwt)	1890	1890
Total Reactor Power (Mwt)	2333	2333
Core Power Density (kw/l)	500	500
No. of Core Fuel Assemblies	426	426
No. of Fuel Pins per Assembly	169	91
Fuel Pin OD (in.)	0.246	0.215
Pin Cladding Thickness (in.)	0.015	0.015
Distance Across Flats of Assembly (in.)	4.09	3.34
Core Volume Fractions		
Fuel	0.426	0.253
Coolant	0.400	0.600
Can and Cladding	0.174	0.147
Core Coolant Flow Rate (lbs/hr)	81.5×10^6	81.5×10^6
Maximum Coolant Velocity (ft/sec)	35	35
Coolant Inlet Temperature (°F)	927	927
Average Coolant Outlet Temperature (°F)	1200	1200
Maximum Coolant Outlet Temperature (°F)	1350	1350
Maximum Clad OD Temperature (°F)	1400	1400
Maximum Fuel Temperature (°F)	4793	5603*
Maximum Fuel Surface Temperature Assumed (°F)	2000	2000

*Actually molten near center of pin.

D. DIVISION OF RESEARCH - 05 PROGRAM

1. Radiation Effects on Metals

This program is directed toward establishing the combined effect of impurities and neutron irradiation on the properties and structure of specific metals, and deducing from thermally activated recovery processes how the damage state can be altered. Present studies involve single and polycrystalline specimens of molybdenum, nickel, and rhenium.

Transmission electron microscope studies of wafers cut from irradiated fractured single crystals of molybdenum has continued. One crystal containing 450 ppm carbon which had been irradiated to 10^{18} nvt was sectioned parallel to the (110). No channels were observed in any of the wafers cut from this crystal. In the gage section, where deformation was greatest, complex dislocation tangles were present. In some cases the tangles developed into a system of subgrain boundaries. Defect structures, approximately $10^{16}/\text{cm}^2$, due to neutron irradiation were observed in the grip end of the crystal. These observations are in agreement with those made on polycrystalline molybdenum foils which were irradiated to a similar exposure. The absence of channels in the foils and single crystal samples is presumably due to an insufficient number of defects in the matrix, thus permitting slip to occur on a number of systems simultaneously. Higher exposures create more defects, which restrict the number of operative slip systems, thus producing channels. Single crystal samples which have been irradiated to 10^{19} nvt and oriented so that the surface of the section is parallel to the (110) are being prepared for electron microscopy.

In order to determine if the formation of channels is a unique deformation behavior of body-centered cubic metals, high purity copper (a face-centered cubic metal) was irradiated to 10^{19} nvt. Transmission electron microscopy revealed defects in the form of small spots and larger prismatic dislocation loops. The irradiated foils were deformed approximately 2% in tension and examined.

Channels similar to those observed in molybdenum were observed in the deformed foils. One micrograph and accompanying electron diffraction pattern have been analyzed and show that the channels correspond to traces of the (111) on the plane of the foil. This result was expected as the (111) is normally the slip plane in face-centered cubic metals. Complex dislocation tangles between the channels indicate that slip may not be as restricted in face-centered cubic as in body-centered cubic metals. However, the existence of channels in both metals suggests that the deformation process in irradiated face-centered cubic and body-centered cubic metals is very similar.

Twelve single crystals of molybdenum, the lattice parameter and the length of which have been determined to 0.001%, have been encapsulated and submitted for irradiation. The specimens were divided into three groups with goal exposures of 1 and 5×10^{19} and 1×10^{20} nvt. After irradiation measurements will be repeated to determine changes as a function of irradiation exposure.

A single crystal of semiconductor grade silicon was used to verify the accuracy of the symmetrical back-reflection technique for determining lattice parameters. Five observations yielded a value of a_0 for silicon of $5.43050 \pm 0.00002\text{\AA}$, uncorrected for refraction. This result is in perfect agreement with accepted values.

A number of calculations have been made concerning the configuration and distribution of defects in quenched nickel. From a consideration of strain energy, dislocation loops should be more stable than spherical or disc-like clusters of vacancies when the number of vacancies in an agglomerate exceeds 9 or 10. However, there should be an effect on the cluster configuration due to internal magnetic fields. Associated with every cluster of vacancies should be an electrostatic interaction energy arising from local demagnetization of the lattice. This electrostatic interaction will not have an effect on the configurations resulting from quenching, since most of vacancy jumps during quenching are made while the metal is in a paramagnetic state. The effect will appear upon subsequent annealing at temperatures below the Curie temperature. The yield strength of quenched nickel tensile specimens which were annealed above the Curie temperature was significantly greater than that of specimens annealed below the Curie temperature for an equivalent time. The hypothesis locally advanced is that the vacancies form stable complexes such as tetravacancies or pentavacancies during quenching; a number of monovacancies may be retained. Upon annealing below the Curie point, the less stable complexes dissociate and the vacancies migrate in part to the more

stable complexes and form spherical cavities (which have the lowest electrostatic interaction energy) comprised of some 10-20 vacancies. Annealing above the Curie temperature results in enhanced vacancy mobility, and the spherical clusters either grow and then collapse into discs (thus reducing the strain energy) which then grow and finally collapse to form dislocation loops. Attempts are being made to confirm this hypothesis by direct observation with transmission electron microscopy. Small and yet unidentified defects have been observed in 99.997% pure nickel which was quenched and then annealed at 700 K (above the Curie temperature, approximately 638 K).

Rhenium single crystal rods were cut into small sections with (001) faces and (100) faces, using a water-cooled diamond saw blade. Difficulty was encountered in preparing the crystal faces on this particular metal, as cutting methods and polishing techniques had to be investigated to find a suitable process. X-ray lattice parameter measurements yielded values of $A = 2.7604\text{\AA}$, $C = 4.4599\text{\AA}$, with $C/a = 1.6157$. Accepted values (Vol. III, International Tables for Determination of Crystal Structure) of these parameters are $2.760\text{\AA}$, $4.458\text{\AA}$, and 1.615 , respectively. In the present experiments there is some evidence of cold work and a few subgrains. Some additional effort will be made to refine the techniques for preparing strain-free surfaces.

Studies are being made of the slip band structure formed by bending in irradiated molybdenum single crystals. A device for bending of the crystals has been constructed. The device allows for the observation under the optical microscope and for replication of the surface while the sample is under tension. The slip band structure in an unirradiated molybdenum single crystal is currently being studied to establish the proper experimental techniques. Results to date indicate a lack of well defined slip lines in the unirradiated crystal, even after very severe bending.

Annealed polycrystalline rhenium foil tensile specimens 0.075 mm thick x 3.25 mm wide were tested in tension at 300 and 78 K. The yield stress (0.2% offset) was 21.2 and 30.8 Kg/mm^2 at 300 and 78 K, respectively. The strain-hardening coefficient, n , was 0.46 at 78 K and only slightly less at 300 K.

2. Plutonium Physical Metallurgy

The objective of this program is to determine some of the basic physical metallurgical properties of high purity plutonium and to establish the effect of certain specific alloying additions on those properties.

The investigation of the steady state creep characteristics of the six allotropes of high purity as-reduced and vacuum cast plutonium have been completed. The activation energies for the steady state creep and the effective melting temperatures have been calculated for each allotrope. They are:

Phase	Effective Melting Temperatures		Steady State Creep Activation Energy Cal/Mole
	°C	°F	
Alpha	290	554	19
Beta	475	887	28
Gamma	530	986	33.5
Delta	570	1058	30
Delta Prime	570	1058	28.5
Epsilon	640	1184	25

Experiments have shown that high purity plutonium containing no microcracks can be produced by quenching to -80 C (-112 F) from the beta phase during casting. Metal cast in this manner after 17 thermal cycles between -80 C and +180 C (356 F)(beta phase) contained no microcracks as revealed by metallography and density measurements. No significant density variation was observed before and after cycling.

A method suitable for measuring minute volume changes associated with transformation of retained phases, damage, and related phenomena is being investigated. Attempts will be made to obtain densities with a variation of less than $\pm 0.002 \text{ g/cm}^3$.

The pole figure goniometer to be used for the study of the deformation texture of plutonium has been installed on the enclosed diffractometer. Prior to any quantitative pole figure work it was necessary to establish the absorption corrections to be applied to the textured samples. These corrections are derived from the pattern of a completely random, fine-grained sample. In the case of alpha plutonium it has become apparent that such a structure is not readily obtained. Normally cooled as-cast samples yielded relatively wide variations in the intensity of the (020)-(211) line, when the sample was rotated in the plane of the cross-sectional surface at a constant 2θ value of 37.43° . While this suggests a coarse-grained material, such a condition is not confirmed metallographically. It seems possible that the results obtained can be explained if large areas of fine alpha grains have essentially the same orientation formed during transformation of a large grained phase normally stable at an elevated temperature.

It has been shown that quenching from the beta phase to room temperature improves the randomness of the structure over that obtained by slow cooling. A sample is currently being prepared which has been quenched from the gamma phase. It is hoped this will provide the required structure.

The elevated temperature specimen holder for the diffractometer has also been received and is currently undergoing vacuum testing. The necessary vacuum equipment for installation in the glovebox is on hand and the furnace will be installed as soon as it has been checked.

E. CUSTOMER WORK

1. Radiometallurgy Laboratory

The results of routine examinations and measurements are, or will be, reported as part of the sponsoring research and development programs. During the period March 28, 1964 to April 28, 1964, Radiometallurgy Laboratory output included:

Metallography samples processed	70
Photomosaics	11
Autoradiographs	11
Replicas (for electron microscopy)	21
Burnup dissolutions	16
Fission gas sample collection	10
Vacuum induction furnace runs	7
Decladding dissolutions	37
Tensile tests	86
Rockwell hardness tests	103
Density determinations	58
X-ray analyses	4
Negatives processed	302
N-Reactor thermo quench tests	3

Methods

The behavior of zirconium-clad, PRTR PuAl fuel rod sections in a typical aluminum decladding solution was investigated. Two fuel rod sections were capped (press fitted) with zirconium caps, and two sections were left uncapped. After reaction in the solution for about two hours, the PuAl and zirconium cladding of the capped

sections was unaffected; the PuAl completely dissolved out of the uncapped sections leaving zirconium cladding intact. This investigation was made to support Radiometallurgy's method of shipping capped fuel rod sections.

Room temperature curing, plastic compounds were tested and found satisfactory for fuel element replication work. Use of room temperature vulcanizing plastic allows a more simple in-cell process to be utilized.

Preliminary work was started on equipment needs for remote spot welding of strain gages on irradiated process tubing before burst testing.

Equipment

Annealing Furnace. During functional testing of the annealing furnace and furnace controls, the controller and programmer failed to operate as specified. Trouble was also encountered when water lines began leaking during cycling of the furnace. The vendor was contacted and it was decided to again return the furnace for repair. The vendor was also requested to calibrate and adjust the complete system at Radiometallurgy following return of the furnace.

High Temperature Tensile Testing Machine. The new high temperature tensile testing machine has been set up outside of the cell for functional testing and operator training. After completion of these tests, the machine will be installed in "H" cell. Tentative completion date is June 30, 1964.

Heavy Duty Extended-Reach Manipulator. The new extended-reach Model "D" manipulator was received and installed in "F" cell where improved in-cell handling and coverage will result.

Hot Cell Stereo Zoom Microscope. Design was completed for a motorized positioner to be used with the stereo zoom microscope and fabrication is in progress. Tests designed to determine lighting requirements for viewing and photography through the microscope were completed and design of the lighting fixtures was started.

Microhardness Blister for "I" Cell. Work continued on design and fabrication of controls for the remote microhardness tester.

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Remote Belt Sander. Preliminary design was started on a sample holding device to position samples on the remote belt sander and a coolant dispensing system to keep samples cool during rough grinding.

2. Metallography Laboratories

Routine Metallography Laboratories activities will be reported as part of the sponsoring research and development component's work; however, items of unusual interest or representing departures from routine operations will be reported here.

During the report month 450 samples were processed, a total of 472 macrographs and micrographs taken, 1946 negatives printed, and 7251 prints processed.

The metallographic procedure for polishing pure beryllium, although quite successful, has been both tedious and time consuming. An additional intermediate step added to the present polishing stages has reduced the required processing time by nearly a factor of two. This intermediate step immediately following the preliminary polishing stage consists of polishing the sample by hand for three to four minutes on a microcloth-covered lap with a slurry of 0.5 micron chromic oxide (metpolish #3) and 10% oxalic acid in water. Considerable care must be exercised during this step since excessive time in contact with the oxalic acid will pit the beryllium. Following this stage of polishing, the sample is returned to its normal finishing stage in the Syntron vibratory polisher. The finished sample also produces better grain contrast with polarized light than was possible before.

Uranium-zirconium alloys are normally difficult to etch for microstructure. Variations of composition and heat treatment cause wide fluctuations of the necessary polishing and etching conditions such that no one procedure can be said to work for all samples. A good method was found for some alloys containing about 2% zirconium. The procedure consists of electropolishing the sample in a solution of 10 g pyrophosphoric acid, 10 g chromium trioxide, 40 ml orthophosphoric acid, 100 ml sulphuric acid, and 100 ml distilled water at 10 volts for a few seconds to a minute, depending upon the previous surface finish. The sample is gently swabbed with cotton during electropolishing and the final tarnish must be removed by a very light mechanical polish on alumina. At this point the sample may be examined with polarized light. Further etching is accomplished in a solution of 10 g citric acid, 10 g concentrated nitric

acid, and 90 ml distilled water at 3 volts for 5 to 15 seconds.

A procedure for polishing low level irradiated samples with the Buehler Automet polishers has been established and found acceptable. Waste containment and disposal techniques have proven satisfactory.

A number of electron micrographs have been selected from the laboratory files on the basis of appearance and general interest. The micrographs have been mounted for display and are intended to serve such multiple functions as: provide the microscopist with easily accessible visual aid references while discussing work requests or results with customers, provide an information backlog to aid in the interpretation of current electron micrographs by comparison with standards, provide a source of readily available references for demonstrating laboratory capabilities to prospective customers, and provide visual aids to accompany talks to touring visitors.

3. High Temperature Lattice Test Reactor

Compatibility of Structural Materials with the HTLTR Environment.

Materials testing in support of the HTLTR continued. Fuel, cladding, control rod, thermocouple, and structural materials were exposed to a simulated reactor environment of nitrogen gas-graphite at 1200 C for 840 hours. The examination of specimens from this test is under way. Tests are being designed to obtain more detailed information on selected fuel and control rod materials.

HTLTR Small Mockup. The $\frac{1}{2}$ inch OD heating element run was terminated after operation for a total of 574 hours above 900 C. Operation was extremely stable with approximate temperature, voltage, and current ranges of $\pm 1\%$ of the set value.

Preliminary data from brick encased graphite electrodes indicate resistances of 50 K ohm, 135 K ohm, 620 K ohm and ∞ for electrodes that are $\frac{1}{2}$ ", 1" and $1\frac{1}{2}$ " and 2" from the heating element at a potential of 16.5 volts AC. The electrodes are in a temperature field that varies from 450 C at the farthest electrode to approximately 900 C at the electrode nearest the heating element. There appeared to be a battery effect occurring between the electrodes. Resistance values of 30 K ohm, 76 K ohm, 260 K ohm and 1 Meg ohm were obtained for the above electrodes in a temperature field which varied from 530 C to approximately 1150 C.

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HTLTR Safety and Control Rods. Fabrication of both assemblies continued during the month and is estimated to be 85% complete. Test stands for both assemblies were completed. The digitork motor and associated controls were received and temporarily connected to check motor operation.

TD nickel and boron nitride material has been ordered for evaluation of fabrication and compatibility problems. Some boron carbide in a graphite matrix material has been received from the physics group for similar evaluation.

Scope of both control elements have been completed.

4. EBWR Fuel Elements

EBWR Fuel Fabrication. The resonant plate vibrational compaction technique has proven highly successful in fabricating EBWR, UO_2 - PuO_2 fuel rods. An additional 250 fuel rods are to be fabricated by May 11, making a total of 800 elements available for approach-to-critical tests.

Ultrasonic inspections of the remainder of the EBWR Zr-2 cladding tubes were completed. Of 761 tubes, 473 (62%) were rejected, predominantly because of inner surface cracks and impressed metal particles. The rejected tubes are being returned for replacement.

5. NASA Fuel Development

NASA Cermets Work. Thermal cycling studies were conducted at high pressure on cermets containing 20 vol% UO_2 in tungsten, to determine behavior under conditions of particular interest in NASA reactor designs. Initial data indicate: (1) weight loss at 1000 psi hydrogen or 8% H_2 -argon is approximately 1/5 or less than that at one atmosphere of the same gas; (2) clad specimens show lower weight losses than nonclad material, even after 50 cycles; and (3) one atmosphere of hydrogen produces a higher loss rate than one atmosphere of 8% H_2 -argon; however, the purity of these gases has not yet been checked. Metallographic examination of the specimens shows an apparent fuel loss somewhat greater than indicated by weight changes; pull-out of UO_2 particles loosened by thermal treatment may be occurring during specimen polishing.

Experiments are in progress to determine the effect of selected additives to the tungsten and/or the UO_2 phase of the cermets. Additive compounds currently being evaluated include ThO_2 , Al_2O_3 ,

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TiO₂, and Nb₂O₅. First thermal cycling tests show no marked increase in fuel retention with any of these materials.

Cermet Fuel Fabrication. An approximately one-inch cube of 1/8-inch hexagonal honeycomb grid having 9/32-inch solid tungsten end blocks for electrical contacts was fabricated by pneumatic impaction. Fabrication techniques were basically the same as those used for fabricating regular hexagonal grids. Over-all quality and dimensions of the grid were good. However, slight distortion was evident due to the large volume of initial low density tungsten powder for the end blocks. The grid will be used in resistance heating tests at Lewis Research Center

A tungsten circular grid consisting of eight concentric rings connected by radial webs was fabricated by pneumatic impaction. An assembly of slotted mild steel cylinders was used as the expendable mold. The tungsten powder was loaded by vibratory compaction. After pneumatic impaction at 1200 C, the mild steel mold was removed by chemical techniques. A grid maintaining good dimensions was obtained. A similar shape has been fabricated using 80 vol% tungsten-UO₂ cermet.

Grid Integrity Studies. The investigation of structural integrity in pneumatically impacted grids involves several approaches. A small test grid utilizing molybdenum mandrels to more closely match thermal expansion coefficients was fabricated. This grid was crack free by visual examination. A large cermet grid that utilized alumina-coated mandrels was also impacted and was crack free by visual examination. The mandrels were alumina coated by first vapor depositing an aluminum mirror and then oxidizing this aluminum. This results in a very thin alumina coating which helps to prevent cracking of the cermet by preventing tungsten-to-steel bond.

Cermet Joining. Special aligning electrodes were designed and fabricated in preparation for cermet welding development to provide continuous clad at welded joints.

Submicron Dispersion of UO₂ in Tungsten. Tungsten-UO₂ dispersions have been successfully produced by coprecipitation from ammonium paratungstate and uranyl nitrate solutions, followed by calcination and hydrogen reduction. In the initial reductions some difficulty was noted with inclusions of unidentified foreign material within the compact. With increasing use of the reduction system further accumulation of impurities was noted and modification of the reduction system was required.

Just prior to modification of the reduction system, as-reduced powders formed a black material on the surface layer of the powder.

This system has now been modified to utilize all stainless steel lines between the hydrogen tank and the furnace, and the original hydrogen supply has been replaced by ultra pure hydrogen (10 ppm maximum impurities). Tests of the system made by reducing tungstic acid to the metal and ammonium paratungstate to the metal indicate that the accumulation of impurities has ceased.

Additional batches of tungsten-UO₂ containing 10, 20, and 30 vol% UO₂ have been prepared and calcined to the mixed oxides. These powders will be reduced with the modified system and their purity checked.

Assuming that adequate powder purity has been attained, compacts will be prepared by high energy compaction and their UO₂ retention characteristics determined by thermal cycling.

NASA Honeycomb Fuel. Coextrusion work to date in this program has utilized either wrought or as-sintered molybdenum and tungsten billet components. The wrought material in all cases has produced rough W-Mo interfaces, and the as-sintered material has produced interfaces which are relatively smooth. The roughness experienced in the former case is felt to be due to excessive grain growth during the billet preheat to 1800 C. This indicates the need for inhibiting grain growth either by use of as-sintered components or by alloy additions.

The direct extrusion of W and Mo powders shows promise as a means of solving the rough interface problem. The powder extrusion would also effect a large reduction in the cost of billet components. Work has been initiated to determine the feasibility of this approach. Three billets were made up containing 0.71-inch diameter, 2-inch long, as-isostatically-pressed, tungsten powder rods of 70% density, vacuum canned in 2.25 diameter cans. Two cans were of mild steel and the third was of 304 stainless steel. All three extrusions were made at 9:1 ratio through a ZrO₂ coated, 90° entrance angle, 0.750 diameter die. Preheats varied from 1250 C to 1100 C.

In all three extrusions the W compacts broke into several pieces and apparently underwent no reduction in area. The gross mis-match between the can and core extrusion constants, coupled with the extremely thick can wall, is responsible for the lack of success.

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Immediate future trials will utilize molybdenum cans and larger diameter tungsten compacts. Constraint thus applied by the die and by the stronger can will, hopefully, permit successful densification and extrusion of the powder.

ZrO₂ Coated Extrusion Liner. Previous extrusions of molybdenum and tungsten composite billets heated to 1800 C have shown a tendency to erode the inner surface of the extrusion liners on both the 2 $\frac{1}{4}$ and 3-inch containers. A thin H-21 liner with the ID coated with 0.035 inch of ZrO₂ by the Rokide process has been made up for use in the 3 $\frac{1}{2}$ inch extrusion container which can then be used for extrusion of 3-inch billets.

Success with the ZrO₂ coated dies makes the ZrO₂ coating appear feasible. The use of thin liners which can easily be removed would be desirable as existing extrusion containers could be used for refractory metal extrusion without damage. The thin coated liners can also be repaired at a small cost compared to the replacement of the entire extrusion container.

NASA Reference Plates. One hundred pounds of tungsten powder has been ordered from the same vendor and under the same specifications as that used by NASA. Additional sample lots of tungsten powder have been ordered from three other vendors.

Final specifications have been issued on the high fired UO₂ particles and a 10-pound order will be placed soon.

NASA drawings on the powder compacting die have been received and are currently being revised to meet the larger plate size and our existing equipment. A 400-ton press will be used to compact the tungsten-UO₂ mixtures.

Small stainless steel twin shell blenders, stainless steel screens, and miscellaneous laboratory hardware are being ordered.

Drawings for the hydrogen cooling chamber to be attached to the 8-inch mill have been completed. The drawings for the sintering furnace (NASA design) are being revised to accommodate the larger sized plates.

The hot work rolls for the 8-inch mill have been designed and a purchase requisition has been issued.

6. Other Customer Work

Thoria Development. Discharge of one ton of irradiated, vibrationally compacted thoria on April 1, 1964, marked successful completion of the first large scale thoria exposure at Hanford. Two additional tons were discharged on April 15, 1964, and replaced with the fourth ton, which will complete the "pilot" run of four tons. Half of the material was subjected to fringe exposure for 70 days and half of the material to core exposure of 15 days. Two reactor tubes of fringe loaded material will be carried to approximately 140 days of exposure before discharge.

Fuel fabrication development work continued to improve the process. An improved weld closure technique was developed which eliminated the need for cleaning the interior surface of the can prior to welding.

Irradiation Specimens for Phillips Petroleum. Irradiation samples consisting of gold-aluminum and lithium-aluminum alloys are being fabricated by a coextrusion process. Five gold alloys, containing up to 25% gold, were chemically analyzed. Six lithium alloys, all containing less than 1% lithium, were cast.

Boiling Burnout Studies for the ATR. Computer calculations were performed to determine temperature and heat flux distributions at the heat-transfer surfaces of the Advanced Test Reactor (ATR) plate-type fuel element. These calculations were made to assist in the design of an electrically heated model of the ATR fuel element on which the boiling burnout characteristics of the configuration will be experimentally determined. The ATR plate-type fuel elements are subject to a strong side-to-side variation in volumetric heat generation in addition to the longitudinal cosine distribution. Ratios of local to mean volumetric heat generation up to 1.7 can be expected in a two-inch wide fuel plate in the direction of the two-inch dimension. The principal material of construction of the element is aluminum, and, therefore, relatively low central temperatures were calculated. At heat fluxes of 1×10^6 , 2×10^6 , and 3×10^6 B/hr-ft², maximum fuel temperatures of 142, 281, and 423 F, respectively, were predicted. The thinness of the fuel plates (0.020-inch) limits the lateral redistribution of the heat by conduction, and large variations in heat flux exist side-to-side.

JJ Cadwell:kb

J. J. Cadwell
For Manager

Reactor and Fuels Laboratory

DECLASSIFIED

PHYSICS AND INSTRUMENTS LABORATORYMONTHLY REPORTAPRIL 1964FISSIONABLE MATERIALS - O2 PROGRAMREACTORN-Reactor Lattice Parameter and Spectral Measurements at Startup

N-lattice parameters to be determined include p , C_0 , ϵ , the neutron temperature, and the r -value of the epithermal flux. The spatial and energy dependence of the flux in the concentric tube fuel is to be determined using Pu-Al, U-235-Al, fuel enrichment uranium, depleted uranium, Lu, Eu, Au, and Cu pins, both bare and cadmium covered.

The hot test irradiation in the fully loaded N-pile was completed April 2 and 3, and the specially instrumented fuel elements were taken to the 300 Area for disassembly and pin counting. A simultaneous irradiation in the PCTR provided monitor pins of each type irradiated in a highly thermal spectrum, and these were counted along with the above pins to monitor decay and any possible counter drift. All data have been processed using computer program APDAC and checked for accuracy, and initial reduction is started.

Relative effective mass measurements of the cold test Lu, Eu, Pu-Al, and U-235-Al pins are complete, and the correction factors are being combined with counting data from the cold test using the APDAC I computer program. Measurement of the radial position of each pin within the fuel is under way using X-ray negatives acquired prior to the tests. These data will be used to obtain plots of activity vs. position which will be useful in inferring the N-lattice parameters.

Spline Worths in K-Reactor Mockup

The informal report, HW-81504, "Spline Worths in K-Reactor Mockup," has been prepared and will be distributed at the end of this month.

NPR Utilization Studies

Utilization studies of the NPR have continued during the past report period. Burnup calculations on one of the more promising configurations have been carried out employing the multi-region version of the burnup code ZODIAC.

Instrumentation

Engineering development continued on the low level neutron monitoring system planned for use during the N Reactor low-level operational testing program. The stability and reliability of the pulse discrimination module was improved through use of a tunnel diode discriminator circuit. Over-all resolving time of the pulse system was determined to be 0.7 μ sec, which is satisfactory for the intended use. Three modules were modified and installed. The diode pump modules were redesigned to: Improve linearity at both the lowest and highest counting rates, improve the zero adjustment function, and reduce diode failure rate. Two preamplifiers were designed, fabricated, and installed. Using the boron-lined detectors, the low-level monitoring system provided the required on-scale readings during reactor shutdown. In addition, performance tests were conducted at the 305 test reactor; these indicated that detector gas gain should be reduced and pulse amplifier gain increased to achieve reliable operation. Appropriate modification work was initiated.

Individual tests were satisfactorily completed on the 24 gamma energy spectrometers used in the slow scan portion of the N Reactor fuel rupture monitoring system.

A review was completed for Fuels Design, HL, on the entryway shielding for the gas loop, which is to be installed in the Advanced Test Reactor (ATR). Primary area of concern involved the possible necessity of conducting a mockup experiment to assure proper shielding. It was determined that an adequate shield design for the loop could be established based on existing data and appropriate calculational techniques without the use of a mockup.

Continued satisfactory data were obtained at the C Reactor storage basin regarding the use of gamma spectrometry methods to determine the cooling age of irradiated reactor fuel. The technique appears acceptable for routine application.

System Studies

The over-all N Reactor plant analog simulation, including both the primary and secondary coolant loops, was tested and debugged during the month. Minor problems were found in the computer program and these have been corrected. Progress is being made toward reducing equipment malfunctions and several significant system runs have been completed.

Initial attempts to "scram" the simulation indicated that the secondary emergency fill water system simulation was in error. The difficulty was traced to incorrect information on the pressure vs. flow characteristics

of the emergency fill pumping system. Corrections were made and the simulation now functions in accordance with calculated check points. Control Instruments and Systems Analysis Operation, NRD, was requested to assist in checking the validity of the simulation. The checking procedure is now considered to be completed for the 100% power case.

Results of scram runs indicate that gain and reset settings on the steam generator and surge tank level controllers are relatively critical. It is also becoming apparent that some modification of the secondary system emergency fill and emergency spill control logic following a scram is advisable. Further study in this general area is required before definite recommendations can be made.

Calculation of simulation parameters for operation of N Reactor at reduced power levels is in progress. Conditions corresponding to the N-2 testing program are being calculated and entered into the simulation.

The N Reactor controller simulator is performing satisfactorily. Nine proportional-plus-reset and three proportional-plus-rate controllers, with their associated error (setpoint) amplifiers, are being simulated with this device. In addition, the dynamics of seven control valves have been simulated. Additional controller and control valve plug-in units are provided for expansion of the over-all N Reactor simulation. Six of the extra plug-in units are still to be tested but no troubles are expected. Use of the control system simulator instead of Bailey controllers permits operation of the simulation at ten and one hundred times real time.

Reactor Design Analysis Operation, NRD, requested assistance in determining a method of adding the rate control mode to the N Reactor pressurizer level controller. The need for this addition was pointed out in earlier simulation studies. A study of the existing equipment indicates that an addition can be made to the existing controller panel mounting rack which will provide the required control while using the standard plug-in controller. The method was tested in the laboratory and found to operate as predicted.

A visit to N Reactor was made to determine cause of difficulties experienced in controlling the primary coolant flow. The trouble was traced to loss of setpoint signal to two of the loop flow controllers.

A memorandum describing adjustments and modifications made last month to the primary injection and coolant control systems was issued.

The second transport lag simulator for the N Reactor simulation has been completed. The memory drivers and sense amplifiers were completed and

installed in the second unit during the month. The complete transport delay system was then checked out and made ready for use. The two simulators are being used to study primary coolant temperature differential between primary coolant and heat exchanger.

The equations and constants received from N Reactor Department for simulating cold water injection were reviewed during the month and used to make a number of detailed test runs. Results are being analyzed.

SEPARATIONS

Critical Experiments with Plutonium Solutions

The critical assembly vessel being used in the current solution experiments is an 11.5-in. diameter stainless steel sphere. Plutonium nitrate solutions under study are in the concentration range of 200-400 g Pu/l. The critical assembly vessel can be reflected with water, or with concrete nesting shells in thicknesses of 5.25 and 11.25-in. Critical data are presented below for two cases in which criticality was achieved with the vessel just full (12.95 liters) when reflected with an effectively infinite water reflector as opposed to the case when surrounded with an 11.25-in. thick concrete reflector.

(Measured Sphere Volume: 12.95 l; Wall Thickness: 0.049-in)

<u>Reflector</u>	<u>Pu Conc. (g/l)</u>	<u>Acid Molarity</u>	<u>H₂O (g/l)</u>	<u>NO₃ (g/l)</u>	<u>H/Pu Atomic Ratio</u>	<u>Critical Volume (liters)</u>	<u>Critical* Mass (Kg Pu)</u>
Water	269	1.06	840	346	83.3	12.95±0.004	3.48
Concrete 11.25" Thick	236	1.16	853	318	96.5	12.94±0.003	3.05

* Pu²⁴⁰ content is 4.6 w/o.

The difference in critical concentrations shows that the 11.25-in. concrete reflector is slightly less effective than the water reflector although the two are nearly equivalent. Previous experiments with more dilute solutions (30-40 g Pu/l) in a 14-in. sphere reflected with concrete had shown the reverse to be true, in which case the concrete was found to be a slightly better reflector than water. This verifies a slight dependence of the concrete reflector worth on the spectrum within the reactor core, as expected.

Bare and cadmium-covered gold foil irradiations were made for determining the neutron flux distribution across the sphere diameter extending into the reflector. Analyses of the data provide values for the effective extrapolation length (λ_{eff}) and the critical or material buckling of the Pu solution. The results for the water reflected sphere are briefly described below. The analysis of the foil irradiations made with the concrete reflector has not been completed.

A symmetrical flux distribution was measured across the sphere diameter. The relative thermal flux (difference between bare and cadmium-covered gold foil activity) was found to have a maximum value about 2.5 cm from the surface of the vessel; at this point the thermal flux peak in the water reflector was $\sim$ eight times the value in the Pu solution at the sphere center. The measured cadmium ratio, using five mil gold foil with a 20 mil thick cadmium shield, was $\sim$ 1.15 at the center of the sphere which indicates a relatively fast spectrum.

A least squares analysis of the gold foil activity to a sine function, $A \frac{\sin Br}{r}$, in the region of constant cadmium ratio yielded a value for the extrapolation length, $\lambda_{\text{eff}} = 6.08 \pm 0.32$ cm. The critical or material buckling for the Pu solution is then, $B^2 = \frac{\pi^2}{(R_{\text{critical}} + \lambda_{\text{eff}})^2} = 23,140 \pm 720 \times 10^{-6} \text{ cm}^{-2}$. During the foil exposures the critical Pu concentration was ~ 254 g Pu/l with an H/Pu atomic ratio of 44.6. Total nitrate (NO_3) concentration of the solution was ~ 340 g/l.

Further critical experiments will be performed with the 11.5-in. sphere fully reflected with concrete to determine the effect of nitrate on critical concentration within the vessel.

Pulsed Neutron Source Experiments

Pulsed neutron source experiments and reactor noise measurements were made concurrently with the critical experiments in the 11.5-in. water-reflected sphere for determining neutron lifetimes and k_{eff} . Values for the prompt neutron lifetime varied between 6 to 15 $\times 10^{-6}$ sec depending on the concentration of Pu in the solution. The following typical data were obtained.

Pu Conc. (g/l)	H/Pu (Atomic Ratio)	Delayed Critical Volume	Prompt Critical Volume	β/l (sec ⁻¹)	l (μ sec)
269	83	12.94 $\pm$.004	13.11 $\pm$.015	176 $\pm$ 4	11.9 $\pm$ 0.3

The solution worth at or near delayed criticality is about \$6 per liter, with the difference between delayed and prompt criticality being about 0.16 liter.

The analysis of the reactor noise measurements which were made has not been completed.

HISMC - Monte Carlo Code for Homogeneous Infinite Systems

The Monte Carlo Code, HISMC, has now been placed on the SPL library tape at the 7090 computer. This code can be obtained by means of a "get" card with title HISMC. The output available from the code for an infinite system includes: The infinite multiplication factor (k_{∞}), the second and fourth moments of the slowing down distribution, age, migration area, mean slowing down time of the neutrons, mean lifetime of the neutrons, the neutron flux as a function of energy, and the number of fissions and absorptions that occur for each isotope as a function of energy.

Miscellaneous Calculations for Nuclear Safety Applications

A series of Monte Carlo and multigroup diffusion calculations were made to evaluate the nuclear safety of the proposed design of the HAPQ-V shipping cask for slightly enriched irradiated fuel elements. The cask contains four fuel regions positioned in a square array in lead with 21-in. between centers. The thickness of lead between surfaces of the fuel regions is about 12-in. The diameter of each fuel canister is about 9-in. A Monte Carlo method was used to calculate an effective albedo (ratio of number of neutrons returning to a canister/number of neutrons leaving) for each canister, taking into consideration the neutron interaction between canisters in the lead. The HFN diffusion code was then used to compute k_{eff} for an individual canister (and of the array), making use of the calculated effective albedos from the Monte Carlo calculations.

For fuel enrichments in the range 0.98-1.5%, the computed k_{eff} of the shipping cask ranged between 0.84-0.96. With uranium enrichments below about 1.18 w/o, the value of k_{eff} will be $\leq \sim 0.9$, which is the maximum value permitted of a container for use in shipment of irradiated fuels.

Critical Mass Laboratory Instrumentation

The instrumentation used in conducting critical experiments at the Critical Mass Laboratory was made ready for the resumption of solution experiments in Hood #1. New cables for the detection chambers were installed in a recently added cable tray. Each channel of the safety circuit and data

acquisition system was checked and recalibrated as necessary. The liquid level manometer system was also put back in operation. The instrumentation performed properly during the conduct of a number of critical experiments.

Additional reactor noise analysis data were recorded with a different and improved detection recording system. The same thermal neutron sensitive detector was used with the addition of a linear pulse amplifier, linear count-rate meter, and summing amplifier. The summing amplifier removes the d-c component of the signal from the count-rate meter and amplifies the fluctuating component before it is recorded.

Consulting Services on Nuclear Safety--Criticality Hazards

1. Nuclear Safety in HL

Temporary Specification No. 4 was issued by the Metallurgy Development Operation. This specification covers the non-routine processing of 50-gram plutonium metallographic samples in the 231-Z Building.

2. Nuclear Safety in CPD

The annual audit and nuclear safety review of CPD operations was completed during the month. A report detailing comments and recommendations of the review committee is now being prepared for transmittal to CPD.

Participation on the Americium-241 Recovery Operation Hazards Review Committee continued. The recovery facility is located in the 234-5 Building. The committee met four times during the month. A hazards summary report is now under preparation.

3. Nuclear Safety in NRD

Twelve changes were made in the nuclear safety specifications for handling slightly enriched uranium in the 333 Building. These changes were reviewed and approved for the N-Reactor Fuels Operation. Most of the changes were of a relatively minor nature.

4. Nuclear Safety Training and Education

The Course B, Group III lecture series in nuclear safety for CPD was completed in April. The course consisted of nine lectures oriented toward the non-technical supervisors and specialists who have need for further understanding of the nuclear safety problem and of criti-

cality control practices.

On April 1, a three-hour lecture on Nuclear Safety was presented to 26 exempt personnel enrolled in the Radiation Protection Operation Course at HAPO.

5. Nuclear Safety in Shipment of Fissionable Materials

An article entitled "The Transporting of Radioactive Materials" is being prepared by L. L. Zahn (GE-Hanford), J. Langhaar (duPont-SR), and C. L. Brown (GE-Hanford). This article will describe significant progress in the field of transporting radioisotopes, irradiated fuels, and unirradiated fissile materials. The three authors met at Hanford on April 14, 15, 16, and 17, at which time a rough draft report was completed. The article is to be included in a forthcoming AEC publication entitled Selected Reviews in Reactor Technology-1964; Dr. Leonard Link, Assistant Director, Reactor Engineering Division at Argonne National Laboratory is the editor-in-chief.

Separations Instrumentation

General laboratory testing was continued on the solid state portable gamma spectrometer for plutonium monitoring in hoods and glove boxes. In addition, a 1 mm thick NaI scintillation crystal was received for detector use in the Pu-239 liquid sample counting system, and it is being applied to a multiplier phototube.

METALLURGY - Nondestructive Testing

N Fuel Testing

1. Uranium Billet Testing

Improved lubricants introduced in the extrusion process by Fuels Engineering have had a significant effect on the clad thickness variations experienced on the inner clad of the outer fuel. This has allowed more time to improve the ultrasonic billet test for the next production test. Work is being done on a probe to increase the depth of response and a summing circuit to improve the readout. After these developments are complete a production test will be run to evaluate the correlation with clad variation in order to increase fuel element quality and reduce zirconium thickness.

Use of eddy current techniques to measure billet inner bore characteristics was also investigated and a memorandum written. It was deter-

mined that the effort was not promising and was dropped.

2. Component Testing

Inspection of copper billet container welds by eddy current techniques is under study. Cracks in the weld area are possible and can cause extrusion problems. Conventional eddy current methods gave poor results due to probe wobble and copper thickness variations. A through transmission technique is being investigated.

3. Fuel Element End Closure Testing

Development of a cap to core bonding test for the "Chevron" end cap has proceeded well. This closure consists of a wedge angled cap which makes the regular testing method ineffective. By use of a lower frequency beam directed at an angle perpendicular to the cap end surface, a lack of signal indicates an unbonded region. If efforts to detect the inside bond region can also succeed the test will be directly applicable in the Fuels Engineering test station with little modification in the present system. An inside probe has been successfully demonstrated but a means of testing from the outside also is being investigated.

4. Uranium Enrichment Testing

Tests on the ERT-1 enrichment tester in the co-extrusion shop have been performed on both "green" and older uranium billets. Initial results have shown that the difference between 0.94% and 1.25% enrichments of U_{235} can be determined reliably on both extremes of aged uranium. Routine testing of billets is planned by Quality Control to determine tester stability and maintainability before it is accepted as a production tests. However, on the basis of results to date, HTR reactivity tests will be discontinued and replaced by the enrichment tester.

5. X-ray Fluorescence Test Development

The new X-ray equipment and associated mechanical and electronic systems were installed in the production area. Because of problems in stability and anomalous response variations, further improvements were made and a production test for evaluation and correlation was decided upon before the test was accepted for use. Precise line voltage regulation was determined to be necessary to increase sensitivity of response over background variations. Also the problem of fuel element dimensional variations was reduced considerably by redesign and restriction of the shielding apertures. Preliminary tests on about 50 element ends

have shown a considerable portion containing regions of over 1.0 weight percent uranium contamination. These are being checked by autoradiograph and alpha counting for correlation.

A production test is being designed with Quality Control to completely evaluate the tester and its results. This will determine the sensitivity level, relate it to the contamination after brazing and also determine the throughput and capability of the equipment. Correlation will be made with alpha counting, autoradiograph and neutron fission tracks.

6. Uranium Surface Contamination Testing

Neutron fission tracking tests have shown that autoradiograph and alpha counting techniques result in higher readings. In all samples checked the contamination covered a relatively large area. This neutron track technique is being used to check standards and correlate with X-ray fluorescence. A simpler neutron source is being investigated.

7. Irradiated Fuel Testing

Modifications to the underwater mechanical equipment have been made and arrangements completed to begin testing and data accumulation on a selection of irradiated N fuels. Prototypical circuits for the improved UT-10 B model have been started using semiconductor circuitry. As-builts on the unit were completed and mechanical design for the improved model is well under way. The instruction manual is about 80 percent complete.

Zirconium Hydride Detection

Testing Methods now has responsibility for developing both eddy current and ultrasonic hydride detection probes for N Reactor process tubes. It is planned to utilize the advantages of each test to assure the utmost reliability of test results.

Ultrasonic tests on six hydrided cylindrical Zircaloy samples from 20 to 7000 ppm hydrogen were made to determine longitudinal and shear velocity effects. An essentially linear increase for the longitudinal velocity, up to 8 percent change at 7000 ppm, compares to about 1 percent change for the shear velocity. This verifies results found on the process tube samples using the combined or complete wave form test technique. Destructive examination and metallography have established that the samples are uniform in hydriding and quite accurate as to hydride concentrations. An invention

disclosure is being prepared on the hydride detection methods. Three additional burst test hydride samples were tested and mapped. Again these were deeply grooved samples causing signal loss in these regions but the hydrided regions were also detected.

Fabrication of the prototype ultrasonic rotating probe has started and is about 30 percent complete. This probe will use a water cell concept with additional water added continually to allow for leakage. The transducer will rotate 360 degrees per 1/2 inch of traverse and coupling is through a wideband rotary transformer.

Al-Si Fuels Testing

1. Irradiated Fuel Testing

Reproducibility of ultrasonic bond test equipment has been tested using three different transducers with unsatisfactory results. Lack of uniformity between test results is related to transducer differences and to test sample distance spacing differences. Laboratory transducer measurements will be performed and test runs will be made to determine optimum water spacing. The test results show that due to surface noise from imperfections caused by corrosion, dents, etc., small voids are not detectable. Signals believed to be from core cracking are also being investigated. Using a high frequency shear wave test the cracks could not be detected so a lower frequency will be investigated.

2. Bond Test Development

Improvements on the outside-in bond test made to increase its reproducibility included a gate follower circuit which is designed to reliably gate defect indications regardless of core size or clad thickness variations. Reproducibility tests have shown a drift in the system believed to be in the amplifier. Circuit development for the new production prototype has begun; timing and delay circuits are being developed and evaluated.

3. Hot Die Sized Fuel Testing

An investigation of an externally applied nickel thickness tester for finished fuels has resulted in ordering an eddy current instrument from a vendor. This instrument has good possibilities for detecting magnetic nickel under the aluminum can wall and it will be thoroughly evaluated.

The UE-3 tester manual is now being typed and as-built prints have been completed.

NEUTRON CROSS SECTION PROGRAM

Measurements have been continued to determine the source of the remaining internal discrepancies in the determination of the scattering law for H_2O at $95^\circ C$. Under some conditions about 30% of the scattered neutrons appear to have a different energy from the nominal one; however, the origin of these neutrons has not been determined. The presence of a subtle difficulty is further established by a discrepancy between a recent check run and earlier published Hanford results on H_2O at room temperature.

Time-of-Flight Spectroscopy for Slow Neutrons

Development of components for the measurement of slow-neutron inelastic scattering by time-of-flight has continued. Phase-jitter measurements and static tensile tests have been made on rotors. A thin slab of a copper single crystal has been plastically deformed in an attempt to increase its integrated reflectivity. A liquid sample holder for a neutron beam of large cross-sectional area is being developed.

Fast-Neutron Cross Sections

Major improvements in the data reduction program for total cross section data have been debugged, including a correction of the energy scale for non-uniform channel widths. When this correction is applied to data for Be and Si, discordant results are brought into satisfactory agreement. Another computing program for comparing and combining cross-section results is almost finished.

Samples of separated Ca^{42} and Ca^{44} have been prepared. The 400-channel analyzer is being overhauled in preparation for an experimental run in May.

REACTOR DEVELOPMENT - 04 PROGRAMPLUTONIUM RECYCLE PROGRAMLow Exposure PuO_2 - UO_2 Lattice Studies

Ceramics Research and Development has completed the fabrication of the fuel for these experiments. One hundred and sixty-four rods, 36 inches long, were vibrationally compacted. The remaining 70 rods were loaded with pellets. The vibrational compaction was adjusted to give the same grams of fuel per unit length as the pellets. Further care was exercised to insure that the pellets near the flux monitoring positions would contain the average grams of fuel/unit length.

Reactor Physics of D₂O Moderated Systems

A document, HW-SA-3546, "Survey of Work at Hanford Laboratories in D₂O Moderated Systems," has been approved and issued. Parts of this survey will appear in "Lattice Studies and Critical Experiments in D₂O Moderated Systems" which has been prepared for the 1964 Geneva Conference.

Analysis of Nickel for Foil Detectors

Various nickel samples have been irradiated in a fast flux in the PCTR for 2000 watt minutes. These foils were then analyzed with the 256 channel analyzer, and it was found that the foils as supplied by Reactor Experiments, Inc., were by far the purest. With these foils, the 0.37, 1.11, and 1.49 MeV gamma spectrum from the β^- decay of $^{28}\text{Ni}^{65}$ produced by thermal neutron capture in $^{28}\text{Ni}^{64}$ was easily detected. This decay has a 2.56 hour half life. After one day the 0.81 MeV gamma peak from the $^{27}\text{Co}^{58}$ produced by a fast (n,p) reaction in $^{28}\text{Ni}^{58}$ showed up quite well. The one day waiting period allowed $^{25}\text{Mn}^{56}$, produced by thermal neutron capture in $^{25}\text{Mn}^{55}$ (present as an impurity), to decay with the emission of a 0.84 MeV gamma; and the upper isomer of Co^{58} to decay to the lower, which in turn gives the 0.81 MeV gamma.

It was concluded that these nickel foils could be used as both a fast and thermal neutron flux detector. One additional nickel sample will soon be analyzed, and at this time the flux spectrum present in the earlier tests will be determined and, consequently, the relative amounts of Ni and Mn present in the samples will be calculated to establish the actual purity needed for their use as detectors.

Code Development

1. Transmutation Logic for S_n Transport Analysis

A paper with the above title has been submitted for journal publication. The abstract of the paper follows:

"The machine analysis of high-exposure spectral-shift breeder reactors has been extended to span bilinearly coupled multi-energy neutron transport theory and nuclide transmutation theory. Scaled-matrix power series evaluation of the matrix exponential fine structure of the transmutation fields proves to be far superior to both chain-tracing and matrix-diagonalization methods, in range, reliability, speed, and simplicity of logic relevant to the occurrence of complex eigenvalues and coalescent eigenvalues."

2. RBU

Logic flow diagrams of the Monte Carlo portion of RBU have been submitted to Graphics Operation for inclusion in HW-80002, the RBU system document. Approximately 15% of the document remains to be completed. Review of RBU - Vol. III, HW-80003 (the operating manual) is complete, and the draft has been returned for some alterations.

Modifications were made to the code PLOTCH to allow plotting of the RBU cumulative cross section tape on the Benson-Lehner plotter. This allows closer checking of RBU problem input and cross section data.

3. Hanford (RBU) Basic Library

Review, evaluation, and updating of cross section data contained in the RBU Basic Library is continuing. The reviewed and updated data will comprise the new Hanford Basic Library. Sixty-nine additional isotopes have been reviewed and their records updated. Plots of the cross section data contained in the RBU Basic Library have been made for an additional 97 isotopes. The Hanford Basic Library now contains updated cross section data for 116 isotopes.

4. Cross Sections

Cross sections for deuterium have been added to the TEMPEST Library. The data for Pu-239 as suggested by Sher has been added to both the TEMPEST and SPECTRUM Libraries. Some minor smoothing of the data originally on these libraries (B. R. Leonard's data) has been performed.

5. High Energy Inelastic Scattering

Excitation level cross sections for 33 isotopes have been obtained using the statistical model code. Ratios of flux weighted partial cross sections are being calculated for these isotopes to generate the excitation level probabilities. Excitation level energy losses and their corresponding probabilities will then be punched directly on cards which will replace the current RBU inelastic spectrum tables in the Hanford Basic Library.

6. BARNS-II

Debugging of many of the recent revisions to BARNS-II has begun. Other revisions remain to be coded. The revisions being debugged will calculate point values of the cross sections and their use will materially aid the work of reviewing and updating the cross section data for the

Hanford Basic Library. The revisions to be coded are concerned with calculations involving the resonance parameter tables of the library.

7. ZODIAC CHAIN

A current ZODIAC User's manual (HW-81302) was published, replacing the obsolete manual of 02-14-64. A ZODIAC Chain Tape incorporating all currently completed improvements was released for production use at the same time. Since that time, a correction has been made permitting treatment of GAM resonance cards. The next scheduled improvement in the ZODIAC Chain is the replacement of CLERKB and NORMAL MODES by an improved, flexible, 20-isotope CLERKB and ALCHEMY.

A program which will thoroughly check the accuracy of new ZODIAC REFIRE burnable isotope tables is in debug.

8. TRIP005

A revision to TRIP005 makes possible the inclusion of more than one ϕ vs. τ table used in computation of non-standard equation 4.6. Two tables are now included in the deck and more will be added. The code will now select a table as specified by input option.

9. KERNEL

KERNEL symbolic and binary decks, together with the necessary documents and explanatory material, were prepared for offsite distribution.

RBU Application

The plutonium aqueous homogeneous spherical critical experiment 12-1 was run using the RBU Monte Carlo. B. R. Leonard's values for the Pu-239 cross sections were used. The Monte Carlo was run until the fluxes were well converged. The RBU diffusion code gave a value of 1.011 for the multiplication constant, k_{eff} . Analysis of the results is continuing.

Burnup Experiments

With the exception of the single outer rod from element 5095, which is to serve as a check on previous data, all of the L_x Pu-Al burnup data have been obtained. They are now being compiled and scrutinized for future publication. The preliminary data obtained from the check rod of element 5095 indicate that the previous burnup data were correct, which means the flow- ΔT data for this element were probably in error. The L_x Pu-Al data when treated by the method used in HW-80067 RD produces a reactor effective

value of $\sigma_{239}^{239} = .44 \pm .026$. It cannot yet be concluded what input values of the 2200 meter cross sections are necessary in ALTHAEA to correctly predict the experimental plutonium atom density variation. Since ALTHAEA necessarily requires other input parameters in addition to the 2200 meter cross sections, the "correct" cross sections will be dependent on these input parameters. The resonance self-shielding parameter for the Pu-239 is particularly difficult to obtain, and the value used must to some degree affect the choice of input cross sections which will provide the correct isotopic variations. Attempts are now being made to provide physically sound methods for obtaining the input parameters so that some conclusion may be drawn concerning the validity of the 2200 meter cross section values.

Isotopic Analysis of PRTR Samples

Isotopic analyses were provided on 38 plutonium samples of PRTR-irradiated fuel elements in support of the Plutonium Recycle Program. Of these 5 were macrodrill samples from UO₂ element No. 1101, 6 were macrodrill samples from UO₂ element No. 1041, 7 were burnup samples from Al-Ni-Pu element No. 5095, 17 were burnup samples from Al-Ni-Pu element No. 5103, and 3 were burnup samples from Al-Ni-Pu element No. 5109.

Isotopic analyses were completed and reported on the following elements: Al-Ni-Pu - No. 5103 - Burnup Samples (HW-81985) and Al-Ni-Pu - No. 5095 - Burnup Samples (HW-82057).

Fast Fuel Test Reactor

The results of the nuclear design calculations for a fast fuel test reactor have been summarized in an informal document, HW-81940, "Nuclear Design Calculations for a Fast Fuel Test Reactor," by J. J. Regimbal and R. E. Peterson dated 4/21/64.

Phoenix Fuel Program

1. Methods Development

Various methods for treating the Pu-240 resonance absorption in high Pu density cores are being compared.

In the usual multi-group calculations for Phoenix cores, the 1 eV Pu-240 resonance falls into "Group #3" which has lethargy width $\Delta u = 1.25$. This group accommodates five quarter lethargy GAM sub-groups. As a result, the Pu-240 group cross section does not equal the infinite dilution value, but represents a crude approximation of the self-shielded value in a homogeneous ("thin" plate Mark I) core.

For low Pu-240 concentrations, this approximation may be adequate, and, in fact, for cores used in early calculations. For higher Pu-240 concentrations, however, the 0.25 Δu sub-group is too coarse, and the calculated σ_{a3} -240 is too high.

Since the 240 resonance absorption is primarily due to the isolated 1.054 eV resonance, the single level Breit-Wigner formula can be used to describe it. In a more recent method for treating Pu-240 absorption, a self-shielding factor, f_{240} , is calculated using the NRLA approximation. This factor then multiplies the GAM generated σ_{a3} -240. In the burnup calculations f_{240} is fitted numerically as a function of N^{240} .

Another possible procedure for calculating f_{240} makes use of the Nelkin water kernel and the SPECTRUM code. The resulting f_{240} can then again be fitted as before for burnup calculations. Schemes of this type are presently under consideration. The present energy mesh in use gives rather large truncation errors and will have to be revised.

2. Plutonium-Burnable Poison Systems

A series of burnup calculations for Pu-B¹⁰ and Pu-Sm systems have been completed. Sm appears to be a potentially interesting burnable poison for Pu cores. Since it results in a rising reactivity characteristic in certain Pu cores, one might consider its use in conjunction with B-10 for reactivity-time flattening.

3. Medium Enrichment Reactors

k_{∞} 's for several isotopic compositions of Pu have been computed for cores containing 10 v/o of a mixture of PuO₂ and UO₂, 40 v/o Zircaloy, and 50 v/o H₂O. The PuO₂ content of the fuel was varied from 0% to 100%.

Initial results indicate that for a given k_{∞} the conversion factors are quite comparable for each of the Pu systems. Burnup calculations are presently being carried out.

4. Pu-Be Systems

A series of PHYSICS CHAIN code calculations were performed on infinite homogeneous systems consisting of Be and Pu. In these calculations a better treatment of the n-2n effect was used than in studies reported previously.

The Pu compositions varied from 95/5, Pu-239/Pu-240 to our standard "reactor grade" Pu which has 50/31/17/2 ratios for 239/240/241/242. The Be/Pu ratio was varied from about 100 to about 5000. The parameter of primary interest is the transmutation ratio for Pu-239 to Pu-241. It was found that this ratio can exceed unity for Be/Pu ratios of about 1000 for the 50/31/17/2 composition. A burnup calculation will be carried out to confirm the expected favorable reactivity lifetime characteristics of such a system.

5. Calculations for the Proposed PCTR-Phoenix Experiments

The homogenized core calculations reported in the February monthly report have been extended to heterogeneous geometries. The thermal flux depression factors were calculated using a double P3 method and the Pu-240 was shielded with the appropriate NRLA approximation. The k_{∞} values for the CH₂ moderated Pu-Al-fueled systems are listed in the following table:

$$V_{\text{fuel}}/V_{\text{mod}} = 0.25$$

<u>Plate Thickness</u>	<u>Pu-239/Pu-240</u>		
	<u>95/5</u>	<u>80/20</u>	<u>70/30</u>
0	1.7372	1.5481	1.4440
20 mil	1.7480	1.5820	1.4845
60 mil	1.7555	1.6096	1.5182
120 mil	1.7575	1.6248	1.5374

$$V_{\text{fuel}}/V_{\text{mod}} = 0.50$$

<u>Plate Thickness</u>	<u>Pu-239/Pu-240</u>		
	<u>95/5</u>	<u>80/20</u>	<u>70/30</u>
0	1.6957	1.4940	1.3896
20 mil	1.7077	1.5201	1.4191
60 mil	1.7221	1.5515	1.4551
120 mil	1.7337	1.5776	1.4852

Instrumentation and System Studies

Laboratory tests were conducted to determine gain shift characteristics on six scintillation detector probes for use in the PRTR fuel element rupture monitor. Both short and long term gain shifts were observed. Further testing is being carried out on two more phototubes to determine if stability can be achieved.

It was determined that Cs-138, I-131 and Rb-88 are the three radionuclides of particular concern for detection and measurement by the PRTR exhaust monitoring system. Required detection limits are 5×10^{-6} $\mu\text{Ci/cc}$. Considerations are being given to possible methods of achieving the required sensitivity. It appears that both a constant flow gas stream monitor and a filter paper, or other, buildup monitor may be required.

Satisfactory results were achieved with the relay circuit fault monitor, which was developed to provide an indication of intermittent open circuits in the PRTR shutdown control system. The fault monitor accurately determined the cause of the persistent difficulties, thereby eliminating the need of replacing considerable conduit-enclosed wiring. The developed fault monitor is directly applicable to other relay circuitry problems and an invention report was submitted describing the instrument.

Preparations are being made for a PRTR test scheduled for early in August in which the automatic controller will be tested in the period mode of operation. Since the analog simulation must cover many decades of output power, one of the large general purpose analog computers will be used. Telemetry of analog data over phone lines is contemplated to complete the loop consisting of the gas balance system and automatic controller (located at the reactor) and the simulated reactor (located at the analog laboratory, 3707-C Building).

Additional runs for a postulated prompt excursion of a mixed U plus Pu core were requested. Using previously furnished equations and parameters, additional analog simulations were made.

EBWR EXPERIMENT PROGRAMPRCF Startup Experiments with EBWR Fuel

The PRCF went critical (3-27-64) loaded with 495 fuel rods and three empty fuel channels with an excess of 0.313 mk. The extrapolation back-to-critical gives a value of 493.9 (5,388 gm Pu) fuel rods for the critical mass. Fuel rods loaded into the three empty fuel channels were worth 0.096 mk each. The critical mass, corrected for the fuel loaded in the

three empty fuel channels (counter positions for approach-to-critical), is 495.9 (5,410 gm Pu) fuel rods.

The rods and sheets were calibrated by subcritical multiplication to obtain the shape of the worth vs. position curves. The total worth of the rods individually and combined were measured by the rod drop method. The average value of the three rods is 13.1 mk and that of the three sheets is 4.6 mk. The combined worth is 1.12 times the sum of the worth of the individual rods and sheets. The end of one rod was calibrated by positive periods in small steps until 1.5 mk of the rod had been calibrated.

A recording of the reactor noise has been made using a tape recorder. The analysis of this data to obtain β/λ has not been completed.

Moderator worth and reactivity coefficient measurements were made for moderator levels between 106 inches and 91 inches. Copper wire is being irradiated in the central fuel channel to determine the vertical flux distribution at several moderator levels.

A report, "Experiments with $\text{PuO}_2\text{-UO}_2$ Fuel Elements in Light Water," has been prepared for inclusion in the Physics Research Quarterly Report, January-March, 1964.

Void Temperature Effects Separation

A theoretical-analytical study of the information derivable from foil activations in the EBWR core is under way. The objective of the study is to demonstrate that the experimental determinations of the "effective neutron temperature" and the void fraction are feasible. The neutron temperature will be measured by lutetium foil activations. A comparison of an analytical calibration curve for lutetium foils has been made with an experimental curve, the results comparing favorably. The neutron temperature is a function of both the moderator temperature and the void fraction, thus the effects of each on the neutron temperature needs to be separated. The effects of voids on a resonance absorber (gold foils) activation are being studied to determine if the activation sensitivity, as a function of void fraction and temperature, is large enough to be measurable.

HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

A graphite heating element of 1/2 inch diameter and 40 inches in length has been run at 16.5 volts and 237 amps for a period of 600 hours at about 1400°C with no indication of any breakdown. The collet type end connections of the heater are similar in design to those of heaters now being designed for the HTLWR. The small diameter of the heated region permitted running

the test at voltages more nearly like that expected to be used during HTLTR operation. During the run graphite electrodes were placed in the brick surrounding the heating element at distances of 1/2, 1, 1-1/2, and 2" from the heater surface. At the nearest and farthest locations the temperatures were 900°C and 450°C, respectively. Measured resistances between electrode and heater were 50 K ohm, 135 K ohm, 620 K ohm, and greater than 2 M ohm, respectively.

The fifth in a series of high temperature materials tests has been completed after 840 hours of running at 1200°C in a nitrogen atmosphere. The test contained samples of BN, B₄C, AlN, and UO₂ in capsules of Ni, thorium dispersed Ni, and graphite. During the operating period the test was shut down twice, once after 75 hours and again after 400 hours. In each case the Ni effluent line was replaced because of corrosion. Spectrographic analysis of material at the corroded point indicated the presence of B, Al, and Na. Initial examination at the end of the run indicates the BN samples to have been unstable and probably to have contributed to the corrosion effects. A further test will be made in which the compatibility of B₄C alone with Ni, T.D. Ni, and graphite will be studied.

In order to evaluate the response of the HTLTR to various reactivity transients, a series of cases were run using the TRIP computer code as a reactor simulator.

This study resulted in several interesting conclusions.

With 5% enriched fuel, the Doppler coefficient is sufficient to shut the reactor down from a prompt critical condition. Using a lower enrichment fuel, which increases the Doppler coefficient, the final temperature reached by the fuel is lowered for a given reactivity addition. Provided the high level trip operates, there is no significant heating of the fuel even though the reactor goes prompt critical.

Contributions were prepared for the Preliminary Safeguards Analysis document and numerous comment blueprints were reviewed.

NEUTRON FLUX MONITORS

Chemical separation and mass spectrometer analyses were completed on all six plutonium regenerating detecting elements and on all six uranium flux monitor elements. The obtained data are being compiled and analyzed for use in the final report. Agreement between the data and the calculated results is good; however, the accuracy of any correlations appears to be limited by the absolute values of the cross sections. In addition, the cross section information appears to limit the accuracy of integrated

exposure evaluations. The unclassified report discussing the use of U-236 to U-235 ratios to measure exposure is being rewritten to include cross section uncertainty. A summary discussion of the regenerating detector work to date was prepared for use at the forthcoming Geneva conference.

Information was received that the U-234, which will be used as coatings in the two uranium regenerating detector chambers, is being shipped to HAP0 from Mound Laboratory. In addition, the purchase order for the two chambers, which will be fabricated by GE-APED, was approved by RL00-AEC and forwarded to the appropriate GE commercial representative.

No satisfactory method has been devised as yet for providing a mechanically stable beta current in-core flux monitor; however, development work is being accelerated in an effort to effect a solution.

The initial in-core test was completed at the KW Reactor test facility on the microwave neutron flux monitor. Promising results were achieved with the monitor which employed a nitrogen-filled resonant cavity for the test. An unexpected temperature rise, monitored by incorporated thermocouples, caused the measured frequency to shift to a lower value; whereas the effects of the neutron flux would cause a positive shift. The temperature change caused the cavity dimensions to increase. Later tests in the laboratory, with similar cavity heating, indicated that an appropriate frequency shift had been measured at KW Reactor which was probably due to the neutron flux, as desired. A new cavity has been designed in an effort to achieve better cooling during the next reactor test. The new capsule will contain He-3 rather than nitrogen to provide a considerably greater neutron cross section and, hence, a greater positive frequency shift.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

Evaluation of multiparameter test principles for testing of tubing is continuing. Special interest is being placed on the application of the multiparameter tubing test to the detection of hydride in process tubes of N Reactor. The proposed scheme of linear scanning with a large circular coil which is concentric with the tube is being evaluated. Determination must be made of the sensitivity with which hydrided areas may be detected using this approach.

The detection of hydride requires increased instrument sensitivity and circuit changes are currently in progress which will give improved performance. A balanced detector circuit has been built which eliminates the opposing battery used in the original circuit, and which should reduce instrument

electrical noise. For use in evaluating the tubing tester, a prototype phase shift circuit using a 400 cps resolver and simple RC network was built and found to give excellent performance up to 2.5 kc/sec. For use at higher frequencies, an order has been placed for a 5 kc/sec to 250 kc/sec resolver.

A test section of an N Reactor process tube having holes drilled on both the outer and inner surface was received. This test sample will be used in evaluating the multiparameter technique for testing tubing.

A study was undertaken to determine the factors affecting the performance of a tuned air core transformer to be used to couple energy from a rotating eddy current test coil to the instrument. This study resulted in the development of a technique which will permit the discrimination against undesirable effects due to variations in mutual inductance between the primary and secondary windings. These coupling variations result from the change in winding separation as the secondary winding is rotated. The analysis of the coupling circuit equations indicated that by proper adjustment of the ratio of secondary loop reactance to the secondary loop resistance, the effect of coupling variation will give the same phase angle as probe lift off. This ratio is determined from the in-phase and quadrature components of the test coil lift off impedance. Employing this technique, both transformer coupling variations and probe lift off can be simultaneously discriminated against by the usual phase discrimination methods.

The coupling variations compensation was demonstrated by using a vector impedance plotter which verified that phase angle from probe lift off and coupling variations could be made equal. Based upon the results of this study, a prototype air core transformer has been built and is being further evaluated for application to the hydride detector and other tubing testers using a helical scanning test head.

The Model 1004 eddy current tubing tester was used to detect cracks in the Zircaloy single-sheathed thermocouple assembly of 1/8 inch diameter. All indicated crack locations were confirmed by destructive examination.

Tests were also conducted on dual-sheathed Chromel-Alumel thermocouple assemblies. These tests indicated that the magnitude of undesired signals caused by the magnetic Alumel wire are reduced by a factor of ten by magnetically saturating the material during test.

Fundamental Ultrasonic Studies

Tungsten loaded epoxy samples were fabricated for use as backing members on shear wave transducers. These backing members are being designed to provide

sufficient transducer damping to give pulses of short duration. Such pulses are necessary for the attenuation measurement studies currently being conducted. The fabrication of the previously designed clamp to be used in these studies for holding transducers and test samples has been completed. The surface grinding of the aluminum, stainless steel, and Zircaloy-2 attenuation test samples have been completed. Experimental requirements necessitate that the surfaces of these samples be parallel to within 100 micro inches per inch. The gating circuit to be used for pulse frequency analysis work was found to have a nonlinear frequency response. Several new gate circuits have been designed and evaluated. In addition, an effort is being made to improve the frequency response of the original circuit by using high frequency transistors.

As a parallel effort in the pulse frequency work, a computer program has been written which will give frequency analysis information from a pulse amplitude as a function of time information. A sampling oscilloscope will be used to measure the pulse amplitude versus time data which is necessary for the computer program.

Ultrasonic wave diffraction and scattering mechanisms are being studied to determine their influence on nondestructive testing applications. Both diffraction and attenuation effects change the shape and amplitudes of ultrasonic pulses and, for an understanding of pulse propagation, it is necessary to distinguish diffraction effects from those caused by attenuation. The analytical treatments which deal with diffraction in transducer fields and the diffraction fields which are scattered from test sample imperfections are mathematically complex. To augment the planned analytical work, experimental measurements are presently being conducted. The diffraction fields of transducers are being measured and samples are being prepared for experiments on the diffraction fields scattered by small objects in water. Methods for obtaining small imperfections of known size and shape in solids are also being investigated.

Since metal fatigue initiates near the surface of materials it is of interest to determine whether ultrasonic critical angle techniques might be applied for fatigue detection. Six metal samples, two each of two types of stainless steel and two carbon steel, were tested for initial conditions prior to subjecting them to fatigue. One of each will be retained as a reference sample to compare with the fatigued samples.

Theoretical studies of Lamb waves in a cylinder were continued and made more general so as to include axial and spiral waves. The theory proved to be not difficult but the equations are lengthy and it is time-consuming to reduce them to usable form.

Zircaloy-2 Hydride Detection

Although N Reactor process tubes have smooth inner surfaces when installed, surface roughness is expected to result from normal fuel loading and discharge. Because this normal tube usage will result in tube roughness and since a large percentage of the received test samples had internal gouges, it was felt necessary to further evaluate the effect of uneven surfaces on the hydride detection instrument. Tests were conducted on a bench standard section having rough areas of 32, 63, and 125 micro inches finish on the inner surface. Of these test areas, only the 125 micro inch finish caused an appreciable change in the hydride detector output and the signal change from it was approximately the same as from a 400 parts per million hydride area.

A bench standard also was prepared with longitudinal gouges using a tool with a one inch radius tip. Testing of these samples indicated that an output signal from a 0.005 inch gouge was approximately equal to that from a 400 parts per million hydride area. Gouges of this depth are not expected in N Reactor process tubes but they have been present in several of the hydride test samples fabricated from rejected tubing.

Metallographic analysis of the hydrided bench standard currently being used revealed that it contained an average of 500 parts per million hydrogen in the prepared longitudinal strip which was 1/2 inch wide. Eddy current tests had previously indicated a 0.4 micro ohm-centimeter increase in resistivity within the hydrided area. Non-uniform hydride distribution made it difficult to obtain a precise correlation between resistivity and hydride content. Efforts are being made to obtain additional samples with precisely known hydride content to obtain a calibration of resistivity changes versus hydride content.

USAEC-AECL COOPERATIVE PROGRAMNondestructive Testing of Sheath Tubing

Use of the prototype sheath tubing test to inspect Hanford sheath tubing continued. Approximately 3000 tubes have been inspected to date which provides excellent experience with operation of the unit and also points out potential points of wear. During the period while the tubing tests were being performed no electric failures were experienced; however, some wear occurred in the mechanical system. The mechanical wear points were strengthened to develop a unit which will require minimum maintenance. The tubing passes through the tank on "O" rings which rotate with the tubing and the tubing translation through the tank is causing some wear on these support rings. The "O" rings are purposely chosen to wear so that a minimum

wear will occur on the tubing. Consequently they must be replaced at regular intervals; however, the remainder of the tester should operate over long periods of time without maintenance.

The general utility of the instrument was further demonstrated by setting it up to test two types of tantulum tubing. The tubing wall was 20 mils in thickness and 1 mil axial and circumferential notches were easily detected with the zigzag shear waves. One tube was 1 inch in diameter so the instrument repetition rate of inspection was increased to 10,000 cycles per second so that the 1800 RPM tube rotation and 18 inches per minute translation could be maintained. A second tube was 1/4 inch in diameter requiring the ultrasonic entry angle for the axial flaw test to be set shallow to obtain a uniform test for both OD and ID flaws. Both standard and natural flaws were reliably detected in this tantulum tubing.

Fabrication of the final tubing test mechanical system was completed. Minor revisions were required and the mechanical system is presently being integrated with the electronics to produce an operating tubing test station. Stability tests on the electronics were also conducted. Some temperature drift problems were encountered and corrected with higher quality temperature stable capacitors.

Assembly of the data obtained on notches in tubing ranging from 10 to 65 mils in wall thickness pointed out some discrepancies which required re-evaluation of the notch depth versus amplitude data. It was noted that notches filled with foreign material gave a lower signal amplitude than predicted. After the notches were cleaned with alcohol the signal amplitudes approached the predicted values. The data from the Zircaloy, aluminum and stainless tubing fall into a pattern predicted by zigzag shear wave propagation analysis. Micro photographs of the stainless tubing showed the average grain size to be 2 mils with some grain sizes well above 4 mils. The tubing noise level or signal received from the grain boundaries was equal to the signal from a 2 mil notch. The minimum size of detectable defects in tubes with large grain is approximately one-half the largest grain size. Another limiting condition for the test is reached when the diameter to thickness ratio of a tube decreases under 5 as reported earlier. Tubing can still be tested below this value but the test becomes less sensitive to ID flaws.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Analysis of turbulence data for atmospheric dispersion studies will be facilitated by a program established on the IBM 7090 computer for determina-

tion of the power spectral density distribution of wind velocity fluctuations. Power spectra derived from the program were verified by hand calculations of the input data. Additional options for the program are now being investigated. These include calculations of velocity autocovariances and cross-covariances, basic parameters relating to atmospheric turbulent motions and the dispersion of material in the atmosphere.

Four dispersion experiments were completed during the month. Three tests were made to obtain data on the suitability of trichloroethane as a carrier for zinc sulfide in the dispersal system for the atmospheric tracer. Use of sodium lauryl sulfate and water limits the rate at which zinc sulfide can be dispersed without forming excessively large droplets. The higher volatility of trichloroethane should permit higher release rates, thereby improving the sensitivity of the atmospheric tracer technique when used over large travel distances. Another experiment was successfully completed in the elevated release series in which zinc sulfide was released for a period of one-half hour from the 185-foot level of the meteorology tower. Exposure data were obtained to a distance of 1600 meters downwind.

Reduction of the data from the elevated source experiments conducted during 1963 was completed and analysis of the 24 experiments started. First results show greater variation in the exposure values at 1.5 meters height than occurred in earlier experiments. These variations reflect primarily the broader ranges of atmospheric stabilities and wind speeds embraced by the 1963 Series.

Rankin counter #3, the standard for zinc sulfide assay, became inoperative during the month. After many years of reliable performance, the rotating member failed to turn. For the first time since 1959, the counter was disassembled, cleaned, and lubricated. This action restored the counter to use. Recounts of a number of filters show that the counting characteristics remain unchanged. In addition, monitoring of the interior of the counter showed no contamination. The electroplated platinum source has remained intact for five years.

Radiological Physics

Results of the whole body counting at Anaktuvuk Pass in March became available. The average body burden of the Eskimos was 28% higher than it was in January. The amounts of Cs^{137} are still not as high as they were last summer. The increase is probably due to eating of fresh caribou meat. Caribou herds appeared at the Pass in late January after an unusual absence.

Counting of urine samples had indicated that body burdens at Anaktuvuk Pass increased about 50% between January and February. The whole body counts of

the people sampled showed a 31% increase between January and March. The urine specimens for March just arrived. No results are available yet for quantitative comparison; however, the urine counting did correctly indicate that a substantial increase had taken place.

The amount of Cs^{134} in the Eskimos has decreased until it is nearly undetectable. Apparently recent fallout does not contain this isotope.

Na^{22} continues to be present in Eskimo urine samples. A careful study of whole body counter spectra of Eskimos showed there was no discernible peak due to the isotope. It was detected in reindeer meat but there was not enough to explain how much there is in the Eskimos. Radiological Chemistry found a similar situation at Richland; they found minor amounts of Na^{22} in meat with much more in wheat and other grains.

Urinalysis gave 155 and 198 nCi for the average Cs^{137} body burdens at Pt. Hope and Fort Yukon. These are slightly less than the values a month earlier.

Radiological Chemistry found Sr^{85} in hair samples from the pig that was injected last month. They also detected Sr^{90} in five hair samples collected at Anaktuvuk Pass last month.

Radiological Physics, Radiological Chemistry, and Inhalation Toxicology conducted a study of the inhalation of dust and radioactivity at two uranium mines in Colorado. The Hanford Mobile Whole Body Counter, modified to permit differential scanning, was used in the study. Measurements were made in a mine that was in operation and in one that was not in use. Six subjects were used. Measurements were made of uranium, radium, radon, and radon daughters and dust-particle size in the mine air. Breathing rate and volume were measured. Whole body counting gave the total deposition and the relative distribution and translocation of radioactivity in the body. Body burdens were high enough to permit accurate counting. There was 150 to 700 nCi of Bi^{214} , the isotope used for measurement, in the subjects plus several other isotopes. The radioactivity had very short half-life so the success of the study was due to having the mobile counter right at the mine. The results of the experiment are not yet available, but preliminary examination of the data shows that new and significant conclusions can be expected.

The dog we counted on our plutonium counter last month was counted again. A considerable difference was found in the counting rates. This was doubtless due to a significant reduction in weight of the dog. The dog died and its body is being radiochemically assayed for plutonium and a determina-

tion of the isotopic composition of the plutonium is being made. Six other dogs were counted so that ultimately a correlation between counting rate and body burden can be established this same way. Correlations obtained by counting a source in a phantom or in a tube stuck down the trachea of a dog were made but are not trusted because the distribution of the radioactivity in the body was not the same as in the exposed dogs.

The positive ion accelerator performed satisfactorily during most of the month. Repairs were necessary on the lens power supply.

Prof. L. W. Seagondollar, University of Kansas, and consultant to Radiological Physics, spent a week at Hanford working with us on adapting the Van de Graaff to accelerate helium ions. We believe that the doubly charged helium beam has been located and that it contains the right amount of current (a few tenths of a microampere). One of the high voltage supplies in the Van de Graaff will have to be changed before a useful beam can be extracted from the machine.

Another leak developed in the He^3 proportional counter neutron spectrometer. Before that, it was partially tested with a small amount of He^3 and found to be working satisfactorily.

Calorimetric measurement of a second Pm^{147} source gave a half-life of 2.66 years. Measurement of the first source gave 2.68 years. These are significantly different from the currently accepted value of 2.5 years but in good agreement with the 2.67 years determined by the Hanford Chemical Laboratory.

Instrumentation and System Studies

Experimental fabrication was initiated on a laboratory mockup assembly for the animal physiological function and dose telemetry system. The system will include thyroid (I-131) dose measurement, respiration rate, and temperature determination channels. A number of miniature detectors were received for I-131 measurements and two required miniature high voltage supplies were fabricated and tested.

Development progress and initial testing were achieved on the solid state circuitry to automatically control air and smoke inhalation for animal biological studies. The instrumentation includes a respiration counter, a counter driver, gate, reset driver, and relays for control of the solenoid air valves. The associated power supply was designed and fabricated.

The 1.5 inch multiplier phototubes and necessary NaI crystals were received and are being assembled for use in the Biology Laboratory dog counter. Resolution and scanning tests were planned.

The alpha energy analysis air monitor, which was developed for use during Biology Laboratory radionuclide inhalation experiments, was completed in final system form and laboratory testing was initiated. The system will be delivered to Biology as soon as the definitive tests are completed.

Careful examination of the experimental portable field-type radiological spectrometer, developed for Biology Laboratory field studies, revealed the causes of the minor drifting and non-linearity problems. Corrections are being incorporated.

Difficulties persisted in the application of a new conducting coating material to the center rod of the ion chamber for the field prototype miniature recharging gamma dosemeter. The material is not only difficult to apply, it also provides a non-uniform coating. Experiments will continue.

Drawings of the electronic circuitry were essentially completed for the advanced experimental model alpha-beta-gamma head and shoe counter. A tester for the many modular circuit boards was designed and fabricated and a comprehensive instruction manual was partially completed.

Several experimental scintillation detector probes were designed and fabricated in final form to evaluate their use in detecting Pm-147 and other low energy beta emitters. The completed probes incorporate both 2-inch and 5-inch diameter multiplier phototubes, 0.010 inch thick terphenyl in polyvinyltoluene scintillators, and solid state preamplifiers. The detector probes can be used directly with the present 95 HAP0 "Scintran" radiation monitors.

The experimental, six decade, logarithmic response radiation monitor was completed. It employs all solid state circuitry and provides a reduction of phototube applied high voltage in proportion to the increased level of the ambient radiation field. The main instrument performed satisfactorily in tests; however, difficulties were encountered in achieving appropriate performance from the phototube.

The spectral density analysis of meteorological data involves wind speed and direction signals. Considerable equipment difficulty was encountered when the tape had to be sped up by a factor of 64 in order to permit analysis of slowly varying signals. Spectrum and cross spectrum analysis was then performed for frequencies varying from 0.05 cps to 2.4 cps. The

spectrum analysis in itself was carried further to 154 cps.

The scope of the meteorology data handling problem was expanded to include more than the wind diffusion parameter measurements. A recommendation was made for a system to measure the wind diffusions parameters, but the expanded scope of the new problem requires additional study and is now under consideration.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples during the month in accordance with current schedules.

Studies of the performance characteristics of the mass spectrometer in an ion-beam-switching and ion-counting mode of operation were resumed after the instrument component malfunctions reported previously were corrected. From a series of analyses of NBS uranium isotopic standards using both single-filament and triple-filament ion-sources the instrument bias was determined to be less than 0.3% at the 95% confidence level. However, the single-filament ion-source appears to exhibit a mass-fractionation effect dependent on sample life.

The unacceptable alumina parts for a vacuum-lock sample changer for the mass spectrometer were returned to the vendor for repair or replacement.

EXPERIMENTAL REACTOR PHYSICS FACILITIES

PCTR Operation

The PCTR was operated on an intermittent basis during the month. There was one unscheduled shutdown due to electronic failure.

Two sets of stainless steel samples containing 0.3% and 0.7% gadolinium were measured for neutron absorption and compared with cadmium and stainless steel samples.

A set of foils was irradiated for the NPR hot test. The foils were irradiated simultaneously with a similar irradiation in NPR for normalization purposes.

TTR Operation

The TTR was not operated during the month.

Subcritical Facility

A rerun was started on the 0.80-inch lattice with Pu-Al fuel 2.0 w/o Pu - 16% 240.

The water dump valve system was modified so that a high level trip on Channel 2 would dump the water in the tank.

COMPUTER FACILITIES

Bids for the process control computer van were received April 20 and are now being evaluated by Transportation and Maintenance. A work order was issued to cover electrical and signal connections to the van.

The letter requesting a change in the computer feasibility study to permit purchase of the equipment has been forwarded to the Atomic Energy Commission for their review.

The analog computer applications class is in its seventh week. All fourteen of the original class members are still enrolled. Three members have indicated interest in having analog computer simulation studies made of problems in their respective work areas.

A Technical Activities Seminar introducing the MIDAS (Modified Integral Digital Analog Simulator) system to Hanford personnel was given.

The analog computer utilization was as follows:

<u>EASE 1132</u>	<u>EASE 2133</u>	
162	164	Hours Up Time
8	8	Hours Scheduled Downtime
<u>6</u>	<u>4</u>	Hours Unscheduled Downtime
176	176	Hours Total

The time shown does not include overtime.

Problems considered during the month were:

1. PRCF Hazards.
2. N Reactor Cold Water Injection.
3. Meteorology Problem.
4. N Reactor Secondary Loop.
5. N Reactor Primary Loop.
6. ANN Recovery (CPD)

The new analog computer has been in use for about six weeks. Although numerous failures have been encountered, without exception they have been component failures and were not unexpected for a new computer installation of this size.

The second week of the maintenance school was completed on April 10. Personnel gained a great deal of practical "know-how" relative to maintaining the new computer.

CUSTOMER WORK

Weather Forecasting and Meteorological Services

Meteorological and climatological services included 1) a review of atmospheric pollution problems associated with Phase III of the Waste Management Program for CPD, 2) preparation of climatological and meteorological data for use in review of the status of the Wapluke Slope for HL, 3) computation of evaporation rates for the N-reactor quench tank for NRD in conjunction with the reactor accident appraisal, and 4) provision to CPD of meteorological data pertinent to the exhaust filter problem.

Meteorological services, viz., weather forecasts and 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	90	84.0
24-Hour General	60	87.8
Special	198	90.9

April was dry and cool. A cool period at midmonth brought freezing temperatures on the 16th, 17th, and 19th and dropped the average temperature to 50.2 degrees, 3.2 degrees below normal. Only 0.11 inch of precipitation fell during the month and 0.07 inch of this fell during one 24-hour period on the 22nd and 23rd. The wind averaged 10.1 mph, which was 1.3 mph greater than normal.

Mass Spectrometry

Isotopic analyses were provided on 10 uranium samples in support of HAP0 U²³³-production studies and on 2 uranium and 2 plutonium samples in support of the Plutonium Regenerating Detector Experiment of Nucleonics Instrumentation.

Determination of the isotopic composition of NBS plutonium standard sample 948 was begun. Nine analyses of this sample were performed using both single-filament and triple-filament ion-sources in the heavy element mass spectrometer equipped with the new scintillation-type ion-detector. A preliminary value of 0.08666 ± 0.00008 (95% confidence) was obtained for the Pu²⁴⁰/Pu²³⁹ atom ratio and reported (HW-81969) to the Analytical Standards Coordinator at the National Bureau of Standards. During the period of these analyses the instrumental bias was determined from thirteen analyses of NBS uranium isotopic standards to be zero to within 0.3% (95% confidence).

Instrumentation and System Studies

An instrument was designed, fabricated, tested, and delivered on a priority basis to Advance Fuels Development, HL, for use in measuring the ohmic value of beryllium pre-weld resistance. Satisfactory instrument operation was achieved and tests indicated that widely-varying pre-weld resistances may be responsible for the persistent welding difficulties. The average resistance measured was 1 milliohm; measuring currents of approximately 50 A were employed.

Minor progress was achieved toward development of a soft beta detecting mask monitor for use at the HAP0 Laundry Facility. Difficulties persisted in achieving a suitable light pipe for the scintillation detector.

The completed U-235 fuel enrichment monitor was delivered to Metal Fabrication Development, HL, following completion of detailed laboratory tests. A combined topical report and comprehensive instruction manual was prepared and issued.

A new sense amplifier for the 400-channel analyzer has been developed. The new circuit, using tunnel diodes, allows operation up to about 120°F without temperature compensation. Parts for the new sense amplifiers have been received and the new circuits are being built.

A simulation of a portion of a high temperature gas loop was conducted for Reactor Fuels Research and Development, HL. The basic purpose of the simulation was to demonstrate the capabilities and areas of applications of the

analog computer facility for problems relating to a high temperature gas loop.

Evaluation of the General Electric GEMAC temperature control system purchased for use on the uranium swelling program has been completed. The system is now installed at the 100-KW test facility and will control the next in-reactor swelling capsule. Design of a silicon controlled rectifier circuit for use as a final control element in the GEMAC system has been accomplished. The circuit makes use of controlled avalanche rectifiers as transient voltage limiters to protect the SCR's. The SCR stack will be installed during the next capsule charging.

Purchase specifications were written for a precision cryogenic temperature control system for use by Reactor Metals Research personnel in low temperature metals research.

A Faraday shield enclosing the logic circuits of the creep data logger is being constructed to further isolate the system from electrical noise. Filter capacitors on all incoming signals will also be installed. Ninety percent of the original noise has been eliminated, by floating the entire system with respect to reactor ground and by careful selection of internal system ground points.

Optics

A camera capable of obtaining a picture of the entire internal surface of an N fuel element on one piece of 35 mm film is being developed in the Optical Shop. The first developmental model of this camera, which is intended for use in tubes approximately 0.4 inch in diameter and 30 inches long, is now being fabricated.

It was requested that a camera be modified to permit photography at natural size of resistance temperature detection units on the reactor rear face. A carrying box, lens extension, RTD holder and identification pad have been designed to permit rapid photography of these units under radiation zone conditions.

A right angle eyepiece with direct target viewing has been designed for the Silicon Solar Cell Pyrometers. One unit will be converted to this type of eyepiece.

During the four-week period (March 15-April 12) included in this report, the following shop work was performed:

1. Two hundred fifty-six (256) stainless steel pieces were aluminized for Advanced Fuels Development.
2. Ten (10) blocks of quartz, steel, zirconium, and aluminum were lapped flat and parallel for use as standards in ultrasonic work.
3. Twelve (12) glass bearings were fabricated for CPD.
4. Most parts of the first developmental model of the Pan-Bore on ID Slit Camera were fabricated.
5. Over 120 man-hours of service work was done on the Purex crane periscopes.
6. A stereomicroscope was rebuilt for Radiometallurgy.
7. A stereomicroscope was repaired for 105-C, FEF.
8. A projection comparator was adjusted for 306 Building.
9. Numerous small machining and glass cutting jobs were performed.

Physical Testing

A newly designed fuel element which uses a 3.070 inch diameter Zircaloy tube with eight ribs is being fabricated by Advanced Fuels. To examine the rib, rib to tube weld, and the tube wall under the rib, a two transducer ultrasonic test was developed. The ultrasound beam transmitted from one transducer propagates through the rib to the inside surface of the tube. At this point it is reflected back and received by the second transducer. The amplitude of the received signal is monitored. The two transducer test proved to be effective in locating defects in the ribbed tubing and should be easily adaptable to any sized diameter as well as during any part of the fuel manufacturing sequence or after irradiation.

Strain gages were applied for axial and biaxial stress measurement on the pump discharge piping on the PRTR Rupture Loop. Failure of a weld in this highly stressed area occurred as a probable result of vibration, and the strain gages have been installed to provide a convenient means of constantly measuring both the strains and the vibrations during various phases of operation.

Engineering assistance in the development and operation of instrumentation has resulted in the successful completion of a program to measure the electrical resistivity of plutonium as a function of density during phase changes. Results have shown that the resistivity and density changes during an alpha to beta phase transformation in unalloyed plutonium occur at the same time and in approximately equal percentages. Resistance measurements can now be reliably applied to measure density and phase changes.

Work continued on the scoping of a facility to inspect waste storage tanks in support of Waste Solidification Engineering Development Operation (WSEDO). As a result of fiscal problems within WSEDO, efforts this month

were restricted to the wall thickness and internal pressure measurements. A novel pressure gage to measure both positive and negative pressure within the tanks, without compromising tank integrity, is being designed. This simple device consists of a sealed bellows mechanism and a pair of permanent magnets for the transmission of motion through the tank wall.

A new test to determine the amount of deterioration of wooden utility line poles has been applied to several hundred poles to collect the necessary data to properly evaluate the test in field conditions. The test, which measures the restoring force of the pole, differs from previous attempts to measure the quantity of deteriorated wood directly. Correlation of restoring force and defective wood have been obtained with all poles removed for examination thus far.

Completed fuel elements for the Experimental Boiling Water Reactor and some fuel capsules were ultrasonically inspected to determine if vibratory compaction loading had created any cracks in the Zircaloy-2 cladding. Some difficulty had been experienced with cracks forming in the cladding near the hold down clamps. Three of ten fuel rods were found to have defects of two mils or larger in the area of the hold down clamps, where no defects greater than a mil had been present. No defects were found in the twelve capsules tested.

Approximately 3800 short tube ends removed from the Steam Generator 4-A at N Reactor are being carefully inspected for location and cataloging of the heat and tube identification numbers. The program is intended to answer one of the remaining questions as to the cause of failure in the steam generators, and that is whether specific heats of steel were more prone to attack than others.

An eddy current probe and test to evaluate the feasibility of testing PRTR process tubes for internal blisters is progressing. The problem is to locate and determine size and depth of the defects. An encircling coil has been made which locates the defective area. A point contact is used that determines the circumferential position of the defect. An area $1/4$ " long by $5/32$ " wide area can be mapped. The extent to which the flaw depth, or remaining wall, can be determined is not yet known.

Radiography continued to be one of the larger aspects of services requested from field testing. New pipe lines, new tanks, and new pressurizers being installed were all radiographed as the final welds were made in the field. Structural members, supporting fuel buckets in the basins, were radiographed as a check on the quality of the welding techniques. Parts of, or complete items of, operating equipment have been inspected to determine the presence of defects, cracks, or voids that may contribute to their early failure.

These items include vanstoning dies, pump impellers, process splitter blades, and retaining rings. In many cases the condition found by this inspection technique measurably contributes to the over-all operating efficiency and economy of the plant.

Nondestructive Testing for Isotope Containment Cells

Continued support in testing prototype containment cells for wall thickness and internal cracks is being performed. Several methods of ceramic transducer fabrication are under investigation. Different types of transducer backing and the bonding of the backing member to transducer are of primary interest.

Although the final requirements for the nondestructive tests on the cells are not known, a general purpose ultrasonic test system is being designed and fabricated. Also, an active program has been started to investigate methods of determining basic thermal and structural properties of the isotope cells.

Nondestructive Testing - NASA

Authorization has been received from Lewis Research Center for development and evaluation of advanced nondestructive tests in support of NASA programs. Objectives of the program are:

- A. Evaluate the feasibility of nondestructively inspecting high temperature nuclear fuel elements and reactor components of interest to NASA programs.
- B. Determine the most promising inspection methods capable of insuring the integrity of these components.
- C. Experimentally evaluate the applicability of the proposed test methods and demonstrate their effectiveness as applied to bona fide specimens through destructive and nondestructive tests.

Work under this program has begun. As the effort involves the investigation of classified materials, subsequent regular progress reports will be issued through classified channels.

INSTRUMENT EVALUATION

The maintenance personnel training program, which was initiated to assure properly trained technicians for the scintillation alpha-beta-gamma hand and shoe counters, was continued. A total of 10 counters have been received from two offsite instrument fabricators. The first seven received have now been carefully checked, adjusted, and calibrated. Two of these have been placed in field service and satisfactory performance is being achieved. The last three received instruments have not been completed in acceptance tests.

All electronic circuitry and appropriate mechanical assembly drawings for the advanced model combined alpha-beta-gamma hand and shoe counter were thoroughly reviewed and appropriate changes were incorporated on the tracings. The laboratory prototype detector probe was evaluated for alpha detection sensitivity. The efficiency averaged 13 percent for the HAP0 standard distributed 500 dis/min Pu-239 source with probe corner efficiencies of 9 to 10% and a center efficiency of 21%.

One pre-production model high range (to 5,000 R/hr) CP-TP portable radiation survey instrument was evaluated. Twelve instruments are being purchased. In addition, one pre-production model, of 100 on order, regular CP (to 5 R/hr) instrument was evaluated.

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CHEMICAL LABORATORY
RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

N-Reactor Fuel Overheating Study

Programmed heating and quenching tests of N-Reactor fully-irradiated fuel elements were completed. Two outer and two inner elements were heated to 1800 F, then quenched with 150 F water in these experiments. All four elements retained cladding integrity. Some blistering, slight wrinkling, swelling, bowing and cladding separation from the fuels were noted. The only radioactive material appearing in the quench water was attributed to activation radionuclides, supporting the visual observation that no cracks or other breaks occurred in the cladding.

The outcome of these experiments yielded encouraging evidence that during a postulated 5-minute delay in operation of the back-up coolant system, cladding of highly irradiated elements will not break and expose the fuel.

Estimate of Isotope Removal from Reactor Coolant by Blow Sand

At the request of WRD Process Evaluation and Control Engineers, estimates were made of the removal of isotopes from reactor coolant diverted to N-Area surge tanks by blow sand in the tanks. Based upon previous work in our laboratory, it was estimated that about 3 percent of the iodine and about 40 percent of the strontium would be removed by the sand.

Electrodeposition of Nickel on Uranium

Recent laboratory studies have pointed the way to a significant advance in the technology for plating nickel onto uranium metal, through the use of a 6.0 M H_2SO_4 - 0.32 M HCl etching solution in the uranium surface preparation procedure. The use of this etching solution, followed by the usual nitric acid smut-removal step, paved the way for the electrodeposition of an adherent nickel coat with a porosity about one-twentieth the porosity of a coat of the same thickness (18 microns) applied following an $HCl-H_3PO_4$ etch. The reduction in porosity thus achieved has permitted a significant (40 percent) reduction in the thickness of the plate needed to produce the same degree of uranium surface protection as obtained following the $HCl-H_3PO_4$ etch. The preliminary data indicate that anodization of uranium in H_2SO_4 produces a uniformly passive grain

surface. The HCl slightly decreases this passivity and provides preferential etching at the grain boundaries to provide adherence by a lock and key effect.

In these experiments, HCl-HNO₃ and HCl-trichloroacetic acid mixtures were also tested as etching solutions, but with less satisfactory results.

SEPARATIONS PROCESSES

Resin Stability Studies

Since the fire in the Redox ion exchange facility, several experiments aimed at evaluating possible resin hazards have been performed. The more important of these include study of the effect of gross amounts of dichromate in the feed, study of the composition of the gases evolved by resin degradation, and studies of the degradation of resin in pressurized nitric acid above 100 C.

Dichromate at high concentration levels was found to markedly increase the rate of degradation of Permutit SK resin. Eight hours of contact at 65 C with a solution containing 7 M HNO₃ and ca. 1 M CR₂O₇⁼ produced quite noticeable softening of the resin. The gas evolution was much less than that which would be obtained with the same degree of degradation due to nitric acid attack alone.

Gas analysis was obtained on gas produced by the degradation of resin by static nitric acid containing iron, chromium, sulfate and thorium (in place of Pu) in the amounts present in Redox ion exchange feed at the time of the fire. The results showed N₂, 77 percent; CO₂, 14 percent; N₂O, 7 percent; NO, 1.7 percent; NO₂, H₂ and hydrocarbons, <0.01 percent; CO, <0.5 percent. This gas will neither burn nor support combustion.

Six experiments have been carried out on the degradation of Permutit SK in nitric acid above 100 C. In the first of these, 40 ml of resin just covered with simulated Redox feed was heated rapidly in an 80 ml bomb. At about 170 C the rate of temperature rise exceeded that due to furnace heating, and at 195 C the temperature-time curve (with temperature the ordinate) became essentially vertical. The temperature indicator went off scale at 400 C, a 6500 psi rupture disk was blown out, a 5000 psi gauge split, and the bomb (1/2-in. thick stainless) was expanded about 5 percent in diameter. It appears that the combination of fast heating and very strong confinement allowed the exothermic reaction to reach a temperature which produced an extremely high reaction rate before the pressure could be relieved through the rupture disk and the reaction stopped.

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In the other five runs, 1.5 to 2.5 ml samples of resin were used in 40 ml of solution. At 150 to 155 C in simulated Redox feed, little if any resin damage occurred in 25 minutes. Resin heated to 200-205 C for 15 minutes in simulated Redox feed was completely dissolved. Resin held at 178 ± 2 C for 25 minutes was very badly damaged but not destroyed. At this temperature unloaded resin is damaged somewhat worse than thorium loaded resin, probably because of higher nitric acid content of the unloaded resin. Resin held at about 180 C for 25 minutes almost exactly duplicates the badly degraded resin from the Redox contactor.

Plant equipment scale investigations have been initiated on the chemical-thermal stability of anion exchange resins. These studies, extending laboratory work, are intended to (1) investigate the possibility of an equipment scale effect on initiation temperatures, (2) explore the mechanism of deflagration, and (3) investigate the physical consequences of overheating and the effectiveness of possible protective measures.

An initial plant equipment scale test has been conducted with a 4-inch diameter by 24-inch deep bed of Permutit SK resin equilibrated with 7 M HNO_3 and drained. The test bed was heated in a tube furnace until bursting of a 4-in. diameter, 300 psig rupture disk occurred. Maximum resin bed temperature is unknown due to biased thermocouple readings but was between 130 C and 270 C. Following rupture disk bursting, the resin remaining in the test bed continued to burn for more than one hour.

Additional experiments are planned.

Uranium-233 - Aqueous Processing

Hot-cell work during the month was concentrated on the gram-scale purification and isolation of "pure" (ca. 1 ppm U-232) U-233 and on the recovery of a small sample of "ultra pure" U-233 (from decay of Pa-233 isolated several months ago). Laboratory effort was initiated on the evaluation of protactinium-selective adsorption or scavenging agents. Adsorbents initially tested included: unfired Vycor, Duolite C-3, silica gel, Decalso, charcoal (Nuclar C-100-W), Linde 4-A, Linde 13-X, Linde AW-500, and Zeolon. Under the test conditions (2 M HNO_3 , tracer Pa-233) charcoal and Vycor were the most effective adsorbents (decontamination factors of about 40), followed by silica gel, Linde AW-500, Zeolon and Decalso. Adsorption was negligible on Linde 4-A, Linde 13-X or Duolite C-3. The more promising adsorbents will be tested over a range of concentrations of thorium, nitric acid, and fluoride (catalyst used in thorium dissolution and a possible interference). Other adsorbents and precipitants (both organic and inorganic) will be tested as well. Possible co-adsorption of ruthenium will also be explored.

Thorium Dissolution Studies

Denitrated, pneumatically compacted thoria in which 70 w/o of the material was in sizes greater than four mesh dissolved completely in boiling 10 M HNO_3 -0.01 M HF -0.03 M $\text{Al}(\text{NO}_3)_3$ in less than four hours to a terminal thorium concentration of one molar. Adequate stirring to keep fines suspended is required. Thoria prepared by way of the Mallinckrodt sol gel process dissolves somewhat slower than thoria which has not been gelled even though it has been pneumatically compacted.

Thoria from an irradiated fuel was dissolved to provide samples for analytical purposes. The thoria had been pneumatically compacted. About 93 percent of the oxide dissolved in less than three hours to a terminal thoria concentration of 0.83 M in a boiling solution initially 13 M HNO_3 -0.01 M HF -0.03 M $\text{Al}(\text{NO}_3)_3$. A second cut with a small volume of solution readily dissolved the remainder. Air sparging was used as agitation.

Extraction of Thorium with Diethers

The extraction of thorium from aluminum nitrate-nitric acid systems by the diethers, 1-4 diethoxybutane and 1-5 dimethoxypentane, was studied. Extraction of thorium by 1-4 diethoxybutane was about the same as by hexone. Thorium distribution ratios for 1-5 dimethoxypentane were a factor of about five higher than for hexone under similar conditions. The concentration of thorium in an organic phase in equilibrium with an aqueous phase 1.0 M $\text{Th}(\text{NO}_3)_4$ -1.5 M $\text{Al}(\text{NO}_3)_3$ -0.2 M HNO_3 was 1.11 M for 1-5 dimethoxypentane, 0.45 M for 1-4 diethoxybutane and 0.53 for hexone. The density of the 1-5 dimethoxypentane containing 1.11 M thorium was 1.431 g/cc (room temperature); its viscosity was 99 centipoises at 25 C.

A Pyrochemical UO_2 - ThO_2 Fuel Recycle Process

A pyrochemical reprocessing scheme for 3 percent UO_2 -97 percent ThO_2 fuels has been tested on a one kilogram scale. In this scheme the UO_2 - ThO_2 is ground to -200 mesh and dissolved in molten LiCl - KCl -0.37 CsCl with HCl at 625 C to give a solution containing 8 w/o Th and 0.25 w/o U. This dissolution requires 24 to 36 hours. Ninety percent of the dissolved thorium is then precipitated as ThO_2 by sparging for 48 hours at 625 C with 50 percent O_2 -50 percent Cl_2 . In the work to date, Th/U separation factors have exceeded 350 in this precipitation step, i.e., $(\text{Th}/\text{U})_{\text{ppt}}/(\text{Th}/\text{U})_{\text{salt}} \geq 350$. After precipitation of the ThO_2 the uranium may be recovered as UO_2 by electrolysis. A 30-hour electrolysis (at a constant 0.75 volts vs. a Ag/AgCl reference) removed greater than 98 percent of the uranium from the melt. The deposit was a solid solution of UO_2 - ThO_2 containing 46 mole percent UO_2 and 54 mole percent ThO_2 , and only 36 ppm chloride.

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Pu-238 Dose Rates and Shielding Requirements

A study has been initiated of the gamma dose rates associated with Pu-238, the probable contribution from Pu-236 impurity, and the possibility of minimizing the Pu-236 content by special irradiation procedures, analogous to those employed to make "pure" U-233. Preliminary calculations indicate that the dose rates at one meter from a 100 thermal watt source of pure Pu-238 would be about 0.9 mrem including 0.4 mr/hr, and from a 1000 watt source 6 mrem including 1.7 mr/hr. The gamma dose rates from pure Pu-238 are principally due to the 0.76 and 0.875 Mev Pu-238 gammas, and the gamma dose is non-linear with source size due to self-shielding. Contamination with Pu-236 would lead to a gradual build-up in gamma activity as the Pu-236 decayed to U-232 and ultimately to Tl-208, which emits a 2.6 Mev gamma in 100 percent of its disintegrations. Tl-208 build-up would reach about 5 percent of saturation in the first year, 15 percent at the end of the second, 50 percent in five years, and approximately full saturation in 10 years. The gamma activity from 1 ppm of initial Pu-236 would equal that from the Pu-238 itself in about 15 months and at saturation would be about 20 mr/hr at one meter from a 1000 watt source. Expected concentration of Pu-236 is about 20 ppm; however, Mound Laboratory reports that they have been unable to find Pu-236 (at an estimated detection limit of ca. 3 ppm), a mystery as yet unresolved.

Scrap Recovery Flowsheets

The currently favored 234-5 Building scrap recovery flowsheet involves coextraction of uranium and thorium in the first column and rejection of plutonium to the aqueous raffinate. It is desired that uranium and thorium concentration in the aqueous raffinate be 0.01 g/l or less. Miniature mixer settler runs were made to establish flow ratio and salting conditions to meet these requirements and for minimizing (by scrubbing) the plutonium in the Th-U product. Under best conditions found to date, the Th-U product still contained 2000-6000 parts of plutonium per million parts of uranium. Specification for plutonium in the Th-U product is 10 or less parts plutonium per billion parts uranium. It may be possible to meet this specification by spinning the stripped U-Th product through the extraction column with synthetic feed for as many cycles as are required.

Disposal to Ground

No significant changes were evident in the areal extent and concentrations of radiocontaminants in ground water during the past month.

Neutron Irradiation of Synthetic Zeolites

Neutron irradiation studies were continued on the high silica-to-alumina ratio zeolites. (The zeolite Linde 13-X, representative of the low silica-to-alumina ratio zeolites at 2.8, was irradiated last month to 3.16×10^{18} nvt with no changes in cation selectivity, capacity or structure detected.) Sodium-based Norton Zeolon with a silica-to-alumina ratio of 10 was chosen as representative of the high silica zeolites. Two 4-gram samples of Zeolon were sealed in silica capsules and irradiated to 3.14×10^{19} nvt in KW reactor. After one week the capsules were opened and X-ray diffraction patterns obtained on the irradiated Zeolon. A comparison of the X-ray diffraction patterns of the irradiated and unirradiated Zeolon samples revealed no structural differences. The cesium-sodium selectivity and capacity of the irradiated Zeolon were essentially unchanged.

Based on the above results, it is believed that the zeolites may be useful in reactions involving the neutron irradiation of adsorbed cations.

WASTE MANAGEMENT AND FISSION PRODUCT RECOVERY

Kilogram Scale Technetium Purification

The anion exchange purification step (on a 6-in. x 5-ft. IRA-401 column) was completed, and denitration and concentration of the 120 liters of product is in progress. Presence of a precipitate in the feed resulted in partial pluggage of the column. Part of the precipitate was removed by dissolution with 1 M HNO_3 , the remainder by backwashing with 120 liters of 0.05 M HNO_3 . Approximately 40 grams of technetium were "lost" in this operation (the wash solutions have been retained for possible reprocessing). The product was eluted with 134 liters 8 M HNO_3 and collected in eight product fractions. The product fractions contained 922 grams of Tc for an overall recovery of 94 percent and a material balance (including the back wash loss) of 99.2 percent. Moderate gas-sing was encountered during the elution, requiring occasional degassing of the column. Decontamination factors across the ion-exchange cycle were 2×10^3 from Ce-144, 10 from Ru-106, $>10^3$ from Cs-137, and 143 from the Zr-Nb that remained after silica gel treatment. Ruthenium removal was poorer than predicted from the results of laboratory studies, and a further ruthenium clean-up step will be needed.

Formaldehyde and sugar were both tried for the denitration step and the former selected for use, because of the presence of residual carbon compounds in a sugar-treated aliquot (due to absence of catalytic ions and radiation which aid the sugar denitration of Purex waste). Batch-wise denitration and boildown is proceeding.

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Purex Tank Sludge Study

Studies are in progress in A-cell of the High Level Radiochemistry Facility to pilot with actual sludge the leaching and dissolution procedure proposed by others for the Hanford Waste Management program (HW-79235, "Process Flowsheet for Purex Tank Farm Sludge Preparation," J.J. Shefcik, October 15, 1963). Two hundred milliliters (286 grams) of the 103 tank sludge sample described last month were used in the first run. The proposed process involves hot water sluicing, a water rinse, slurry transport, dissolution of the bulk of the sludge with nitric acid, and caustic digestion and acid leaching of solids remaining after the acid dissolution step. The various acidic solutions are to be combined for strontium extraction (by carrier sulfate precipitation), and the residual solids eventually returned to the waste tank. Significant observations to date are that the sludge solids remaining after the acid dissolution and caustic metathesis steps are very voluminous and that their settling rates are extremely poor, suggesting that it may be necessary to resort to centrifugation, rather than decantation, for phase separation. Analytical data on plutonium, neptunium and the fission products are not yet available.

Purex Ammonia Problem

Recent pluggage of the Purex Plant filter, apparently with ammonium nitrate, has raised a question as to whether the recently adopted sugar denitration of Purex waste produces significant amounts of ammonia and was therefore responsible. A series of laboratory experiments, simulating both "normal" and "over-killed" sugar denitrations, were performed to explore the matter. In all runs, the amount of ammonium ion formed was extremely small, ranging from undetectable to less than 10 ppm. These results strongly imply that the denitration operation was not the source of the offending material.

Cesium Removal from High Level Wastes

Laboratory experiments were conducted to determine the cesium loading on Linde AW-500 from a feed solution of simulated Purex alkaline supernatant waste containing additional cesium recovered by phosphotungstate precipitation from simulated Purex LWW. One part of a solution of cesium phosphotungstate dissolved in 2 M NaOH was blended with 11 parts of Purex alkaline supernatant waste solution to give a feed solution 4.2 M Na⁺ and 1.7×10^{-3} M Cs⁺, at pH 12.5. The cesium loading estimated on AW-500 with this feed solution was about twice that of Purex alkaline supernatant waste (0.16 meq/g compared to 0.084 meq/g). Experimental results indicate that 40 column volumes of the blended feed solution can be processed in one cycle to remove 99 percent of the cesium which is nearly the same volume as that for unblended waste. Analysis of

the eluates from the column runs showed Na.Cs ratios of about 7:1 after scrubbing with 5.5 column volumes of 0.2 M $(\text{NH}_4)_2\text{CO}_3$ solution. The higher alkalinity of the blended feed is not expected to reduce significantly the cesium capacity of the AW-500 based on results with AW-500 and Redox supernatant waste.

Strontium Semiworks Solvent Washing

Because of HA column flooding and high strontium losses, Strontium Semiworks solvent was washed with caustic and acid following recommendations based on laboratory studies. Examination of the washed solvent indicates the washing effectively removed gamma emitters and uranium. Disengaging times for the washed solvent are about the same as for fresh solvent. All presently available data indicate the solvent should show normal performance in Strontium Semiworks use.

Stability of BAMBP in CSREX Solvent

The extraction of cesium by CSREX process solvent (BAMBP-D2EHPA-Soltrol) decreases significantly during simultaneous exposure to gamma radiation and aqueous solutions encountered in CSREX processing of Purex LWW waste. Batch contacts simulating the CSREX process were made in which the solutions were irradiated with Co-60 gamma radiation during contact. During nine cycles simulating extraction, scrub, partition strip and solvent wash steps the cesium distribution ratio decreased from 5.7 to 3.0. The total gamma dose received by the solvent during the nine cycles was 44 watt-hrs/liter. During nine similar cycles but without radiation, the cesium distribution ratio remained essentially constant at 5.7. Trebling the concentration of sulfamic acid and hydrazine used as nitrite suppressors in the extraction column feed and strip column extractant, respectively, did not affect the rate at which cesium distribution decreased with succeeding cycles. Following the nine-cycle run (with radiation) the solvent was washed repeatedly with hot sodium carbonate solution. Only slight recovery of cesium extraction capacity was obtained indicating that solvent loading was not the primary cause for the reduced cesium extraction.

In-Tank Solidification

Evaluation of in-tank solidification using direct immersion heating is proceeding by means of both bench scale and pilot testing. Bench scale tests on various cold coating waste compositions have now been in progress for 350 hours with no appreciable scale formation. Included in this test series is one in a cobalt radiation source. In the pilot scale tests, prototype heaters from each of four manufacturers are being evaluated from a scaling and durability standpoint. One heater

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with a rating of 500 watts per square inch failed after 268 hours' operation. Lower rated heaters (18-40 watts per square inch) have operated satisfactorily up to 556 hours. Build-up of a deposit has been observed on the piping in the vapor space above the air circulator, presumably due to splashing and evaporation of the concentrated waste solution. Such deposits could prove troublesome in the operation of the plant evaporator, using either the hot air technique or immersion heating. Samples of the deposit have been taken to determine solubility characteristics, and methods of averting the problem are under investigation.

Thermal Consequences of Leaks in Radioactive Waste Storage Tanks

A major criterion in the large scale safe storage of radioactive wastes is the maintenance of ability to dissipate heat generated by radioactive decay. Under all normal and expected conditions of waste storage tank operation heat dissipation is considered in tank design. However, in the event of a leak of radioactive solution into the surrounding soil with or without ion exchange capacity and with relatively high insulating value, local hot spots may occur external to the tank. A document, "Thermal Consequences of Leaks in Radioactive Waste Storage Tanks," HW-80848, containing an analytical solution for determining the consequences of a leak of radioactive heat generating solution into soil has been prepared.

As an example problem, the case of a leak of Purex 241-A supernate containing 5×10^{-4} M cesium (0.18 Btu/hr/gal) is considered. The calculations indicate that for a leak of 5000 gallons the maximum steady-state temperature rises encountered will be 14 F and 62 F at the concrete-soil interface and in the soil at a point 4.2 feet from the concrete liner, respectively. For a 50,000 gallon leak comparable temperature rises will be 29 F and 301 F, the latter at a point 10 feet from the concrete. It is concluded that even relatively large supernate leaks from the Purex waste storage tanks do not constitute a thermal hazard.

A correlation is presented which allows the calculation of maximum steady-state temperatures for hemispherical leaks of solutions containing fission products of varying activity and adsorbability on the soil. The method of solution of this problem is presented and may be useful in solving problems of temperature distributions near tanks, pipes, and other radioactive waste solution sources in the ground.

EQUIPMENT AND MATERIALSGadolinium-Containing 304-L Stainless Steel

Samples of cadmium foil, 304-L stainless steel and gadolinium-containing 304-L stainless steel were tested in the PCTR to compare their neutron capture. Data given below are being analyzed by Critical Mass Physics personnel:

<u>Material</u>	<u>Thickness, mils</u>	<u>Poison, cents</u>
Cd foil	30	22.6
0.7 w/o Gd-304-L	125	20.7
0.3 w/o Gd-304-L	187	17.1
304-L	250	5.7

It was reported previously that weldments of 0.7 w/o Gd-304-L base metal and 308-L welding rod were severely attacked at the base metal-weld metal interface when exposed in the as-welded condition to 65 w/o HNO_3 . A solution anneal following welding does not eliminate this attack. However, a one-hour heat treatment at 1250 F does prevent the preferential attack and also reduces the general attack.

Radiation Induced Discharge in Shielding Window Glass

A sample of Penberthy 3.8 density glass which was being irradiated by mixed fission product irradiation from fuel elements discharged after a total integrated dose of ca. 8×10^6 r. Dose rate at the beginning of the test was 16,000 r/hr, and was 14,600 r/hr at the end of the test. Total integrated dose was about 8×10^6 r. Temperature during the test varied from 10 to 13 C. This glass is typical of that used in the 325-A Building hot cell windows. A sample of Corning 8362 glass, the type selected for 324 Building windows, is currently being exposed to fuel element radiation.

PROCESS CONTROL DEVELOPMENTComputer Control Study

Programming of the GE-412 computer for on-line optimal control of the experimental C-column proceeded. The definition and flowcharting of the scanner control program was completed. This program provides the basis for communication between the process instruments and the computer; it is sufficiently general to handle additional inputs and outputs from the C-column or other similar chemical process. The data logger program has now been defined; this routine provides for automatic print-out of

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date and information in any desired format. A program has been obtained from GE, Phoenix, which enables use of a GE-225 computer for assembling 412 programs. The assembler prepares paper tapes for input to the 412 from a symbolic deck of cards. Considerable difficulty has been encountered in using EDPO's 225, and it appears a logic wiring error is present. Efforts are underway to correct the problem since use of the 225 is essential to timely and efficient assembly of the process control computer programs.

A-C Column Facility

Following completion of instrumenting the A-column, shakedown operations with uranium feed were conducted. Operation of all control instrumentation was satisfactory, although some minor column hardware modifications are needed for fully satisfactory system performance. Connection of A-column instrumentation to the data logger is now in progress.

Preliminary studies of the organic stream uranium photometer have shown that a continuous reading instrument using a three-way beam splitter and three phototubes is not practical. The difficulty lies in the fact that gain characteristics for these phototubes continually change with time and no two tubes seem to change in the same fashion. Since the uranium concentration is calculated from a ratio of the currents generated by these phototubes, it is necessary that the ratio of their gain characteristics remain constant. To overcome this problem, the instrument has been redesigned with a single light path, a single phototube and a rotating filter holding mechanism.

In-Line Plutonium Monitors

Design of the in-line plutonium monitoring system using scintillating glass and X-ray detectors has been completed. The two detectors will be installed in series in the sample line from the Purex plutonium ion exchange unit. Fabrication of the two detectors was also completed. Two charge-sensitive, solid state preamplifiers were built and tested satisfactorily. In addition, a tube-type count rate meter has been checked out for use with the system. Exposure of the scintillating glass to a simulated Purex XAF stream for a total of ten weeks has resulted in no apparent damage to the glass. Installation of the system will proceed on a schedule determined by Purex operation and maintenance requirements.

REACTOR DEVELOPMENT - 04 PROGRAMPLUTONIUM RECYCLE PROGRAMSalt Cycle Process

Electrochemistry of Plutonium in Molten Chloride Solutions - Rapid-scanning polarography is being used to study the mechanism of reduction of Pu^{+++} in LiCl-KCl fused salt solutions. Although quantitative interpretation is difficult, results support the mechanism proposed by Braun and Stromatt, viz.:

Quantitative interpretation is difficult because (1) the two steps of the reduction are separated by only 100 m.v. and reduction occurs at only a slightly lower voltage than that required to reduce an alkali ion, and (2) the equation involving the relationship between peak current and the number of electrons transferred is strictly applicable only at room temperature.

Since the $\text{O}^=$ and OH^- ions in a melt cause spurious peaks to appear in polarograms, a cursory study of the electrochemistry of these ions in LiCl-KCl melts has been made. Oxide ion has a strong oxidation peak at +0.2 volt and a reduction peak at -0.1 volt (Ag/AgCl reference). Hydroxide ion exhibits a strong reduction peak between -1.5 to -1.75 volts. The data indicate that the reaction $\text{OH}^- = \text{O}^= + \text{H}^+$ is measurable at temperatures above 450 C.

U(V) in Fused LiCl-KCl Solutions of U(VI) - U(IV) - Additional spectrophotometric studies of the new uranium species reported last month support the thesis that the species is $\text{UOCl}_3(\text{UOCl}_5)^-$. It has been demonstrated this month that the concentration of the species depends on the square root of the uranyl(VI) concentration, as well as the U(IV) concentration, as required by the equilibrium postulated above.

Chloride Contamination in UO_2 - Additional studies on the effect of impurities on the chloride content of electrodeposited UO_2 from LiCl-KCl melts show that addition of RuO_2 , CeO_2 and Nd_2O_3 to the melt does not increase the chloride content of the deposit.

Engineering Development - The results of analyses received on the three electrolyses reported last month are as follow:

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The first electrolysis run under partition conditions produced a deposit with an overall plutonium decontamination factor of 8.0. Radiochemical analysis showed decontamination factors on an uranium basis for Cs-137 of 500 and 2100 before and after washing, respectively. Decontamination factors on an uranium basis for Zr-Nb-95 and Ru-Rh-106 were 1.5 and 67, respectively. The decontamination factors on a plutonium basis for Y-91 and Ce-144 were 2.5 and 4, respectively.

The second electrolysis run under co-deposition conditions yielded a deposit with 0.34 w/o plutonium or a plutonium enrichment factor of 0.58. The decontamination factor on an uranium basis for Zr-Nb-95 was 0.32. The decontamination factors on plutonium basis for Y-91 and Ce-144 were 2.7 and 8.1, respectively.

The third electrolysis also run under co-deposition conditions yielded a deposit with 0.32 w/o plutonium or a plutonium enrichment factor of 0.65. The decontamination factors on an uranium basis for Cs-137, Zr-95 and Ru-106 were 1330, 0.60 and 263, respectively. The decontamination factors on a plutonium basis for Y-91 and Ce-144 were 2.0 and 5.6, respectively.

A new salt pot fitted with a graphite liner extending just below the salt level was introduced into C-cell of the High Level Radiochemistry Facility and processing of irradiated mixed plutonium-uranium oxides was continued. After dissolution of 21 pounds of mixed oxide it was found that all of the plutonium was in solution indicating that the graphite liner had, as expected, prevented the plutonium from plating out on the quartz at the salt interface. However, it was noted that there was uranium(IV) present indicating the reaction of uranyl chloride with chlorine and carbon. The melt was then electrolyzed for 24.7 hours with a constant cell voltage of 1.70 volts and an initial current of 85 amps to give a 16.25 pound deposit. The melt at 550 C was sparged during the electrolysis with 2.7 l/min O₂ and 0.3 l/min Cl₂. After the electrolysis the melt was observed to contain particulate graphite indicating that severe corrosion of the liner had taken place. It was felt that since the graphite anodes had not corroded during electrolysis that the corrosion of the liner was due to air in-leakage during the high temperature dissolution step. Due to the high carbon content of the melt the plutonium was not oxidized to the hexavalent state and as a result high decontamination factors from plutonium were found in the deposit. The plutonium decontamination factors for the bottom, middle and top of the deposit were 19.1, 18.3 and 12.4, respectively. It is evident that dispersed graphite must be removed from the bath before a co-deposition can be made. It is hoped that this can be done by sparging the melt with chlorine.

Dissolution of PRTR UO_2 - PuO_2 Fuels

Fourteen 1-in. long sections cut from a rod from PRTR Element 5787 (irradiated to 2000 MWD/T) were procured for dissolution studies. This swaged fuel rod was incrementally loaded to an average PuO_2 content of 0.48 percent. Cladding was intact on the sections except for the ends. A black residue (not yet identified but presumed PuO_2) remained after exposure of the core material from one section to boiling 11 M HNO_3 for two hours. Apparently irradiation did not convert the oxide mixture to a mixed oxide from which plutonium oxide is readily dissolved by nitric acid.

Other studies with non-irradiated UO_2 - PuO_2 fuels were continued toward development of a satisfactory dissolution procedure. The most promising procedure developed to date involves:

1. Zirflex process decladding (2 hours in boiling 2.0 M NH_4F -0.17 M NH_4NO_3).
2. Expose core material for two hours at 80 C in 11 M HNO_3 -0.1 M NH_4F -0.0085 M NH_4NO_3 .
3. Make solution from (2) 0.3 M in $\text{Al}(\text{NO}_3)_3$ and boil for 30 minutes to one hour.

Complete dissolution of core material was attained in four separate trials of this procedure.

RADIOACTIVE RESIDUE PROCESSING DEVELOPMENTExamination of Calcined Products

Selected pots from the full-level A-cell calcination runs are being sectioned and examined at Radiometallurgy to observe the physical status and general appearance of the calcines (or melts), to determine the degree of possible segregation of fission products, and to obtain samples for leaching rate measurements. The first pot sectioned was from an ORNL-type pot calcination run with Purex waste, under conditions which were expected to produce a melt. Such was not found to be the case. The solids had not melted and had pulled away from the wall and draft tube, leaving a gap which would result in poor heat transfer and (in the case of a full-size storage container) cause excessively high storage temperatures. Material from the center of the pot was easily dissolved for fission product analysis, but dissolution of that from the outer edge was very difficult. No data are yet available on the degree, if any, of fission product segregation, but rather poor autoradiographs failed to show any segregation.

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It is interesting to note that the above pot calcination run was preceded with a "cold" pilot run in the 8-in. x 10-ft. calciner, using a synthetic feed which was thought to duplicate the full-level feed, and that the cold run gave an excellent melt.

Spray Calcination Test

Difficulty has been encountered with certain feeds in the 18-in. Chemical Development spray calciner due to build-up on the walls or draft tube. Because of the time required to clean up from an abortive run, a simple diagnostic test was desired which would predict the performance of a feed composition. A hot wire test is under development which shows promise of doing this. A nichrome wire is heated electrically to the temperature of the calciner and then dipped into the powder in question (prepared in a laboratory dryer). If the powder adheres to the wire, it will probably foul the calciner. Good correlation has been obtained with the test on powders from earlier 321 Building runs.

Cold Semiworks Spray Calciner

Development studies continued on the 18-in. spray calciner. Operating conditions were evaluated for feeds which incorporated all of the melt or glass forming constituents in the feed solution to produce a powder for melter operation at a low temperature.

Four Redox borax-phosphate runs were made with the Astrosonics #1407 ultrasonic nozzle without a draft tube. Calciner shell temperatures were about 500 C. Heavy, yellow, glassy wall deposits were observed in most of the runs. Feed rates of about 7.5 gallons per hour could be maintained for at least four hours before wall deposits hindered heat transfer to the extent that some internal temperatures fell below 200 C. It appeared that equilibrium build-up of wall deposits had been approached in the four hours and, thus, that a feed rate of somewhat less than 7.5 gallons per hour could be maintained indefinitely. The equilibrium wall deposits could be completely removed by melting out at 600 C followed by thermal shocking of the thin remaining layer with a stream of water.

Purex 1WW run with the ultrasonic nozzle and no draft tube was much coarser than that observed with conventional pneumatic nozzles. The tap density, however, was lower - 0.37 vs. 0.56 g/cc.

One preparation of ORNL lithium phosphate "glass" powder was made with the ultrasonic nozzle, no draft tube and a shell temperature of 650-675 C. Extensive sticking and melting occurred on the calciner walls causing a plugged outlet line after 3-1/2 hours' operation. The wall deposits were largely removed by shocking the 650 C walls with a stream of water.

Although early attempts were unsuccessful with the Astrosonics #1407 ultrasonic nozzle, the modified nozzle has operated satisfactorily in six subsequent runs. Caking and plugging have been eliminated by using four 0.067-in. orifices in place of the original eight 0.040-in. orifices and using saturated rather than superheated steam for atomization.

Calcine Melter

Six melt runs using Redox borosilicate and borophosphate calcine powder feeds were completed in the stainless steel 310 continuous melter. The overflow and the freeze discharge valves functioned satisfactorily during runs with borosilicate melt, although the length of the freeze valve had to be shortened by two inches to eliminate a cold section extending below the freeze valve furnace. Melting of Redox borophosphate calcine feed produced large volumes of foam on the surface of the melt. Feeding cold powder onto the foam did not suppress its formation; however, melting rates between 4.5 and 5 pounds of powder per hour were attained. In an attempt to increase the melting capacity, the outside wall temperature of the melter was raised to 1100 C with corresponding melt temperatures ranging from 900 to 1050 C. Although prior laboratory data had indicated only mild corrosion at 900 C, gross internal corrosion penetrated the melter wall after a melter operating life of 47 hours. Further evaluation of the failure is underway.

Calcined Waste Melt Pot Materials

Crucibles of 310 stainless steel and of Inconel have been exposed to several different calcined waste compositions at fusion temperatures. In every case, the 310 stainless steel showed as good or better corrosion resistance than the Inconel.

Samples of Inconel, 310 stainless steel, Hastelloy X and platinum have been prepared for exposure in the 321 Building melter and in the off-gas stream from the melter in an attempt to obtain corrosion data under actual operating conditions.

Continuous Glass Making

Three further runs (Nos. 2 through 4) were made during the month in the prototype Hot Cell glass experiment equipment. These were aimed at testing equipment (which included a new feed splitter, new coiled tube evaporator, and stainless steel melter bonnet) and at exploring the effects of increased iron concentration and of substitution of aluminum for iron on the quality of the glass produced. Iron was increased from 0.3 M to 0.8 M in an effort to decrease solubility. Although the melting point was increased somewhat (to 1200 C) by this change, the solubility

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(leaching rate in boiling water) also increased by a factor of about two. Substitution of 0.5 M aluminum for the added iron required a yet higher melter temperature (1250 C). Leachability of the aluminum glass has not yet been determined; however, the surface tension of this glass was undesirably high. Further runs will be required for shakedown of new equipment.

Intermediate-Level Waste Treatment

Distribution coefficients were obtained for radioruthenium on organic ion exchange materials before and after electrodialysis of acid condensates. The results indicate that fission product ruthenium is not removed by a simple ion exchange process alone. Ruthenium equilibrium between solution and exchanger was approached as rapidly after the bulk electrolyte was removed by electrodialysis as before. Distribution coefficients for ruthenium on cation, anion and mixed bed resins all increased by factors of 15 to 100 on removal of 90 percent of the bulk electrolyte by electrodialysis. However, the kinetics of ruthenium removal are poor, and to have practical application the rate must be accelerated.

Ruthenium DF's of 100 were obtained by electrodeionization of 3000 column volumes of an acid condensate passed through the mixed resin incorporated into electrodialysis apparatus. The experiment was terminated at this point for inspection and the resin and membranes were observed to be in good condition.

Ruthenium sorption on gibbsite and activated carbon was compared with sorption on olivine and fayalite (the iron end member of the olivine series). Gibbsite and activated carbon were superior sorbents for five acid (0.01 M HNO_3) condensate samples studied. At pH 5 and 9 the distribution coefficients were improved for all the sorbents, with the best distribution coefficients at pH 5.

Columbia River Sediment Studies

Samples of reactor effluent water were examined to determine the chemical and physical forms of some of the radionuclides. The effluent was filtered through 100 mu filters, then passed through anion and cation resins to determine the amounts removed in these forms. The data are tabulated below:

Form	Radioisotope (%)			
	Sc-46	Cr-51	Zn-65	Np-239
On 100 mu filter	43	0.8	6.7	2.1
Cation form	36	1.8	9	50
Anion form	11	97	1.0	42
[redacted] ionic form	10	0.2	0.4	6.7

Samples of river water containing reactor effluent will now be examined to determine the changes in nature of these materials with distance downstream.

CONTAINMENT SYSTEMS EXPERIMENT

Facility and Equipment Procurement

The bid opening dates for the containment vessel and the reactor simulator vessels are April 28 and May 15, respectively.

T Plant modifications are proceeding on a schedule such that the building will be ready for the containment vessel vendor in ample time.

Flowsheets on utilities, wastes, water circulating loops, and ventilation were started. Criteria drawings on auxiliary equipment layout and the waste disposal system were issued for comments.

Test Program Planning

An outline of the CSE program document was prepared and distributed to the Program Review Group for comments. This document will present a more comprehensive test program outline and facility description than was possible in the proposal report (HW-78722). A detailed discussion of the subject of simulation, particularly with respect to fission products, will be included in the document because of the key relation of simulation to the utility, generality, and applicability of the data and correlations to be obtained by the CSE.

Fission-Product Simulation

Planning continued for experiments to select and demonstrate the behavior of simulants for fission products in the Containment Systems Experiment. Stainless steel clad UO_2 specimens containing about 100 grams of UO_2 will be fabricated and irradiated to low burnup to provide the sources of real fission products. Simulated fission product aerosols will be generated by (1) heating synthetic fuel specimens; e.g., unirradiated UO_2 mixed with stable fission-products elements (plus short-lived tracers) vibrationally compacted and clad in stainless steel; and (2) generation of iodine and particulate aerosols by standard techniques in the absence of UO_2 and cladding.

Measurements to enable a comparison of transport behavior will include (1) deposition velocities on various surfaces, (2) degree of reversibility of adsorption on surfaces and condensation nuclei, and (3) retention on cleanup system components.

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Another goal of this experimental program will be to demonstrate prototype aerosol generation and sampling equipment into and out of a steam-air atmosphere at pressures expected in the CSE program, up to 50 psig.

Aerosol Generation

A 50 kw induction heater to enable the melting of UO_2 has been specified for purchase. This will be necessary to inter-compare generation methods. Conversion of an existing 20 kw induction heater to variable output control is underway to provide a heater for use in the interim period. Pending these tests, work will continue on the generation, transport and characterization of elemental iodine for use in CSE. Additional tests of the vaporization and transport of elemental iodine through a heated (150 C) tube, as noted last month, were made. Losses to the 20-ft. long tube were small; less than ~ 1 percent of the iodine passing through the tube was retained. The form of the iodine was not specifically determined in this test; however, because of the good retention in the scrubbers the form is thought to be primarily elemental.

Iodine Sampling

Some additional tests of the effectiveness of aged brominated charcoal for sorption of organic iodine were made. Brominated charcoal (1.5 grams) was exposed to air for 23 days and then to 170 μ g of methyl iodide. The air elution had not impaired the efficiency of the treated charcoal, as it retained all of the iodine activity during the nine day test period. At six days, 23 mg of additional methyl iodide were adsorbed by the charcoal with no loss or exchange of iodine.

These tests indicate a capacity of at least 15 mg CH_3I /gram charcoal and an exposed life of at least 23 days in clean air.

Shock Loading Calculations

Calculations were prepared to determine the dynamic response of the primary and dry well vessels to shock loading.

Basic modes of vibration were determined and dynamic load factors corresponding to triangular pulse loading were calculated.

BIOLOGY AND MEDICINE - 06 PROGRAMTERRESTRIAL ECOLOGY - EARTH SCIENCESHydrology and Geology

A numerical procedure has been developed to solve the problem of imbibition of liquid by a homogeneous porous media of uniform initial moisture content for three geometries of the flow system: horizontal, linear (one-dimensional) flow; horizontal, radial (two-dimensional) flow; and spherical, radial (three-dimensional) flow.

The mathematics of the procedure have been developed and the problem is ready for programming for the digital computer. The existing program for Philip's iterative procedure will require only minor modifications plus the inclusion of three more sub-routines to handle all three geometries.

Although the procedure is applicable only to homogeneous soils and neglects the weight of the liquid (gravity effects), the solutions can be applied to the analysis of waste management problems such as tank and pipeline leaks. Solutions will provide estimates of the advance of the wetting front and the amount of liquid imbibed as functions of time, and also will evaluate the distribution of moisture at any given time.

A new computer program "KPAR" was written to compare statistically permeabilities derived from streamlines numerically generated and those permeabilities determined analytically. Results of the evaluation include differences, sum of differences squared, standard deviation and average difference for each streamline, and standard deviation over all the streamlines examined.

Water level data from wells penetrating confined basalt aquifers on Wahluke Slope were supplied by the U.S. Geological Survey, Ground Water Branch, Tacoma. Those data show a rise in piezometric head ranging from about 80 feet in 10 years at the east end of Wahluke Slope to 10 feet in 6 years north of 100-D and 100-H Areas. The timing, amount and direction of propagation of the head rise indicates that it is attributable to irrigation north and east of the Saddle Mountains. Irrigation there has raised the unconfined ground water table, and through the creation of greater heads is charging the basalt aquifers that become confined southwestward toward the Hanford Works.

Northwest of 100-B Area, toward the west end of Wahluke Slope, rises of about 10 feet in the few similar wells there in the last several years

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suggest recharge of the basalt aquifers through the 90-ft. head rise behind Priest Rapids Dam. U.S. Geological Survey measurements indicate that several thousand second-feet of water are lost from the Columbia River between Wanapum Dam and the tailrace of Priest Rapids Dam. This loss indicates the amount of recharge of the aquifers eastward beneath Wahluke Slope, the only direction in which appreciable recharge is possible because of the geologic structure, stratigraphy and existing ground water table gradients.

Wells 199-H4- and 199-B3-2 were drilled into basalt in 1952 and 1953, respectively, in part to determine the changes in artesian head with time. Both wells showed no obvious water level changes following the time of drilling and for several years afterward. Beginning sometime since 1959, the head in well 199-B3-2 has risen two feet to date. Since 1958 the level in well 199-H4-2 has risen four feet to date. The timing and magnitude of the rise suggest that the basalt aquifers tapped by 199-H4-2 are being recharged from the northeast and that those tapped by 199-B3-2 from the northwest.

The above data reinforce a possible conclusion that some of the noted increased piezometric head in wells north of the 300 Area may be caused by irrigation recharge east of the Columbia River, with head transmission beneath the Columbia River. The direction and rate of movement of Separation Plants' effluents in the contaminated basalt aquifer thus will reflect the relative recharge from the Purex swamp site, the Rattlesnake Hills, the Yakima River and the Columbia Basin Irrigation Project.

RADIOLOGICAL AND HEALTH CHEMISTRY

Uranium Ore Inhalation Study

During the period April 1-10, the Hanford Mobile Whole Body Counter was used to make uranium ore inhalation studies at two mines in southwestern Colorado. A main objective of this study was to use the short-lived radon daughter Bi-214 (28 m) as a tracer of the ore dust and other suspended material in mine atmospheres. In both operating and non-operating mines more than 95 percent of the Bi-214 was attached to sub-micron particles which are mainly diesel fuel exhaust particles. Essentially all of the inhaled submicron particles are retained in the respiratory tract and show no movement over periods of about one hour. About 25 percent of the activity was found in the head, 10 percent in the neck and 65 percent in the chest. Increasing the dust level by two or three orders of magnitude by shoveling ore did not significantly change the respiratory tract deposition of radon daughters and did not increase the fraction of radon daughters on larger particle fractions.

Analytical Procedures

Some procedural modifications were made in a Los Alamos rapid plutonium bioassay procedure which provides results in about two hours. Following nitric acid digestion of the sample, a double, batch ion exchange separation is made. With the ion exchange resin in a column, the plutonium is converted to the chloride complex and eluted with hydrochloric and hydrofluoric acids. The eluate is evaporated and counted. Recoveries of metabolized plutonium from rat urine was 85 percent. The procedure is not completely specific for plutonium, but removes 3 percent, 13 percent and 37 percent, respectively, of uranium, thorium and neptunium.

Bioassay Studies

Twenty samples of human hair were analyzed for Sr-90 during the month with positive results obtained with two samples of Eskimo hair and one of local origin. Small amounts of Cr-51, Mn-54, Co-60, Zn-65 and Cs-137 were also present in the local hair sample. Relatively large amounts of Cs-137 were measured in the Eskimo hair samples. Further measurements are in progress to determine if these measurements are indicative of the body burdens of these radionuclides.

Radiation Chemistry

More experiments were completed which provide information on the mechanism of radiation damage to human red blood cells and the abilities of chemical additives to reduce this damage. The release of hemoglobin into the surrounding media and cellular volume changes were measured as functions of time following irradiation. From an initial cell volume of $85 \mu^3$ the modal volume increased to $160 \mu^3$. At this maximum volume the rate of loss of hemoglobin to the surrounding medium is proceeding at the maximum rate. Following loss of hemoglobin the "ghost" cells appear to return to the normal red cell volume. Three protective agents tested gave protection which paralleled the protection indices determined by the dye competition method developed in this laboratory.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Aerosol Sampling Study

The atmospheric sampler filter holders with a rolled crepe paper support for the membrane filters were shown to collect a sample insignificantly different from a sample obtained using a screen-backed membrane filter. A particle size range of from 3 to 25 microns was used and an air flow rate of 2.7 mph. This observation alleviates the concern that the nature of the flow pattern of the approaching air and the particle

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collection pattern on the filter were being changed materially by the nature of the filter support and backing. Although the intercomparison was made at isokinetic flow through each filter, the evidence is good that only minor differences in performance of the two filters will occur for subisokinetic sampling situations.

A relationship was identified which holds promise for correlating the subisokinetic error in particle sampling with the parameters involved. The divergence between the measured impaction efficiency for a particle when no air is drawn through a filter and the ratio of subisokinetic flow to isokinetic flow at other sampling rates appears to be a normalizing factor within the experimental errors in the data. The impaction efficiency for particles from 3 to 25 microns ranged from about 0.002 to 0.10 for a wind speed of 2.7 mph.

Particle Generation and Characterization Studies

Satisfactory performance of the spinning disc generator was demonstrated after air motor overhaul. Monodisperse aerosols of methylene blue labeled with uranine or rhodamine B were generated.

Fallout Studies

Further measurements of Na-22 in foodstuffs indicate that milk and wheat are the main sources of this radionuclide. Concentrations measured in meat and vegetables were one to two orders of magnitude lower than in milk and wheat. The concentrations of Na-22 in the troposphere measured during 1963 are an order of magnitude higher than the only other reported value, which was measured prior to the 1961 nuclear test, and may indicate fallout origin of some of this radionuclide.

Fallout radionuclides collected in large trays containing a layer of water have been found upon filtration through a 5 μ filter to be largely associated with filterable particles. Essentially 100 percent of the Zr-Mb-95 and Ru-Rh-106 are particulate, with about 50 percent of Mn-54 and Cs-137 being particulate. Other radionuclides are present for which analyses have not been completed.

ISOTOPES DEVELOPMENT - 08 PROGRAM

Radiochemical Calculations

The computer code described last month for calculating radioisotope generation and decay has been revised to obtain output in atoms, grams, curies, beta watts, and kilowatt-hours. Provision has been made to sum results for up to 50 nuclides or nuclide groups in a series of chains

to provide more convenient tabulation of results. Provision has also been made to calculate fission product generation by the constant fission rate method, with reactor power being fed in as megawatts per unit weight of fuel instead of flux. Neutron cross sections are also now used in a modified Westcott notation so that the effect of resonance can be included.

Heat Source Technology

Major efforts during the month were devoted to the preparation of materials for trial pneumatic impaction, to impaction of two new can variations, design of the promethium glove box facility, and several additional acid-side promethium ion-exchange runs.

Materials prepared for "Dynapak" compaction included SrF_2 , SrS and SrH_2 . All are thought to be materials with potentially good burn-up characteristics but with melting points and power densities suitable for thermoelectric generator use. The SrF_2 prepared this month was by reaction of SrCO_3 with $\text{NH}_4\text{F}\cdot\text{HF}$, rather than with HF . The reaction was smoother, less "dusty", and is believed more suitable for hot cell use. None of the materials have yet been compacted. Cesium borate was also prepared by a different technique than previously used - from CsOH and H_3BO_3 rather than from CsCl . Using CsOH it was possible to make a single phase borate containing very nearly 75 weight percent Cs (corresponding to $\text{Cs}_3\text{B}_3\text{O}_6$) whereas the practical limit with CsCl was about 60 percent.

Three cans loaded with SrTiO_3 were compacted to test two new can variations. The resulting compacts were very smooth on the outside and are being examined by non-destructive testing techniques prior to sectioning.

Pneumatic Impaction Source Fabrication

Three prototype heat sources were compacted during the month. Two M-9 top loading Inconel cans with tapered walls were loaded with SrTiO_3 to a green density of 3 g/cc and compacted to a bulk density of 4.5 g/cc (88 percent of theoretical). The M-5 two-section Inconel can with a tapered wall was loaded with SrTiO_3 to a green density of 2.6 g/cc and compacted to 4.3 g/cc (84 percent of theoretical).

The walls of all three cans forged uniformly without folding or fracturing. The walls were still tapered after forging. One of the M-9 cans was hit on the thin end of the can and the other one on the thick end. No difference in the forging of the walls is detectable and deformation of walls appears uniform over entire length of the can.

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C-25

One hundred 310 stainless steel can blanks were received. Twenty cans similar to the M-9 design with straight walls are being fabricated to compare the forging characteristics of 310 stainless steel to Inconel. Lower cost 310 stainless steel will be used as a stand-in container material for impaction process development.

W H Reas

Manager
Chemical Laboratory

WH Reas:cf

BIOLOGY LABORATORY

A. ORGANIZATION AND PERSONNEL

No significant changes occurred.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Reactor Effluent Monitoring

On April 30, 1964 reactor effluent at concentration levels of 0, 2, 4, and 6% was started through eight troughs each holding 500 chinook fry. Effluent through one set of four troughs was being cooled to regular river temperature to study chemical toxicity in the absence of thermal effect. Fish at the start of test weighed about 0.7 g.

Columnaris

Re-infection of trout continues to occur as long as contact is maintained with other infected fish. One of the ten fish under observation completed two cycles of infection within a two-month period and at no time cleared itself completely of the organisms.

Field sampling of fish continues to give a negative response with respect to columnaris associated with the fish gills.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Zinc

Rainbow trout after feeding levels of 0.01, 0.1, and 1.0 $\mu\text{C Zn}^{65}/\text{g}$ fish daily except on weekends for 14 weeks showed no overt response or mortalities. For the highest feeding level the total intake for the 14-week period was 11 mc/fish. The average body burden estimated from fish killed on the fifth and tenth week was 43 and 38% of total administered dose, respectively. The GI tract with the contents removed was high in activity and contributed about 31 and 21% of total body burden at the fifth and tenth week, respectively.

Strontium

Eleven young adult female miniature swine were started on daily feeding of 125 or 625 $\mu\text{C Sr}^{90}/\text{day}$ (or held as controls). All of these animals will be bred. The increasing skeletal burden of Sr^{90} will be followed closely to provide data needed to evaluate the radiation dose. Three animals will be killed at varying stages of gestation to provide additional information on the transfer of Sr^{90} to the developing fetus. Offspring from the animals fed 625 $\mu\text{C Sr}^{90}/\text{day}$ will be closely observed

for evidence of the severe ectopic myelopoiesis previously observed in animals at this level.

Radioiodine

Skin exposures to I^{131} vapor were performed on five miniature swine. The skin area was not washed or covered after exposure. The concentration of I^{131} in the skin decreased during the first day to about 60% of the initially deposited dose (probably vaporization), followed by a more gradual loss with an effective half-life of three days. Maximum thyroid concentrations occurred on the fifth day and varied from 0.2% to 0.5% of the initial skin deposition. The effective half-life of I^{131} in the thyroid was about 13 days. These findings suggest that I_2^{131} movement from skin to blood is slow and very limited in amount.

Five pregnant ewes were administered a single dose of 100 mg of stable iodine 2 to 10 days prior to parturition. Eleven lambs from the above five ewes and four lambs from three control ewes were administered 5 μ c of I^{131} when less than one day old. There was a depression in 24-hour thyroid uptake of I^{131} to 40% control levels in lambs born within three days of stable iodine administration. Lambs born ten days after stable iodine administration showed only a slight depression in thyroid uptake.

Cs¹³⁷ Dosimetry

Preliminary experiments were performed in cooperation with G. W. R. Endres of Radiological Development and Calibrations to evaluate thermoluminescent dosimetry in animals with burdens of Cs^{137} . The thermoluminescent material used in these studies was Li^7F . Estimation of the radiation dose from Cs^{137} is complicated by its decay pattern which includes 0.52 and 1.18 Mev β -rays in addition to an 0.66 Mev γ -ray from the metastable Ba^{137} daughter. Studies with P^{32} beta radiation and Ra^{226} gamma radiation indicate that the use of dosimeters with a wall thickness of 3 mm of teflon should adequately shield the β -radiation from Cs^{137} without appreciably reducing the gamma radiation.

Np²³⁷

To complete Np^{237} liver toxicity studies, three control sheep were administered intravenously a buffered acid solution identical to the one used for previous injections, but without Np^{237} . Liver biopsies and I^{131} -labeled rose bengal tests were performed according to the schedule used with earlier animals in the study. In two of the sheep, results of all rose bengal tests were approximately the same as those of pre-injection tests, while the third animal showed a marked depression in rose bengal clearance on the eighth day post-injection. Pending examination of the biopsy sections, an explanation of the inconsistency may be the introduction of a pathogenic agent during the biopsy procedures or concurrent liver damage from a source unrelated to the biopsy technique or the rose bengal.

This experiment completes the study designed to correlate rose bengal clearance with observations on serial biopsies of liver following the administration of a hepatotoxic (chemical) agent whose presence can be detected autoradiographically.

Plutonium

Desferrioxamine B (DFAB) is a chelating agent which has been effectively employed in combating iron disorders in humans. It was compared with the DTPA for its ability to remove plutonium from rats. Both agents were administered intraperitoneally at 1, 5, and 24 hours post-plutonium injection at a dose level of 0.1 mmole/kg/injection. The DFAB reduced femur deposition as compared with untreated controls by a factor of 30. It was twice as effective in this respect as DTPA. DFAB was less effective than DTPA in reducing liver deposition, and increased deposition in the kidney as compared with controls. DFAB was equally effective when administered intramuscularly. Combined treatment with DFAB intramuscularly and DTPA intraperitoneally gave the best results, with liver deposition reduced to 0.4% of administered plutonium dose, kidney deposition 0.5% and femur 0.13%. Total bone deposition was estimated at 1.5% which compared with 61% estimated for untreated controls. This is probably the most effective plutonium removal yet obtained in prompt treatment of rats and is the first instance of two agents acting in an essentially additive fashion.

Inhalation Studies

Further results from studies of the metabolism of lung tissue indicate that fatty acid synthesis occurs only in the mitochondria fraction of homogenates prepared from rat lung tissue.

Experiments were initiated to determine whether cigarette smoke inhaled by rats altered the rate of clearance of inhaled radioiron particles from the lungs. Preliminary results showed no gross effects following acute administration of cigarette smoke.

Co-Carcinogenesis Studies

A second tumor induction study was initiated to further examine the effect of $Ce^{144}-Pr^{144}$ (20 μ c/rat) on liver tumor induction by dimethylaminocazobenzene (DMAB). Conditions in this experiment should result in a larger tumor incidence, and therefore more clear cut results, than were obtained in the previous study. Changes which should enhance tumor induction include a lowering of the riboflavin and protein content of the diet and the use of male rats rather than the less susceptible females.

Preliminary results from an experiment to determine the effect of DMAB on liver retention of $Ce^{144}-Pr^{144}$ suggest that retention of the radioisotope may be somewhat higher in the DMAB animals. Fecal excretion of $Ce^{144}-Pr^{144}$ during the first 10 days post-injection was decreased 60% in the DMAB rats as compared with untreated controls. Feces were the main excretory route in both groups.

Gastrointestinal Radiation Injury

The ability of the intestine to synthesize mucus has been quantitated by measuring the incorporation of S^{35} in mucin extracted from segments of the duodenum, ileum, cecum, and colon of rats. Abdominal irradiation (1,000 or 1,500 r) decreased such S^{35} uptake by approximately 50% in both the small and large intestine. No such cellular response of the large intestine has previously been associated with acute intestinal radiation exposure.

Secondary Disease

Studies continue on the effectiveness of drugs in suppressing immune response in mice receiving lymphoid tissue as antigen. Melphalan is completely effective in suppressing antibody formation, only when injected prior to antigen. Weekly injections must be continued to prevent antibody formation. Methotrexate, if administered simultaneously with antigen, delays but does not completely eliminate antibody formation.

Effects of X rays on Cichlids

Adult cichlids, Aequidens portalegrensis, exposed to 3,000, 2,000, and 1,000 r suffered 100% mortality by day 18; 38% mortality by day 53; and no mortality by day 53, respectively. Before death the fish often showed radio-dermatitis followed by fungus infection. Anorexia resulted in slight growth depression.

Effects of Water Temperature on Fish

The gross effects of temperature on the metabolism of poikilotherms are well demonstrated; however, the mechanisms that the fish utilize to meet water temperature extremes are not clearly understood. Initial work on temperature acclimation in trout by tissue respirometry indicates active utilization of the pentose shunt in carbohydrate metabolism at low water temperature. Incubation of tissue minces from trout acclimated at 8 C with glucose-1- C^{14} and glucose-6- C^{14} shows maximum evolution of $C^{14}O_2$ at three hours with average C_6/C_1 ratios of 0.21, 0.47 and 1.2 for liver, red muscle and white muscle, respectively. Further work continues to determine if the pentose shunt is maintained in the highly active state indicated by these ratios, when water temperatures are increased.

Microbiology

To more critically identify and isolate the tryptophan pool which serves as a precursor of protein, microsomes were isolated by ultracentrifugation from cells grown in H^3 tryptophan. The results suggest that the tryptophan which sediments with the microsomes is, in fact, the tryptophan which is normally a part of the metabolic pool. The expandable pool contains unbound tryptophan. Further testing with double labelling will be required to confirm this point.

Plant Nutrition

Pea root tips grown in varying concentrations of cesium showed a decreased frequency of cell division with increasing cesium concentration. Measurements of DNA in the nucleii of these pea root tip cells showed a slight rise in DNA content of cesium-treated cells as compared with those cultured in the absence of cesium. There did not, however, appear to be a correlation between DNA content of the nucleii and the frequency of non-dividing cells.

Using standard Neubauer techniques, leaf soil ratios for the uptake of W^{185} were 3.8, 0.8 and 0.07 for Ephrata, Milleville and Cinebar soils. It appears that soil type is of greater influence on the uptake of this anion than on the uptake of most cations previously studied.

Terrestrial Ecology

The radionuclide content of common plants in the Cummins Creek area of the Blue Mountains is greater in plant parts which have been above ground for long periods of time than in the shorter-lived plant parts. Concentrations of Cs^{137} ranged between 16 and 18 pc/g dry tissue in wheat grass and between 6 and 7 pc/g in forbs. The wheat grass appears to be the primary forage for elk in the Blue Mountains during winter. An interesting result of the sampling was the appearance of relatively high levels of Cs^{137} in elk and deer fecal pellets, ranging from 30 to 50 pc/g dry weight.

The annual Canada goose nesting survey was begun slightly later than usual due to Alaskan field work. Paired geese within the Hanford section of the Columbia River numbered 240, the lowest population since comprehensive studies were begun in 1953. The yearly average number of pairs during the past ten years is 290. A total of 150 nests were found during the first survey and this is substantially less than found in previous years.

Cliff swallows arrived and began nesting activities. Four colonies were selected for potential banding and collection studies.

Ring-billed and California gulls have begun colonial formation at Ringold and Coyote Rapids. Nests are being formed but no eggs have been laid.

Alaskan Eskimo Studies

Body burdens of Cs^{137} in adults at Anaktuvuk Pass increased by about 40% since measurements in mid-January. Present levels are about the same as those of last June. Average amounts in 29 adults counted during all three of the periods - June 1963, January 1964, March 1964 - were 630, 410 and 570 nc, respectively. A maximum value of 1250 was measured in an adult male. This same person had the maximum value of 1240 nc during June 1963.

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The marked rise since January appears to be due to the eating of freshly killed animals taken during this present period, whereas during the early fall months the Eskimos had been utilizing animals killed in October and held in permafrost pits. These October animals presumably contained lower concentrations of Cs^{137} because of their having browsed on summer vegetation which is lower in Cs^{137} than the lichens which are eaten more heavily during the winter months.

Special Techniques

Gross autoradiograms of alpha radiation can be obtained with less than 1/500 of the exposure time normally required by NTB plates. This reduction in time is obtained by using ZnS as a scintillator and a fast film to record the position of scintillation.

RC Thompson:es

TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORYI. Speeches Presented

a. Papers Presented at Society Meetings and Symposia

None

b. Seminars (Off-Site and Local)

Dean, J. M. Physiological effects of temperature acclimation in Crustacea. Biology Department, DePaul Univ., Chicago. April 18, 1964.

c. Seminars (Biology)

Bair, W. J. Experiments on inhaled radioactive particles and cigarette smoke. April 8, 1964.

Hart, J. Sanford, National Research Council, Canada. Energetics of temperature regulation of birds and small mammals. April 10, 1964.

Cline, J. F. Cesium toxicity in pea roots. April 15, 1964.

Wood, D. H. Neptunium toxicity in sheep. April 15, 1964.

Buschbom, R. Use of computers in Biology. April 22, 1964.

Tombropoulos, E. G. Lipid metabolism of lung tissue. April 29, 1964.

d. Miscellaneous

Bustad, L. K. Guest speaker at National Secretaries Banquet, Kennewick, Washington. April 22, 1964.

II. Articles Published

a. HW Documents

None

b. Open Literature

McClellan, R. O. 1964. Calcium-strontium discrimination in miniature pigs as related to age. Nature 202:104-106.

McClellan, R. O. and L. K. Bustad. 1964. Metabolism of some SNAP radionuclides in miniature swine. (Abstr.) Aerospace Medicine 35:276.

III. Visits and Visitors

a. Visits to Hanford

Three scientists from Japan, accompanied by M. Kegan, AEC, toured facilities with R. F. Palmer. April 3, 1964.

Nine graduate students from Washington State University toured Biology. April 8, 1964.

University of Washington students and Professor toured Biology. April 8, 1964.

Dr. J. Sanford Hart of the National Research Council, Canada, presented a seminar and toured facilities with J. M. Dean. April 10, 1964.

R. C. Erickson, Univ. of Washington graduate student discussed research with R. E. Nakatani. April 13, 1964.

Admiral J. A. Waters, R. E. Georgi, LM Brenner, and M. L. Nash of the AEC Security Division, Washington, D. C. toured Biology. April 14, 1964.

J. L. Liverman and J. E. Whitnah, discussed research program and facilities with H. A. Kornberg. Both are from the Division of Biology and Medicine, Washington, D.C. AEC office. April 14, 1964.

Ten students and two teachers from Lower Columbia Basin College, Vancouver, Washington, toured Biology with L. A. Temple. April 17, 1964.

The Tri-City Ministerial Association (about 35 total) toured the Biology facilities. April 21, 1964.

Two Dow Chemical representatives toured the Animal Farm on April 21.

Dr. R. Daubenmire, Washington State University, Pullman, checked plots at Rattlesnake Springs with W. Rickard. April 21, 1964.

J. Stout, G. Stone, J. Willey from Walla Walla College consulted on a statistical problem in biology with L. L. Eberhardt, April 22, 1964.

Dr. R. T. Overman, Director of ORINS, Oak Ridge, Tennessee, toured facilities and discussed program with H. A. Kornberg. April 24, 1964.

Eve Stevens, reporter for the Home and Gardens Section of the Oregonian, from Portland, toured Biology facilities on April 24, 1964, particularly interested in plant research.

T. J. Mahoney, pathologist from Kadlec Hospital, attended seminars and discussed research with M. F. Sullivan on April 22 and April 29, 1964.

S. L. Hansard, Director, Radioisotope Lab, Louisiana State University, toured facilities and discussed research with L. K. Bustad, et al., on April 22.

Twelve Explorer Scouts from Prineville, Oregon toured Biology with L. A. Temple on April 28, 1964.

Mr. Day of the University of Washington toured facilities and discussed research with J. M. Dean, et al., April 29.

Dr. Lauren Donaldson of the University of Washington discussed research with R. E. Nakatani on April 29, 1964.

b. Visits Off-Site

3/23 - 4/9 - W. C. Hanson conducted Eskimo counting studies in Alaska.

4/4 - 15 - R. F. Keough attended ACS Meeting in Philadelphia and discussed research with personnel at HASL in New York and Brookhaven National Laboratory.

4/11-15 - R. F. Palmer discussed research at HASL and Brookhaven in New York.

b. Visits Off-Site (continued)

- 4/6-10 - L. L. Eberhardt collected fallout samples in the Blue Mtns.
- 4/10 - W. H. Rickard collected fallout samples in the Blue Mtns.
- 4/7-8 - W. A. Ragan attended a veterinarian meeting in Pullman.
- 4/10-16 - L. K. Bustad attended the Federation Meetings in Chicago.
- 4/7-18 - E. W. Uyeki attended the Am. Soc. for Cancer Research Meeting and the Federation Meetings in Chicago.
- 4/4-10 - B. O. Stuart collected air samples from mines and monitored miners by whole-body count at Naturita, Colorado.
- 4/12-17 - E. G. Tombropoulos attended the Federation Meetings in Chicago.
- 4/15-19 - J. M. Dean attended the Fed. Meetings and discussed research at DePaul University in Chicago.
- 4/18-21 - R. E. Nakatani attended Intl. Pacific Salmon Fisheries Commission Meeting in West Minister, B.C.
- 4/17 - L. E. Haverfield picked up insects at WSU, Pullman.
- 4/23-24 - F. P. Hungate attended the 3rd Plowshare Symp. at Davis, Calif. and discussed research at Livermore, Calif.
- 4/23-24 - L. L. Eberhardt and D. G. Watson collected fallout samples at Packwood, Washington.

IV. Achievements

None

V. Honors and Recognitions

None

VI. Professional Group or Organization Assignments

None

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APPLIED MATHEMATICS OPERATION
MONTHLY REPORT - APRIL, 1964

ORGANIZATION AND PERSONNEL

W. A. Yungen, who has been on the Technical Graduate Program, joined the Applied Mathematics Operation on permanent assignment as a Jr. Scientist, effective April 1, 1964.

ACTIVITIES FOR OTHER HAPO COMPONENTS

N-Reactor Department

Consultation services were provided on methods of constructing mathematical models for certain phases of the fuels production and scheduling process.

The results of a covariance analysis of fuel data relative to welding speed, weld penetration, weld overhang, and internal clad melting were interpreted and reported.

Consulting services were provided to NRD personnel seeking to learn the proper use of FORTRAN language.

Irradiation Processing Department

Production Fuels

The patterning characteristics of a 6x6 Latin square were used to develop a production test which compares different fuel element treatments. The main effects of interest are the differences between the incremental dimensional characteristics for normal and small HDS core diameters and between high and low sizing pressures. The tube arrangements allow power and ingot effects to be subtracted out resulting in a test of improved precision.

A test is being analyzed to determine the best conditions for the can-sleeve preheat time together with the total submerge time in the canning of AlSi fuel elements. The test, using a 4^2 factorial design uses the external and internal UE-1 and UT-4 values as yield variables.

Assistance was given in the design of a test using depleted uranium in which the cores are to be classified according to canning line and hydrogen content. A pull and examination of a sample of fuel elements short of goal will be used to decide the performance procedure for the other fuel elements.

A study of the dimensional behavior of fuel elements due to out-of-reactor thermal effects alone indicated internal diameter and length increases but these were hard to relate to fuel element performance in reactors.

Assistance was given in some of the statistical aspects of a program for controlling the safety climate by periodic inspections of work areas for unsafe practices.

Results from an integral-spire core test based on bond integrity measurements have been analyzed to assess the effects of the preheat times, the incremental temperatures and the different vibrators. The results strongly indicate the superiority of a particular vibrator.

Discussions were held on methods of constructing optimal visitor orientation scheduling through the fuels production facilities.

Research and Engineering

Some previous work on rupture rate estimates and confidence limits is being updated in light of present rupture rate information.

The analysis of the results of a test to assess the accuracy of temperature monitoring thermocouples was reviewed.

Work is continuing on the analysis of PT-572 to characterize weight loss and dimensional distortion effects according to the support height of self-supported fuel elements.

Financial

Information Systems work continues as scheduled. Extensions are being made to cover preventive maintenance activities. Problems of computer availability have been raised. Conferences continue on this subject. The matter may be resolved over the next four-to-six weeks.

Chemical Processing Department

The required demonstration of weapon component chemical and isotopic specifications was made for both "A" and "C" line components for the first quarter of CY-1964.

A meeting of representatives from all sites was attended during April 12-14. There was a review of data obtained in the Plutonium Sample Exchange Program, and, based on a LASL recommendation, a uniform reporting system was adopted for individual measurements.

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Consultation services were provided to discuss mathematical models of inventory control processes.

Contract and Accounting Operation

Technical consultation was provided on possible methods of revising the current linear programming EDPM program so as to eliminate some of its present shortcomings. Investigation of this subject is under way.

Assistance was given in the interpretation of results obtained using both the customary and minimum variance BPID estimates.

Business Planning and Transfer Operation

Work on Tri-City Area economic problems was continued. Estimates were made of Area Retail Sales potential during 1963 as a basis for projecting such potential through the period of employment adjustment that lies ahead. The study of air transport service conditions from Pasco to normal HAPO destinations was carried forward to the point of a final report. This report will be available by mid-May.

A model of Hanford material and information flows has been developed for use as a guide to the reassignment of various production activities among new contractors. When the preliminary report is approved it will be circulated for comment among the production departments.

ACTIVITIES WITHIN HANFORD LABORATORIES

2000 Program

C-column Computer Control

Discussions were held with Chemical Laboratory consultant Dr. Rutherford Aris concerning numerical problems associated with the solution of the nonlinear system of differential equations which explain mass transfer in the C-column and statistical problems associated with analysis of twin filter photometer data. Mathematical calculations are in progress in line with his suggestions.

The first draft was finished of a GE-412 computer subprogram to decode rotary switch inputs from the manual control panel. This program will be used to start and stop the C-column, change scan point limits, and initiate various other activities.

Ground Water Models

Progress continues on the mathematical models which have been devised to study soil permeability. Two of the six proposed experiments (EDPM Programs based upon the so-called shoe-box model) have been completed.

3000 Program

Numerical Control of δ -w Lathes

Recent tests on the prototype numerically controlled δ -w lathe indicate that the error producing subroutine in the EDPM program CUPID has been successfully corrected. Pulse counts on the magnetic control tapes are now known to be correct and their spacial distribution appears to be satisfactory. Test pieces are soon to be fabricated and gauged.

Numerical Control of Rotary Contour Gauges

Following the successful performance of the numerically controlled prototype rotary contour gauge, plans are under way to convert a production gauge to this mode of control. Control tapes are being prepared with the EDPM program SPIG in anticipation of tests for this equipment.

Power Roll Forming

Consultation services were given on geometric problems which arose during the fabrication of graphite molds used to form Floturn blanks.

4000 Program

Sonic Transmission in Visco-elastic Media

An EDPM program was completed which computes the Fourier expansion of a function which is specified by a set of equally spaced data. This data is then transformed so as to yield the spectral distribution of the data.

Particle Packing Studies

A table specifying particle sizes and proportionate mixes required to attain high density vibratory packing of annular cylinders has been compiled. If ρ represents the ratio of inner-to-outer diameter of the tube to be packed, the table lists three size fractions and the weight percentage of each necessary to yield a maximal density.

Weight percentages and screen sizes were calculated for use in fabricating an "inverted cluster" fuel element. The customer reported a satisfactory element on the first attempt.

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Radiation Damage to Fissionable Materials

The FORTRAN language BUBBLES Program for the quantitative analysis of metallographic data collected on the Ziess particle size analyzer was completely debugged. Documentation of the program is now in progress.

Other

Mathematical consultation on the derivation, validity, and use of leak-rate formulas for the PRTR containment vessel was provided.

Work is continuing on the estimation of the total amounts of various isotopes in a channel based on curies/foot estimates along the channel. The analysis consists of fitting an appropriate model and integrating between limits. An aspect of the analysis is the comparison of different models.

A nested components of variance model and analysis was used to assess the magnitude of error associated with different methods of estimating the alloy content of a part of a casting. It was concluded that the errors associated with analysis and dissolution were too large to detect any heterogeneity which might exist in the casting and that the best estimate of content was obtained from the original constituent amounts.

A FORTRAN language program was specified and is being written to unfold an observed two-dimensional particle size distribution into its three dimensional counterpart. The program uses the classical methods developed by Saltykov. A statistical analysis was done of the error structure of the unfolding formulas and as a result standard deviation estimates are being coded into the program. The program will be available as a subroutine using the same input format as the BUBBLES Program.

5000 Program

Computation and Statistical Analysis

Work continued on the calculation of the power function of the Poisson index statistic used to check stability of counting instruments. Power calculations were completed for the alternative of a linear drift during the counting periods assuming a fixed denominator in the index and a normal approximation for the Poisson variables. A program is now being debugged to perform the power function calculations assuming a random denominator in the index and a normal approximation for the Poisson variables.

Work was started on deriving the two-dimensional sufficient statistic for the joint estimation of time zero disintegration rate and the individual atom

disintegration rate based on the random observation of the times between successive disintegrations of a radioactive material. This joint distribution will be used to derive a procedure for judging whether or not a sequence of disintegration times indicates the presence of a short-lived activity or are merely indicative of a constant background source.

Radiochemical Analysis

Specifications of the BLU file to EDPO personnel was completed during the month. Programming of the BLU system will start soon. A program was completed which will translate 2000 existing IBM cards of program data into cards compatible with the BLU system. Debugging of this program has just begun.

Planning is under way on a new system to analyze multidimensional gamma-ray spectrometer data. The new system will perform channel summing, background subtraction, and spectra filing in addition to the spectra plotting which was the only function of the previous program MUD.

Actinide Element Research

Work on crystal indexing problems and programs continued.

6000 Program

Ecological Models

Work began on a program to study mathematical models used in predicting atmospheric transport of small particles.

An essay on a systems approach to economic-biological interactions, postnuclear attack, was prepared and presented. Further work and discussions took place during a visit in Washington. The results are to be incorporated in an expanded version of the initial essay.

Biology

Completed the statistical analyses to investigate the zinc-cadmium relationship in various rat tissues. Comparisons of percent dose retained of Zn^{65} with Cd^{115m} tracer and Cd^{115m} with Zn^{65} tracer in several tissues were made for different dose levels of Zn^{65} and Cd^{115m} .

The statistical analyses to investigate the effect of incubation temperature and rearing temperature on the growth of cichlids was completed. Three incubations and three rearing temperatures were considered in this study.

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Other

Work is continuing on the problem of obtaining acceptable fits of density distributions with observed particle size distributions. The work testing whether there is a significant interaction between sampling locations and frequency distributions has been finished.

General

A closed form solution has been obtained for steady state heat diffusion in a long cylindrical body which possesses nonuniform and nonlinear thermal sources. The problem arose while studying reactions in an ion-exchange column in connection with the investigation of possible causes of the recent Redox fire.

An EDPM program was written which utilizes a two-dimensional search technique to minimize a complex function associated with cohesive energy in metal quenching tests.

An EDPM program was written which performs an iterative least squares procedure associated with a functional relation between stress and strain in metallic media.

An improved model was obtained for propagated errors in results obtained from a mass spectrometer. The model was within modular form to facilitate programming for the 7090.

Carl A. Bennett

Manager
Applied Mathematics

CA Bennett:dgl

PROGRAMMINGREACTOR PROGRAM - O4 PROGRAMPLUTONIUM UTILIZATION PROGRAMOvermoderated Reactor Simulations

The variation of the infinite multiplication factor (k_{∞}) in the simulation of a water-cooled overmoderated graphite reactor is under study. Due to the overmoderation, when water is removed from the lattice, k_{∞} appears to increase in all cases, except in fuels involving thorium. The primary cause of the increase in k_{∞} is the relative increase in the thermal utilization that results when the water is removed from the lattice. In the case of the thorium-uranium fuel, thermal utilization increases by a lesser percentage and the resonance escape probability decreases by a greater percentage than with U^{238} fuels. However, since many competing factors can contribute to ameliorate such effects, more detailed computations will be required to firm up predictions of the effect of water loss on reactivity.

Uranium Conservation Studies (YUKON)

The YUKON code has been revised to allow simultaneous considerations of up to seven distinct reactor types and is now being debugged. Each type of reactor can be introduced into any hypothesized future economy independent of the other types. Variable reactor lifetimes are considered so that a given reactor type can be removed from the economy as it is replaced by another type. For example, after breeders become economical and self-sustaining within a hypothesized economy, the long-term ratio of breeders to converters may vary as competing economic factors change. Variations in this and similar interactions can be handled by the revised YUKON code.

ALTHAEA

The cross section libraries for MELEAGER-ALTHAEA are being updated in view of additional references and data accumulated in the past year. While this updating is proceeding, the use of MELEAGER is being curtailed. During this updating period, the capacity of MELEAGER-ALTHAEA to reflect greater amounts of integrated cross section information is being increased. Normally, MELEAGER-ALTHAEA uses what appears to be an inadequate two-group cross section, consisting of a thermal and epithermal group (the epithermal group to reflect the major resonance effects). To

allow over-all resonance integral data to develop meaningful eipthermal cross sections, cross-shielding factors for overlapping resonances have been developed. Cross shielding appears to be important only to the low-lying resonances, in particular, those of Pu^{239} , Pu^{241} , Am^{241} and Np^{237} . However, there may be some significant cross shielding between some of the high thorium and U^{238} resonances. The power of the 1 ev Pu^{240} resonance to cast "shadows" on the other epithermal resonances is uncertain for the general case and must be treated in a very special fashion with codes far more detailed than MELEAGER-ALTHAEA. Doppler broadening relationships are being added to the MELEAGER-ALTHAEA code system. In the future, two temperatures must be supplied as input (1) the moderated temperature and (2) the effective fuel temperature. MELEAGER-ALTHAEA employs the moderated temperature with the degree of blackness to compute a neutron temperature which is appropriately changed with burn up. The fuel temperature is a constant throughout a run even though the heat distribution within a fuel element may change. To reflect fuel temperature changes, the code would have to be stopped and re-initialized. An alternate feature will be provided which will print the fuel and moderator temperature coefficients at the beginning and end of each MELEAGER-ALTHAEA run. If these coefficients are desired during burn up, a special option must be employed. By utilizing the latter alternative, the temperature coefficient can be examined during burn up - which is likely necessary only when gross changes in fuel makeup occur, such as the early stages of irradiation of a slightly-enriched uranium fuel in a high neutron economy reactor.

The basic cross section data are being processed from the RBU Library tape via the GANDS program to obtain a complete file of cross section data from -100°C to 1300°C for all isotopes of interest. Since ALTHAEA cross sections are currently limited to a fourth order polynomial fit versus temperature, adequate accuracy is not obtained over the full temperature range, so, two files are being prepared. Error comparisons are being made to determine the ranges for the low temperature file and the high temperature file which provide the best data over the entire range, as well as mutually consistent data at the temperatures common to both files. Preliminary functions for (Doppler broadening) corrections have been obtained using resonance parameters obtained from the RBU Library. The compiled data will be reviewed by Applied Physics personnel before the present data are replaced.

Comparisons of hand calculated seed-blanket fuel costs with those obtained from the QUICK code are continuing.

MINIMIZER

The MINIMIZER code has been re-written to fit, mathematically, cost components as a function of exposure instead of enrichment. This was instigated by interest in assessing the potential of metallic uranium fuels which are often durability limited at fuel exposures substantially less than the exposure yielding mathematical minimum costs. The code is also instructed to prepare a table of fuel costs as a function of exposure. This table can then be inspected to reveal the economic incentive for improving the durability of metallic fuels. A special test showed that it is no more accurate to determine minimum cost points as a function of exposure than as a function of enrichment. On this basis, either version of MINIMIZER can be selected to determine minimum costs.

Nuclear Safety Activities

Information was presented to a Subcommittee of the Advisory Committee on Reactor Safeguards on the N Reactor. Results of tests to date, consequences of postulated accidents and the waste management program were the primary topics. A meeting with the full Committee is scheduled for May 7 and 8.

The AEC approved Supplement 4 to the Safeguards Analysis for PRCF extending the range of fuels and lattices for light-water moderated experiments.

J. W. Woodfield
Manager-Programming

FW Woodfield:ch

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF APRIL 1964

A. ORGANIZATION AND PERSONNEL

J. F. Honstead joined Environmental Studies and Evaluations as a Technical Specialist on April 27. E. C. Watson transferred to Programming from ES&EO on April 1. T. A. Fleming, a temporary employee with CDS&R, accepted a permanent position with Radiological Analysis Unit, Analytical Laboratories, effective April 20.

B. ACTIVITIES

Occupational Exposure Experience

There was one new plutonium case confirmed by special bioassay analysis during the month. This case resulted from inhalation following a glove rupture in the Weapons Manufacturing Building (234-5) and was estimated to be 4% of the maximum permissible body burden (MPBB, plutonium with bone as reference, is 0.04 μ Ci).

The total number of individuals who have received internal plutonium depositions at Hanford is 343, of which 247 are currently employed.

In April there were five incidents involving six employees which required special bioassay sampling for plutonium analysis. The following is a brief description of the more significant incidents:

A CPD laboratory employee received a possibly contaminated injury while working in the Weapons Manufacturing Building (234-5). The middle finger of the right hand was cut when a glass test tube, contaminated to 40,000 d/m of plutonium chloride, broke. A tourniquet was applied and the wound flushed. An examination, using the plutonium wound counter, showed less than the detection limit of 1×10^{-4} μ Ci.

A CPD millwright received skin contamination to 40,000 d/m and nasal contamination indicated by smears, of 200 d/m when a hood glove ruptured while the employee was working in a hood in the Weapons Manufacturing Building (234-5). All contamination was easily removed.

A CPD operator received an acid burn on the right side of his face when an overhead hood glove in the Weapons Manufacturing Building (234-5) dripped plutonium nitrate solution. The contaminated skin was immediately flushed with water, reducing the contamination to 20,000 d/m. Tongue and nasal smears were negative. Final skin decontamination was successful.

In addition to the incidents involving plutonium, there were four incidents involving ten employees that required evaluation for possible intake of other radioisotopes. The only incident resulting in significant depositions occurred on April 27, 1964 when four

HL employees were performing experimental work with sodium iodide in the Cold Semi-Works Facility (321-A). A sample of sodium iodide was being dried in an oven when the spread of contamination occurred. Examination at the Whole Body Counting Facility, using the thyroid crystal, indicated depositions of I-131 for the four ranging from 7 to 14 percent of the maximum permissible body burden (MPBB, I-131 based on the thyroid = 700 μ Ci).

There were three incidents of special interest involving exposure of employees from external sources. These are summarized below:

An IPD employee received particulate skin contamination on the back of his neck while working on the rear face of the 105-F Reactor Building. An uncorrected dose rate measurement of 1 rad/hour was obtained using a CP meter. The dose to a localized skin area was calculated to be 10 rads, assuming a ten minute exposure period.

An IPD maintenance foreman received an exposure of 1.5 rems whole body penetrating radiation during April as measured by his beta gamma film badge dosimeter. This exceeds the Hanford Operational Control limit of 1 rem per badge period. The employee's accumulated exposure, including the above, does not exceed any permissible limit.

A CPD monitor wore a pair of coveralls for an estimated one hour before a spot of contamination was detected on them. The results of a film study indicate the maximum dose to a localized area of the skin on the employee's leg was 40 rads.

Extensive repair and maintenance was performed on PRTR shim rods in the Fuel Element Examination Facility (FEEF). Full utilization of the FEEF was not possible because of repairs and temporary lead shielding was installed for the shim rod work. Personnel dose rates behind the portable shielding varied from 10 to 500 mR/hour. The maximum contact dose rate on the shim rods was 150 rads/hour at 1 inch and flash dose rates up to 100 rads/hour were measured to the hands.

Environmental Experience

Through a valving error, waste water from the 100-K Reactor was directed to the original waste water crib rather than the K trench. For an undetermined period between noon, April 12, and 10:00 a.m. April 13, reactor effluent water overflowed from the crib to the river shoreline at a rate of approximately 10,000 gpm. It was determined that the concentrations of radionuclides in the drinking water at downstream reactors could not have resulted in exposures exceeding a few mrems. The effect further downstream was indistinguishable from normal releases. Ground contamination at levels up to 25,000 c/m on a GM meter was found in an area

about fifty feet wide between the crib and the river. There is some grass cover in this area which probably will be effective in preventing wind spread of the contamination.

Ten thousand pounds of Vertan 690, a proprietary compound based on EDTA, were discharged to the river following a cleanout of the N reactor graphite cooling loop. Samples from the mixed stream entering the N outfall line to the river showed a maximum concentration of 30 ppm of Vertan 690, as compared to a recommended control limit of 50 ppm. The maximum concentration in the river is estimated to have been about 5 ppm; samples at the 181-D river pump house showed < 2 ppm as expected.

Concentrations of particulate fallout materials in the air of the Pacific Northwest averaged $1.5 \text{ pc } \beta/\text{m}^3$ air during April, compared to the averages of 1.0 noted during March and 0.8 noted during February. The gradual increase from the low values noted during the winter months is expected because of the spring influx of long-lived fallout materials.

No detectable environmental contamination resulted from the volatilization of ~6 millicuries of I-131 within the 321-A Building on April 27, 1964. The contamination appeared to be confined to the interior of the building. The building is vented by a short stack ending at roof level.

Studies and Improvements

On the basis of laboratory recommendations, the measurement of P-32 in fish flesh was changed from beta counting methods to gamma counting of the bremsstrahlung. Greater accuracy at reduced cost is expected to result from the direct count which avoids wet chemistry procedures.

Occasional checks by the old procedure will continue until the accuracy of the new procedure is satisfactorily demonstrated.

A report was prepared at the request of the Commission comparing the actual cost of radioactive waste burial for the first six months of FY-1964 with a conceptual situation where the number of active burial grounds would be reduced to one in all reactor areas and one each in 200-E and 200-W. It was estimated that for this period the operating costs would have increased by 15 percent, and that there would have been an \$80,000 non-recurring expense.

Film badge dosimeters were assigned to contractor personnel working on the Washington Public Power Supply System Facility in 100-N Area. A total of 21 dosimeters was assigned to Washington Public Power Supply System, Burns and Roe, Inc., and Combustion Engineering personnel. These will be processed, and the results maintained in the current exposure records system on a full cost recovery basis for the AEC.

A new pocket dosimeter read-out system was placed into service. The new system has a lighted push-button keyboard for payroll entry, an audible alarm for pencil results that exceed control settings, and a self-cleaning socket. A higher charging voltage of 130 volts provides for a measurement of 300 mR at complete discharge.

Radiological design features which were approved for the new Radio-Surgery Facility included: (1) a floor exhaust system, (2) provision of a floor to storage sump drain in the wash room, (3) tile floor covering for all potential radiation zones, (4) installation of recessed receptacles in floor for temporary storage of waste material, and (5) inclusion of an ambulance enclosure at the patient unloading entrance. Design criteria were defined for a stack 15 feet above the roof including a provision for monitoring the related effluent.

An HL criticality alarm, "Linear Nuclear Accident Alarm", has been tested with the Flash X-ray generator for response. The alarm was calibrated so that 5 R/hour of radium gamma gave full scale deflection on the 20 ma meter. Less than 1 second lag time between radiation pulse and alarm was noted. An accumulated dose of 1.2 R was measured during the lag time. Linearity of the alarm was demonstrated using the radium source.

Internal Dosimetry personnel participated in a study initiated by the Radiological Physics Operation of lung burdens of radon and its daughter products under field conditions in operating and non-operating uranium mines near Naturita, Colorado. The object of the study was to determine the amount of deposition and the relative distribution of radon daughters within the tracheobronchial region after breathing known concentrations of radon. Lung depositions were determined using the Hanford Mobile Whole Body Counter. Cooperating in this study were the Union Carbide Nuclear Company, who provided use of two of their uranium mines, and the Colorado State Bureau of Mines who provided an inspector to accompany participants in this study as a safety precaution.

A method based on the surface dose rate and on the activity ratio of Th-228:Th-232 has been derived to provide a means to calculate the lapsed time following the purification of thorium. The validity of the time equation has been verified on two samples whose date of purification was known.

A study proposal was drafted to define the relationship between background concentration of Pu-239 aerosol and the incidence of routine positive body burdens of Pu-239 in the workers at the Weapons Fabrication Facility. The draft proposal defines a long-term aerosol study, including air sampling equipment, analysis of filters, and evaluation of results.

Research StudiesEffect of Reactor Effluent on the Quality of Columbia River Water (02)

Routine temperature and meteorological data collection and reduction continued. Dye studies were still halted pending the availability of the new boat. The portable meteorological station was reassembled and again placed on test in K Area.

Nuclear Facilities Monitoring Guide (04)

Work on the Guide continued with primary attention given to describing the requirements of liquid effluent sampling, particularly the type of sampling and size and frequency of samples required. An attempt was made to use some Hanford environmental monitoring data as an estimate of the population to be sampled. The results were informative, but not directly useful. Actual operating data from power reactors is needed.

C. RELATIONS

There was one minor injury and no security violations during the month.

One new suggestion was received. Eight were evaluated, two adopted, and six rejected. Five suggestions are still outstanding.

Radiation orientations were presented to 50 personnel from the Fire Department, 314 Building, maintenance personnel, and technicians in the 306 and 309 Buildings.

Field training for AEC Couriers was continued with six couriers receiving instrument familiarization and radiation monitoring technique training in the 308 and 325 Buildings.

Radiation orientations were presented to Maintenance and Critical Mass Physics personnel.

The final three sessions of the eighth round in the Radiation Monitoring Refresher Course were held. Total attendance was 37, of whom one was a CPD employee. The sessions in the eighth round dealt with air sampling. The first group of Radiation Monitors was given the final examination for the course. Eight men were present, all from Hanford Laboratories.

Sessions 10, 11 and 12 of the Radiation Protection Training Course for Exempt Personnel were held. The postponed half of session 9 was also presented. This completes the course. Average attendance was 25. A questionnaire will be sent to each of the attendees concerning course content, presentations and schedule.

A forty hour training course entitled "Radiological Monitoring for Instructors" (Civil Defense) was attended by K. R. Heid.

D. SIGNIFICANT REPORTS

- HW-77274 - "Radioactive Burial Ground Consolidation Cost Study"
Sup. 1 By G. E. Backman, April 1964.
- HW-80892-3 - "Radiological Status of the Hanford Environs for March,
1964", by R. F. Foster, April 15, 1964.
- HW-78262 - "Ringold - A Hanford Environmental Study", by I. C.
Revision Nelson and R. F. Foster.
- HW-SA-3434 - "Plutonium Aerosol Particle Size Distribution in Room
Air", by B. V. Andersen, dated March 30, 1964.

External Exposure Above Permissible LimitsApril 1964

Whole Body Penetrating	0	0
Whole Body Skin	1	1
Extremity	1	1

Hanford Pocket Dosimeters

Dosimeters Processed	3939	14,993
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Hanford Beta-Gamma Film Badge Dosimeters

Film Processed	9806	39,478
Results - 100-300 mrad	174	644
Results - 300-500 mrad	23	80
Results - Over 500 mrad	12	25
Lost Results	0	37
Average Dose Per Film Packet - mrad (ow)	8.5	7.4
- mr (s)	7.4	42.52

Hanford Neutron Film Badge DosimetersSlow Neutron

Film Processed	445	3555
Results - 50-100 mrem	0	24
Results - 100-300 mrem	0	29
Results - Over 300 mrem	0	5
Lost Results	0	5

Fast Neutron

Film Processed	226	1767
Results - 50-100 mrem	2	34
Results - 100-300 mrem	53	164
Results - Over 300 mrem	1	3
Lost Results	0	3

Hand Checks

Checks Taken - Alpha	36,591	145,198
- Beta-Gamma	55,511	226,150

Skin Contamination

Plutonium	17	70
Fission Products	41	215
Uranium	1	6
Tritium	0	0
Thorium	0	0

Whole Body Counter

<u>Subject</u>	<u>Number of Examinations</u>			
	<u>747-A WBC</u>	<u>1964</u>	<u>Mobile WBC</u>	<u>1964</u>
<u>GE Employees</u>				
Regular	15	168	31	460
Incident Cases	7	39	1	4
Terminations	12	29	0	0
New Hires	16	92	0	0
Special Studies	12	27	0	10
<u>Non-Employees</u>	<u>4</u>	<u>6</u>	<u>0</u>	<u>0</u>
Total	66	361	32	474

Bioassay

<u>Analysis</u>	<u>Current Reporting Limit</u>	<u>Results Above Reporting Limit</u>		<u>Samples Assayed</u>	
		<u>Apr.</u>	<u>1964</u>	<u>Apr.</u>	<u>1964</u>
Plutonium	2.2x10 ⁻⁸ µc/sample	74	344	678	2305
Fission Prod.	3.1x10 ⁻⁵ µc/sample	0	0	0	0
Strontium	3.1x10 ⁻⁵ µc/sample	0	11	3	60
Tritium	5.0 µc/l	137	409	223	753
Uranium	0.14 ugm/l	0	0	169	699
Special Studies		0	0	55	145

Calibrations

	<u>Number of Units Calibrated</u>	
	<u>April</u>	<u>1964</u>
<u>Portable Instruments</u>		
CP Meter	1188	4363
Juno	285	1054
GM	531	2120
Other	270	956
Audits	115	418
Total	2389	8911
<u>Personnel Meters</u>		
Beta-Gamma Film	972	2750
Rings	75	345
Other Film	357	1059
Total	1404	4154
Miscellaneous Special Services	238	953
Total Number of Calibrations	4031	14,018

Environmental Monitoring

<u>Samples</u>	<u>April</u>	<u>1964</u>
Air		
Filters	231	1370
Scrubbers	271	955
Water		
Raw	22	169
Sanitary	75	303
Process	8	81
Vegetation	97	337
Test Well	423	1229
Fish	153	506
Waterfowl	0	77
Food Products	36	171
Beef Thyroids	54	157
<u>Measurements</u>		
Control Plots	87	210
Aerial Monitoring	3	7
Ionization Chambers	288	925

Manager
RADIATION PROTECTION

AR Keene:lsp

FINANCE AND ADMINISTRATIONACCOUNTINGCost Accounting

Authorized funds for nonproduction research and development programs assigned to Hanford Laboratories were adjusted in April as follows:

	<u>Increase</u>
<u>04 Program</u>	
Plutonium Recycle Program	\$ 20 000
Fueled Graphite Studies	220 000
EBWR Program	<u>60 000</u>
Total	<u>\$300 000</u>

The Budget for FY 1966 and Revision of Budget for FY 1965 was essentially completed by midmonth. The following additional research and development proposals were submitted to RLOO-AEC:

Fast Fuel Test Reactor
Plutonium and U²³³ Fueling of a Fast Reactor
Effect of Irradiation Temperature on the Accumulation of Radiation
Damage in Materials

Special accounting codes were established during April for the activities described below:

- .5B - Evaluation of Tungsten - Technetium Alloys for NASA. \$60,000 has been authorized for fabrication and evaluation of experimental alloys.
- .5C - Exhibits for Third Geneva Conference. \$1,000 has been authorized for improvements to (1) Eddy current device to inspect long U tubes in Heat Exchanger, and (2) Ultrasonic tester for thin walled diameter tube.
- .5D - Exhibits for Third Geneva Conference. \$17,000 has been authorized for preparation of Waste Calcination and Isotopes Development displays.
- .2C - Mass Spectrometer Analyses for Vallecitos Atomic Laboratories. \$2,000 has been authorized for the period of 3-9-64 to 12-31-64 for determination of the isotopic distribution of Xenon and Krypton in samples of fission gases.

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.2D - Advance Test Reactor Heat Transfer Burnout Studies for Phillips Petroleum Company. \$150,000 has been authorized for acquisition and analysis of boiling burnout data for a model of the ATR fuel element and its expected operating conditions.

Program code .34 was established during the month for a new 04 Program activity entitled "Fueled Graphite Studies."

General Accounting

Approval letter No. AT-339, PRTR Fuel Fabrication Movie, was sent to RLOO-AEC on April 27, 1964.

The following revised OPGs were issued during the month of April:

<u>OPG No.</u>	<u>Title</u>
8.20	Shipments of Property
22.1.11	Programming Operation
3.4.16	Continuity of Service and Leaves of Absence

Hanford Laboratories' net material investment at April 1, 1964 totaled \$25.9 million as detailed below:

(In thousands)

SS Material	\$24 290
Reactor and Other Special Materials	1 325
Spare Parts	367
Yttrium	<u>26</u>
Subtotal	26 008
Reserve: Spare Parts	\$77
Yttrium	<u>26</u>
	(103)
Net Inventory Investment	<u>\$25 905</u>

The cumulative value of nuclear material consumed in research by Hanford Laboratories during FY 1964 (at April 1, 1964) is as follows:

02 Program	\$ 58 359
03 Program	279 375
04 Program	<u>449 194</u>
Total	<u>\$786 928</u>

A certification type inventory of Other Special Materials in the custody of 111 Hanford Laboratories' material holders was conducted as of April 30, 1964. The inventory revealed a deficit of \$36.22 due to one holder inadvertently indicating the wrong quantities on two Material Transaction Reports prepared to document consumption. This was verified, and the necessary correcting documents are in process to relieve the inventory account.

A report of findings in connection with the FY 1964 physical inventory of movable catalogued equipment in the custody of Hanford Laboratories was prepared. This inventory, which was a certification type inventory, represented a verification of all movable catalogued equipment, prototype (RDX) equipment, and property of others in the custody of Hanford Laboratories. The Inventory for FY 1964 disclosed 15 missing items valued at \$4,733, compared to 40 missing items valued at \$14,496 in the FY 1963 inventory. The amount of missing equipment (15 items) represents 0.08% of the inventory, consisting of 17,765 items.

A ten-year forecast of SS Material for the 06 Program was prepared and submitted to Nuclear Materials Operation.

The status of Hanford Laboratories' heavy water inventory at April 30, 1964 is as follows:

	<u>Current Month</u>		<u>FY to Date</u>	
	<u>Pounds</u>	<u>Value</u>	<u>Pounds</u>	<u>Value</u>
Beginning balance	55 653	\$746 351	29 774	\$ 400 600
Acquisitions			54 890	755 988
Scrap returns			(19 091)	(240 940)
Consumption -1)				
(Loss & Scrap)				
PRTR	(945)	(14 037)	(10 585)	(177 730)
PRCF			(245)	(3 943)
All Other			(35)	(484)
Adjustments (SROO new price)				(1 176)
Ending balance	<u>54 708</u>	<u>\$732 314</u>	<u>54 708</u>	<u>\$ 732 315 -2)</u>

(1- Consumption quantity reflects losses only. Consumption value reflects both losses and scrap.

(2- Includes 17,961 pounds of heavy water scrap valued at \$224,060.

Laboratory Storage Pool activity is summarized as follows:

	<u>Current Month</u>		<u>FY to Date</u>	
	<u>Quantity</u>	<u>Value</u>	<u>Quantity</u>	<u>Value</u>
Beginning balance	1 971	\$1 546 977	1 480	\$ 811 520
Items received	104	51 946	1 753	1 225 702
Items reclaimed by custodians	(18)	(4 915)	(197)	(93 135)
Equipment transfers	(22)	(6 842)	(318)	(133 454)
Items disposed by PDR	(4)	(427)	(150)	(11 816)
Items disposed by excessing	<u>(124)</u>	<u>(29 772)</u>	<u>(661)</u>	<u>(241 850)</u>
Ending balance	<u>1 907</u>	<u>\$1 556 967</u>	<u>1 907</u>	<u>\$1 556 967 -1)</u>

(1- Includes 176 items valued at \$119,738 on loan at April 30, 1964.

	<u>Current Month</u>		<u>FY to Date</u>	
	<u>Quantity</u>	<u>Value</u>	<u>Quantity</u>	<u>Value</u>
Loans & Transfers in Lieu of Purchase				
Loans	160	\$ 74 581	847	\$ 344 907
Transfers	<u>22</u>	<u>6 842</u>	<u>318</u>	<u>133 454</u>
Total	<u>182</u>	<u>\$ 81 423</u>	<u>1 165</u>	<u>\$ 478 361</u>
Operating Costs (3-29-64)		<u>\$ 1 222</u>		<u>\$ 13 274</u>

The value of material and equipment in custody of the Laboratory Storage Pool at April 30, 1964 totaled \$2.5 million, as detailed below:

Equipment	\$1 556 967
Reactor and Other Special Materials	315 985
SS Material	154 800
Other Material	<u>490 243</u>
Total	<u>\$2 517 995</u>

An extensive excessing program is currently under way within the Laboratory Storage Pool. Five hundred forty-five items valued at \$309,194 are in the process of being excessed, of which 236 items valued at \$82,636 have been forwarded to the Excess Yards at Central Stores. The remaining items will be transferred during May 1964.

Action as indicated occurred on projects during the month:

New Money to Hanford Laboratories

CAH-962 Low Level Radiochemistry Building \$20 000

Contract CA-438, M. E. Emsinger, was processed during the month.

Personnel Accounting

Personnel statistics follow:

<u>Employee Changes</u>	<u>Total</u>	<u>Exempt</u>	<u>Nonexempt</u>
Employees at beginning of month	1 802	803	999
Additions and transfers in	11	5	6
Removals and transfers out	<u>16</u>	<u>7</u>	<u>9</u>
Employees on payroll at end of month	<u>1 797</u>	<u>801</u>	<u>996</u>

Overtime Payments During Month

	<u>April</u>	<u>March</u>
Exempt	\$ 2 950	\$ 4 204
Nonexempt	12 025	13 342
Total	<u>\$ 14 975</u>	<u>\$ 17 546</u>

Gross Payroll Paid During Month

Exempt	\$ 802 397	\$ 804 463
Nonexempt	570 864	697 885
Total	<u>\$1 373 261</u>	<u>\$1 502 348</u>

Participation in Employee Benefit Plans at Month End

	<u>April</u>		<u>March</u>	
	<u>Number</u>	<u>Percent</u>	<u>Number</u>	<u>Percent</u>
Pension	1 618	99.5	1 614	99.5
Insurance Plan - Personal	405		402	
- Dependent	1 392	99.9	1 396	99.9
U. S. Savings Bonds				
Stock Bonus Plan	150	37.0	150	36.9
Savings Plan	66	3.7	67	3.7
Savings and Security Plan	1 240	88.8	1 249	89.5
Good Neighbor Fund	1 291	71.7	1 294	71.8

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H-6

HW-82001

	<u>April</u>		<u>March</u>	
	<u>Number</u>	<u>Amount</u>	<u>Number</u>	<u>Amount</u>
<u>Insurance Claims</u>				
<u>Employee Benefits</u>				
Life Insurance	0	\$ -0-	1	\$10 139
Weekly Sickness & Accident	12	957	17	1 733
Comprehensive Medical	74	4 976	80	5 032
<u>Dependent Benefits</u>				
Comprehensive Medical	<u>141</u>	<u>13 922</u>	<u>170</u>	<u>14 132</u>
Total	<u>227</u>	<u>\$19 855</u>	<u>268</u>	<u>\$31 036</u>

TECHNICAL ADMINISTRATION

Requisitions for 14 nonexempt employees were filled; 7 remain to be filled.

Suggestion Plan activity:

Suggestions received	33
Suggestions adopted	12
Suggestions rejected	41
Suggestions in process	104

Visitors Center activity:

April attendance	2 060
Average attendance per day open	79
Cumulative attendance since 6-13-62	66 029
Conducted groups	20 (totaling 453 people)

Plant Tour activity:

	<u>Number</u>	<u>Total</u>
General public relations tours	16	332
Special tours	4	108

Overall recruiting results for March are as follows:

Offers extended	15
Offers accepted	13
Offers rejected	34
Added to roll	2

Advanced Degree - Ten Ph.D. applicants visited HAPO for employment interviews. Three offers were extended; one acceptance and five rejections were received. Two offers are currently open.

BS/MS (Direct Placement) - One offer was extended. One acceptance and one rejection was received. There are three offers currently open.

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BS/MS (Program) - Eleven offers were extended. Eleven offers were accepted and 28 rejected. Nineteen offers are currently open.

Technical Graduate Program - Eight Technical Graduates were placed on permanent assignment. No new members were added to the roll and none terminated from the roll. Current Program members total 50.

FACILITIES ENGINEERING

At month end, Facilities Engineering Operation was responsible for eight active projects having total authorized funds in the amount of \$7,543,500. The total estimated cost of these projects is \$10,550,000. Expenditures through March 31, 1964 were \$2,914,000.

The following summarizes project activity in April:

Authorized projects at month end -----	8
New projects authorized -----	0
Projects completed -----	0
New projects submitted to the AEC in April -----	1
CAH-136, Services Addition - 327 Building	
New projects awaiting AEC approval -----	6
CAH-114, Critical Mass Laboratory Addition	
CAH-123, Laboratory Fire Protection System	
CAH-126, Waste Transport System	
CAH-128, Heat Transfer Apparatus for Model Studies	
CAH-136, Services Addition - 327 Building	
CAH-137, Temporary Physical Sciences Center	
Project proposals being prepared -----	4
Atmospheric Physics Building	
308 Building Addition	
Geological and Hydrological Wells - FY 1964	
Variable Spectrum Test Reactor	

The status of active projects follows:

CAH-916 - Fuels Recycle Pilot Plant - Construction is 48 percent complete compared to a scheduled 36 percent. Drawings of the shielded windows were submitted by the vendor for approval. The Company has not approved the drawings because the design fails to satisfy our specification. Construction at the site is progressing very well.

CAH-922 - Burst Test Facility for Irradiated Zirconium Tubes - A solution to calibration problems on the control instrumentation for the heating and ventilating equipment has not been received from the vendor. Installation of the prototype vessel has been completed.

CAH-962 - Low Level Radiochemistry Building - Construction started April 7, 1964 with layout, grading, and other site work. Excavation for the building was completed and some minor concrete pours made. A design change will be issued to reduce the scope of work by about \$20,000. The design change will eliminate ground cover, sprinkling system, south parking lot, power pole relocation, numerous cabinets, and some block walls.

CAH-977 - Facilities for Radioactive Inhalation Studies - Design is complete. An alternative plan to locate these facilities in the 300 Area near the proposed 300 Area Biology site was reviewed jointly by representatives of the Company, Richland Operations Office, and Division of Biology and Medicine.

CAH-982 - Addition to Radionuclide Facilities, 141-C Building - Design is complete. Future work on this project is also being integrated into the plan to relocate the Biology Laboratory into the 300 Area.

CGH-999 - Plutonium Recycle Critical Facility Conversion to Light Water - The Commission requested that this project be closed and that a new project be initiated to provide facilities permitting the handling of irradiated fuel. The physical completion notice had previously been issued.

CAH-100 - High Temperature Lattice Test Reactor - Vitro Engineering Company reports detailed design at 55 percent complete on April 30, 1964. No official design schedule has been established and approved by the Commission. Design costs through April 19, 1964 were: Vitro Engineering Company - \$205,000, General Electric Company - \$11,000. A directive was issued authorizing total design funds in the amount of \$320,000.

Design of the buildings and utilities is scheduled to be complete by May 15, 1964. A total of 50 detailed design drawings have been received for review and comment by the Company.

Several meetings were held among Commission, Vitro, and Company representatives to review project responsibilities and design progress and to resolve design problems. It is felt the meetings have improved coordination of the work.

CAH-114 - Critical Mass Laboratory Addition - The project proposal was submitted to the Commission on November 4, 1963. Action by the Commission in Washington is still awaited.

CAH-116 - PRTR Decontamination and D₂O Cleanup - Detailed design is 24 percent complete compared to a scheduled 62 percent. No Title II design drawings have been received by the Company. Comments on Title I design were submitted to the Commission on March 18, 1964. No Title II drawings have been received for comment.

CAH-119 - PRTR Storage Basin and Experimental Facilities Modifications - Detailed design of the building and basin by Vitro Engineering is about 42 percent complete compared to a scheduled 72 percent. The design criteria for the underwater equipment were approved by the Company and returned to the Commission on April 16. The criteria document had been rewritten to delete the welding equipment.

CAH-123 - Laboratory Fire Protection System - Action on the project proposal was deferred for further study by the Commission at its April 30 Project Review Board meeting.

CAH-126 - Waste Transport System - Action on the project proposal was deferred for further study by the Commission at its April 30 Project Review Board meeting.

CAH-128 - Heat Transfer Apparatus for Model Studies - 185-D Building - The Commission plans to return the project proposal "unapproved."

CAH-136 - Services Addition - 327 Building - The design criteria were approved early in April. The project proposal was transmitted to the Commission about April 1. Action was deferred by the Commission at its April 30 Project Review Board meeting.

CAH-137 - Temporary Physical Sciences Center - The Commission's Project Review Board approved the project proposal and associated Contingency Maintenance Request on April 30, 1964.

Engineering Services

Engineering work was performed in support of design and construction on active projects, project proposals, preliminary planning, and design criteria for new projects. Other work items included preparation of as-built drawings for CGH-995, 309 Air Conditioning Modifications.

Facility planning and budget study work included: (1) completion of a draft of a formal report on the Biology Facility Relocation to 300 Area; (2) initiation of a study of single building plan for certain laboratories proposed in FY 1966 budget; (3) preparation of budget facility descriptions for the Chemical Metallurgy Laboratory, the Service Addition to the 325-A Building, and

Extension of the 327 Building Canyon; (4) initiation of engineering study on relocation of certain 231-Z Pu Metallurgy Laboratory equipment to the 326 Building; and (5) preparation of preliminary analysis on relocation of Physical Testing Facilities from White Bluffs Building C-25 to proposed addition to 306 Building.

Engineering and consulting work was provided to research and development personnel as requested. Major work items included engineering assistance or consultation: (1) in providing a decontamination glove box, a transfer glove box and means for transferring plutonium bearing specimens to and from an electron microscope in 325 Building without contamination spread; (2) in selecting equipment for automatically sampling the reactor effluent water for isotopic analysis; (3) in developing a variable thickness slab tank for critical mass experiments; (4) to delineate scope of service piping for 324 Building waste calcination; (5) on instrumentation on several high pressure loops in the 314 Building; (6) preparation of specifications for computer trailer ventilation equipment; (7) on an experimental Pu glove box for analytical laboratories in 325 Building; and (8) engineering of the core for the second loading of the critical facilities in PRTR.

Plant engineering work during the month included: (1) study of use of 325 Building basement space; (2) engineering of a metal spacer to permit installation of a charcoal filter in a high-efficiency filter box in 325-B Building; (3) review of 747 Building utilities and mechanical equipment for presentation to prospective bidders; (4) evaluation of a classified scrap disintegrator recommended by Vitro; and (5) evaluation of feasibility to repair or replace the damaged freon compressor in the 329 Building.

Emergency engineering service and miscellaneous consultation was provided on: (1) removal of a highly contaminated filter from the 327 Building; (2) power supply components required for a motor generator set to be used to isolate the NMR laboratory power supply in 325 Building; and (3) electrical requirements for a mixing and dump tank combination for plutonium solutions in the Critical Mass Laboratory.

Consulting service for other departments included: (1) engineering on revision to Standard Specification HWS-7511-S on high efficiency filter; (2) consultation to CPD on Purex filter problem; (3) assistance to CPD in hazards evaluation study for Projects CAC-945 and CAC-970; (4) assistance to CPD on impact problem with closure of disposal tunnel door; and (5) instrumentation consultation to CPD on the Americium Recovery Facility, the 234-5 Building dry air system controls, and the Fab Oil Prototype Facility.

Pressure Systems

Engineering work on the geometry of the sodium loop to obtain acceptable thermal stresses was checked by ADP calculations and found to be satisfactory.

The Travelers Indemnity Company inspector reviewed 13 Laboratories vessels for compliance with code. Twelve were accepted and recommendations for modification to the other were made.

Facilities Operation

Landlord costs for the month of March were \$154,680, which is 89 percent of the forecast for the month. Fiscal costs for nine months are 102 percent of the predicted. Reduction of the budget overrun is so favorable that the maintenance work level is being increased.

The following table summarizes the Waste Disposal Operation:

<u>Item</u>	<u>February</u>	<u>March</u>
Concrete waste barrels disposed to 300-Wye burial ground	4	7
Concrete waste barrels disposed to 200-W plutonium burial ground	5	8
Loadluggers of dry waste disposed to 300-Wye burial ground from 300 Area sites other than the 325 building	19	26
Loadluggers of dry waste disposed to 300-Wye burial ground from the 325 building	2	2
Loadluggers of dry waste disposed to 200-W plutonium burial ground from 300 Area sites	7	7
Crib Waste Volume, gal.	310 000	365 000
Total Beta Conc., $\mu\text{c/gal.}$	131.0	602.0
Total Beta, Curies	40.6	219.7
Total Alpha Conc., $\mu\text{c/gal.}$	86.3×10^{-3}	656×10^{-3}
Total Alpha, mc	26.8	239.4
Pu Concentration, $\mu\text{g/gal.}$	0.215	4.15
Amt. Pu g.	0.067	1.5
U Concentration, $\mu\text{g/gal.}$	1510.0	1740.0
Amt. U g.	468.1	635.1

No unusual incidents were experienced this month in the waste disposal operations.

About 4,000 gallons of liquid waste were disposed from 325-A through the 340 system. This waste was previously being shipped in a 500 gallon cask to CPD.

Building Operations

Flow and pressure tests were completed for river pump discharge on April 22. Tests were made with one and with two river pumps in service and with the filter plant operating at maximum capacity of 2800 GPM.

Bearings were replaced on #1 supply motor at 326 building. Access panels were installed in the duct of #4 system, and all sand accumulation was removed from this system.

Deionizers at 327 building were put in service for the first time on April 22. Water quality was excellent. Water use from this system will be only as required to fill storage basins.

The 25 HP freon compressor at 329 building that services the "C" block failed on April 26. The unit was dismantled for inspection and repair costs would be excessive. New compressor unit was purchased and should be in service in early May.

Supply units in buildings 3760, 329, 326, 327, 328, and 325 were inspected for dust and sand leakage around the pre-filters. All cracks were caulked to prevent dirt bypassing these filters.

Drafting

The equivalent of 160 drawings were produced during the month for an average of 27 man-hours per drawing.

Major jobs in progress are: (1) ultrasonic tester for irradiated fuel elements, N Reactor; (2) high temperature furnace, 306; (3) bellows tank for critical mass studies; (4) 309 building piping service drawings; (5) FRPP service piping to waste calcination equipment; and (6) critical mass physics safety and control rods.

Construction

	<u>Unexpended Balance</u>
Orders outstanding beginning of month	\$498 577
Issued during the month (inc. supp. & adj.)	114 583
J. A. Jones expenditures during month (includes C.O. cost)	199 515
Balance at month end	413 645
Orders closed during month	159 342

Maintenance Work Orders active - 4, Face Value - \$15,679.

Major nonproject jobs in progress during the month were: (1) construct swine farrowing barn and pens, 141-F; (2) modify heating and ventilating system, 144-F; (3) install hoods and exhaust duct work, 1705-F; (4) construct new dog runs and modify existing dog runs, 144-F; (5) install canopy and services, 1705-F; (6) install fire detection, 100-F Area; (7) construct pasture and sprinkler system at meteorological tower; (8) renovate room 125 and install platform, stairway, and monorail, 308; (9) construct office addition and building services, and construct HTLTR mock-up, 314; (10) construct thorium laboratory room 417, install emergency generator, filter boxes and enlarge men's rest room, 325; (11) install lighting, fire detectors, and construct roof, 3718-A/B; (12) soundproof rupture loop annex, construct manhole and retention waste by-pass line, install snubbers in high pressure lines and install jib crane and hoist, 309; (13) modify E-cell and install safety valves, 327; (14) install electrical circuits to computer, 321; and (15) install hydrogen system, 306.

J. A. Jones Company issued invitations to bid for the design and construction of the containment vessel and the reactor simulator. Chicago Bridge and Iron Company was low bidder for the containment vessel. Bids for the simulator will be opened May 15, 1964. Construction work at 221-T is proceeding expeditiously. The ventilation barrier wall is complete, and the crane stops and cable relocated. The main access opening to the canyon through the five-foot thick concrete wall is completed. The new exhaust fan is installed. Change room and rest room facilities are 80 percent complete.

J. A. Jones Company awarded contract for renovation of 10 wells to Storey Drilling Company. Work will start early in May.

Acting Manager
Finance and Administration

DS Parsley:RDT:whm

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REACTOR DEVELOPMENT - O4 PROGRAMPLUTONIUM RECYCLE PROGRAMPlutonium Recycle Test ReactorOperation

Reactor output for April was a record 1851 MWD, for an experimental time efficiency of 91% and a plant efficiency of 88%. There were five operating periods during the month, four of which were terminated manually and one extended through month-end. A new record was set for the longest sustained critical period for the PRTR. A summary of the fuel irradiation program as of April 30, 1964, follows:

	Al-Pu		UO ₂		PuO ₂ -UO ₂		Other		Program Totals	
	No.	MWD	No.	MWD	No.	MWD	No.	MWD	No.	MWD
In-Core	0		7	1468.2	78	12136.9			85	13605.1
Maximum				294.6		288.5				
Average				209.7		155.6				
In-Basin	7	569.9	25	2720.1	33	2426.7			65	5716.7
Buried							1	7.3	1	7.3
Chem. Process.	68	5465.8	35	1965.8					103	7431.6
Program Totals	75	6035.7	67	6154.1	111	14563.6	1	7.3	254	26760.7

Note: (MWD/Element) X 20 $\approx$ MWD/TU for UO₂ and PuO₂-UO₂.

Heavy water and indicated helium losses for April were 945 pounds and 114,411 scf., respectively.

During the month, 1300 pounds of scrap heavy water was cleaned up, bringing the total of scrap heavy water ready for shipment to Savannah River to 17,000 pounds.

Equipment Experience

Only 20 reactor outage hours were charged to repair work this month, 11 of which were for heavy water and helium leak repairs.

Preventive maintenance required 154 hours or about 3.4% of the total maintenance effort.

Information gained in PRTR Test No. 90 indicates that the gas chromatograph is grossly sensitive to deuterium, appearing to be approximately a factor of eight more sensitive to deuterium than to the equivalent hydrogen.

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Improvement Work Status (significant items)Work Completed

Flash Tank and Blowdown Piping Modification
Gatling Can Waste Cask Balance Modification
Turbidity Panel Modification
Improvements in Operability - PRTR Demineralization System
Panel Radiation Indicating Instrument

Work Partially Completed

Storage Basin Work Platforms
Jib Crane at M&M Basement Access Well
In-line Gas Sampling
Process Tubes Level Indicator
Inlet Gas Seal Replacement
Shim Rod Shroud to Top Cap Modification
Improved RTD Connector Sealant and Bracing
Instrument Power Transfer System
Installation of New Alarm Annunciator
Thermistor Probe Installation in FEEF
Flow Monitor Tubing Snubber Installation
Indication of DC Solenoid Failure
DT-1 Storage Tank Sight Gauge
Battery Power for Galvanometer Light
Vibration Snubbers for Earthquake Protection
Process Sewer Bypass Manhole #2A
Process Tube Examination Equipment Installation
Compressed Air Modifications

Design Work Completed

High Pressure Filter Installation
Sump Filter - High Pressure Helium Compressor No. 2
Thermal Barrier Modification
New Effluent Monitor Cables
Elimination of Exposed Voltage on Terminal Strips
Automatic Controller Modifications (DC-198-3)
Process Sewer Bypass - Manhole #2
Pressurizer Level Switch Repair

Design Work Partially Completed

Additional Fuel Storage and Examination Facility (CAH-119)
Decontamination Building and D₂O Cleanup Facility (CAH-116)
PRTR Steam Utilization (CAI-120)
Flux Wire Scanning System
Supplemental Emergency Water Addition
Permanent Installation of Closed Circuit TV
Rupture Monitoring System Modifications

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Procedures

Revised Operating Procedures issued	14
Revised Operating Standards issued	9
Temporary Deviations to Operating Standards issued	5
Revised Process Specifications accepted for use	3
Maintenance Procedures issued	1

	<u>April</u>	<u>Total</u>
Drawing As-Built Status:		
Approved for As-Built	18	1 330
In Drafting		110
In Approval		9
Deleted or Voided		87
		<u>1 536</u>
Scheduled for Review		79
Total		<u><u>1 615</u></u>

	<u>Manhours</u>
Personnel Training:	
Qualification Subjects	335
Specifications, Standards, Procedures	85
Emergency Procedures and Safety	67
FERTF and PRCF	112
Maintenance Procedures	19
	<u><u>618</u></u>

Status of Qualified Personnel at Month-End:

Qualified Reactor Engineers	9
Qualified Lead Technicians	6
Qualified Technicians	22

Process Engineering and Reactor Physics

HW-79929, The Effect of Primary Coolant Degradation on Core Reactivity in PRTR, has been revised to include references to the Final Safeguards documents for PRTR and a discussion of critical test results for voiding the primary coolant channels.

PRTR Test No. 93 is being performed to permit evaluation of the effect of a 0.3% hydrogen addition to bulk helium on oxygen content in all reactor systems. Preliminary indications are that the added hydrogen provides considerable reduction of oxygen in all systems. Further testing will be required to establish an optimum concentration.

Analysis of reactor safeguards at 120 MW was continued. Core kinetic studies have been completed. PRTR Test No. 73, Supplement No. 1, has been completed. Preliminary analyses indicate that the increased flow to the barometric condenser, that operation at 120 MW would require, can be accomplished by changing out a flow restricting orifice. Flow rates and resulting pressures in the water chest are well within the design capabilities of the equipment and the needs.

On the reactor startup of April 8, a very minor burst of primary coolant activity was observed. A primary coolant analysis taken at the time showed the I-131 activity to be a factor of 200 higher than the normal. The noble gas activity was unusually low in comparison. Reactor operation was continued with the defective element and the I-131 activity dropped back to the normal background by April 10. The reactor was operated continuously through month-end without any further indications of a defective element.

Process Engineering and Reactor Physics

The status of the various test elements at the end of April 1964, is shown below. Those elements which had reached their assigned goal exposure or had been permanently discharged for other reasons prior to April 1, 1964, have been deleted from the table.

Test Channel No.	F.E. Location	F.E. Number	Description	Date Initial Charge	Date Dis-charged	Approximate Accumulated MWD
14	1956	5097	Moxtyl-Swaged	4/2/62	--	182.3 repad
14	1352	5098	Moxtyl-Vipac	5/8/62	--	273.9 repad
14	1758	5099	Moxtyl-Vipac	5/8/62	--	195.0 repad
48	1156	5150	Moxtyl ($\frac{1}{2}$ " x $\frac{1}{2}$ " pads)	8/1/62	--	191.7
54	1542	5116	Moxtyl (clip-on pads)	5/8/62	--	197.6
54	1554	5118	Moxtyl (clip-on pads)	5/8/62	--	288.5
61	1249	5185	Moxtyl-Physics	5/28/63	--	182.8
61	1354	5186	Moxtyl-Physics	5/28/63	--	175.0
61	1445	5192	Moxtyl-Physics	6/13/63	--	183.7
67	1152	5119	Moxtyl (Repaired Wire)	10/20/63	--	107.9
67	1457	5117	Moxtyl (Repaired Wire)	10/20/63	--	148.6
80	1544	5214	Moxtyl(1% PuO ₂ , Swaged)	11/18/63	--	118.7
85	1847	5230	Moxtyl(1% PuO ₂ , Vipac)	1/30/64	--	67.6
37	1548	1098	UO ₂ -Physics	5/12/62	--	171.8
37	1550	1097	UO ₂ -Physics	5/12/62	--	183.1
37	1451	1100	UO ₂ -Physics	5/12/62	--	150.7
37	1552	1099	UO ₂ -Physics	5/12/62	--	152.8

Two fuel elements were inspected in the basin during the month, one of which was rejected due to a deep "pock" mark. Three rods of fuel element 5187 were shipped to Radiometallurgy for testing.

An eddy current test of a blistered process tube (5526) was conducted in the basin. A preliminary, verbal report indicated a defect 1/2 to 1" long, ~1/2" around the circumference and at an average depth of .010" from the inside surface.

Plutonium Recycle Critical Facility

Experiments with EBWR fuel continued. Data was taken for rod and sheet calibrations, moderator level sensitivity, horizontal and vertical traverses, and reactor noise analysis. Void and substitution measurements were in progress at month-end.

Design of a second H₂O core for PRCF was partially completed.

One Process Specification and one Temporary Deviation to a Process Specification were accepted for use.

During the month, one TRA0 engineer became qualified as a PRCF supervisor and a PII physicist became qualified as a PRCF experimenter.

Fuel Element Rupture Test Facility

The FERTF was not operated during April. The test section had been removed from the reactor to permit a thorough inspection of the piping.

The failed weld was repaired. Eight other welds were checked by X-ray, and found to be in good condition.

Improvement work completed consisted of NPC-213, Rupture Loop Revised Piping Supports and NPC-186, Rupture Loop Pump 1A and 1B - Pump Vent Connections.

The final report of FERTF Test 4, Irradiation of Al-Pu Leakers, was issued.

Processing of Spent PRTR Fuels

The chemical processing of 35 Al-Pu fuel elements was completed at Redox. Plutonium remaining (0.126 Kg) from the first campaign was added to 6.147 Kg from this (third) campaign, for a total 6.273 Kg. One batch of 2.263 Kg had a Pu²⁴⁰ content of 22.44% and the remaining batch of 4.01 Kg had a Pu²⁴⁰ content of 27.34%. A shipment of 3.7 Kg was made to the 308 Building.

Approval to use the lighter shipping basket was received from Redox. However, fourteen of the 4-hole and one 3-hole shipping basket of previous design are still available and will be used up on future campaigns.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 26,703 hours. This includes 17,582 hours performed in Technical Shops, 6,618 hours assigned to J. A. Jones Company, 2,284 hours assigned to offsite vendors, and 219 hours to other project shops. Total shop backlog is 15,026 hours, of which 90% is required in the current month with the remaining hours distributed over a three-month period. Overtime worked during the month totaled 272 hours or 1.2% of the total available hours.

Distribution of time was as follows:

	<u>Manhours</u>	<u>% of Total</u>
N Reactor Department	2 366	8.8
Irradiation Processing Department	4 638	17.4
Chemical Processing Department	763	2.9
Hanford Laboratories	18 936	70.9
Hanford Utilities and Purchasing Department	0	0

LABORATORY MAINTENANCE OPERATION

Total productive time was 23,000 hours of 25,200 hours potentially available. Of the total productive time, 85% was expended in support of Hanford Laboratories components, with the remaining 15% directed toward providing service for other HAPO organizations. Craft overtime worked during the month was 3.9% of total available hours.

Manpower utilization (in hours) for April was as follows:

A. Shop Work	2 500
B. Maintenance	8 700
1. Preventive Maintenance	2 600
2. Emergency or Unscheduled Maintenance	2 000
3. Normal Scheduled Maintenance	4 100
C. R&D Assistance	11 800

W. K. Williams

Manager

Test Reactor and Auxiliaries

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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>
F. B. Quinlan	The Art of Gripping Wire
F. B. Quinlan	Tube Expanding and Mandrel Removing Machine
R. D. Weed and L. W. Wilson	Quantitative Determination of Oxalate Ion in the Presence of Peroxide and Organic Compounds
J. J. Regimbal	D ₂ O Swimming Pool Reactor Employing a Floating Liquid Hydrocarbon Radiation Shield (HWIR-1720)
E. M. Sheen	Relay Circuit Fault Monitor (HWIR-1719)
W. L. Bunch, J. L. Stringer, and D. E. Cooley	Irradiated Fuel Age Determination (HWIR-1724)
J. R. Lundquist and R. W. Stromatt	Preparation of Uranium Metal for Electroplating, by Anodization in H ₂ SO ₄ -HCl

Manager, Hanford Laboratories

END

**DATE
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4 / 13 / 93