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HANFORD LABORATORIES MONTHLY ACTIVITIES REPORT

JULY, 1963

AUGUST 15, 1963

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HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

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MONTHLY ACTIVITIES REPORT
JULY, 1963

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Compiled by
Section Managers

August 15, 1963

By Authority of CP-PP-2
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HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

PRELIMINARY REPORT

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

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Table I - Hanford Laboratories Force Report

Date: July 31, 1963

	At Beginning of Month		At Close of Month		Total
	Exempt	Salaried	Exempt	Salaried	
Chemical Laboratory	147	132	149	133	282
Reactor & Fuels Laboratory	193	185	200	193	393
Physics & Instruments Laboratory	107	82	110	82	192
Biology Laboratory	46	65	45	64	109
Applied Mathematics Operation	20	5	20	5	25
Radiation Protection Operation	47	94	47	94	141
Finance & Administration Operation	127	113	150	120	270
Programming Operation	15	4	15	5	20
Test Reactor & Auxiliaries Oper.	59	308	58	308	366
General	3	4	3	4	7
TOTAL	764	992	797	1008	1805

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BUDGET AND COST SUMMARY

Information received too late for inclusion in last month's report reveals June operating costs totaling \$3, 441, 000. Hanford Laboratories' FY 1963 research and development costs for June and the completed fiscal year compared with the control budget are shown below:

	COST			
(Dollars in thousands)	<u>June Cost</u>	<u>FY Total Cost</u>	<u>FY 1963 Budget</u>	<u>% Spent</u>
HL Programs				
02	\$ 123	\$ 982	\$ 994	99%
03	25	165	175	94
04	1 715	12 649	12 683	100
05	127	1 303	1 353	96
06	310	3 144	3 154	100
08	10	83	97	86
	<u>2 310</u>	<u>18 326</u>	<u>18 456</u>	<u>99</u>
Sponsored by				
FPD	--	493	493	100
NRD	175	1 279	1 270	101
IPD	75	907	909	100
CPD	100	1 374	1 391	99
Total	<u>\$ 2 661</u>	<u>\$ 22 378</u>	<u>\$ 22 519</u>	<u>99%</u>

July operating costs totaled \$2, 418, 000 or 7% of the \$32, 455, 000 preliminary control budget. Hanford Laboratories' research and development costs for July compared with the control budget were:

(Dollars in thousands)	<u>July Cost</u>	<u>FY 1964 Budget</u>	<u>% Spent</u>
HL Programs			
02	\$ 70	1 180	6%
03	25	250	10

(Dollars in thousands)	<u>July Cost</u>	<u>FY 1964 Budget</u>	<u>% Spent</u>
HL Programs, Cont'd			
04	1 010	13 485	7
05	109	1 456	7
06	254	3 604	7
08	15	100	15
	<u>1 483</u>	<u>20 075</u>	<u>7</u>
Sponsored by			
NRD	121	1 761	7
IPD	56	696	8
CPD	<u>118</u>	<u>1 668</u>	<u>7</u>
Total	<u>\$ 1 778</u>	<u>\$ 24 200</u>	<u>7%</u>

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

Sixteen experimental lithium-aluminum target elements are under irradiation for evaluation of a tritium producing element for N-Reactor. Initial specimens will be discharged in August.

Autoclaving N-fuel elements produced oriented hydride platelets in the inner supports which drastically reduced their ductility. Annealing supports at 750 or 700 C before welding them to the fuel elements prevents subsequent orientation of hydrides and allows supports to retain their ductility after autoclaving.

"Buggy spring" supports from N-inner fuel components irradiated to 1500 Mwd/ton show little hydrogen pickup in-reactor but do show oriented hydride platelets. However, no indications of inadequate support performances were evident.

In the studies of the effect of clad uniformity on strain limits for N-fuel cladding, 25 rod specimens irradiated to 1100 Mwd/ton have been examined. Clad necking occurs more rapidly on unalloyed fuel rods than on U-2 wt % Zr fuel rods.

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On specimens with the cladding deliberately striated to a depth of 0.004 inch, the localized cladding strain was approximately 20 times the average cladding strain.

Continuing tests on damage to process tubes by scratching have shown that a 400 C autoclaving cycle does not accelerate damage by causing recovery of cold-worked scratched surfaces. However, reautoclaving between tests causes increased damage rates; this effect is apparently associated with the autoclave film.

Metal Fabrication Development Operation recently completed 16,275 N-outer supports. The total number of outer supports produced by MFDO is 113,474.

Ductile Zr strip has been successfully made for cold swaging 1-inch-diameter extruded rod, followed by turk's heading to finished strip size. This fabrication procedure reduced scrap loss to about 10% as compared to the 30% loss entailed in the rolled sheet method.

Pure vanadium wire of 0.030-inch diameter has been drawn from 0.25-inch rod. This wire was successfully fabricated after it could not be procured from off-site vendors.

Three prototype thorium-uranium fuel elements have been irradiated three ETR cycles to an exposure of 2400 Mwd/ton. After two cycles, no detectable volume increases had occurred.

Zircaloy-2 clad thorium-uranium rods containing a solid Zircaloy-2 core have been successfully fabricated. These rods will be thermocouple fuel pieces for heat transfer studies.

Testing in KER-1 of NH_4OH as an alternate for LiOH for pH control indicates that radiolysis of the water at low temperature is considerably less with NH_4OH than LiOH . Initial, although not conclusive, results indicate that turbidity (crud level) is no worse with NH_4OH than LiOH .

A comparison of radiation in KER-1 (using NH_4OH) and KER-2 (using LiOH) indicates that the radiation levels in KER-2 are about 1-1/2 times those in KER-1 during high temperature operation, and 1/2 or 1/3 times as great during startup.

Examination of two zirconium-clad N-Reactor target elements which had been subjected to a rupture test showed that although the corrosion of the aluminum outer layer was very severe, the attack had not penetrated to the Li-Al core. There was enough Al release to the coolant for the detection of such a rupture before the Li-Al would be affected.

An investigation of the possible contaminants which could have caused the stress cracking and intergranular pitting found in the N-Reactor steam generators has uncovered no likely cause. The units were found to be very clean except for vapor phase inhibitor (VPI) and a small amount of residue from chemical cleaning. Analyses of this residue are not yet conclusive. Accelerated tests are being conducted to determine whether VPI could have caused stress cracking.

A test was started in TF-3 to determine whether corrosion in the heated crevice under N-Reactor fuel supports can be prevented by using NH_4OH rather than LiOH . An electrically heated zirconium-clad rod with N-Reactor supports welded to it is being tested under conditions of surface boiling.

Preparations were made to perform heat transfer experiments with an electrically heated test section representing a half-length column of N-Reactor fuel elements.

Comparison of boiling burnout results for data obtained at 1000 psi with flow through a heated tube from Columbia University and Hanford showed excellent agreement.

Apparatus for study of the kinetics of the reaction $\text{H}_2\text{O} + \text{CO} \rightarrow \text{CO}_2 + \text{H}_2$ has been completed and a run has been made. The results agree reasonably well with rate constants previously determined for the homogeneous reaction.

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In static experiments under irradiation, a G value of 4.1 ± 0.7 for hydrogen formation has been determined. There is indication that energy absorbed in the helium affects the water-carbon monoxide reaction.

N-Reactor graphite irradiations are proceeding satisfactorily in the GETR. An excellent, uniform black oxide layer was observed on the surface of an irradiated PRTR test element that had been charged into the reactor without the usual autoclaving of the completed fuel rods; irradiation of the fuel element is being continued. Two additional nonautoclaved fuel elements were charged into the PRTR.

One hundred thirty-one days of exposure in the TF-7 facility were completed by a special UO_2 Mark-I fuel element having ring end bracket contacts; only negligible wear was observed. Assembly of other test elements for evaluation in the EDEL-1 loop is proceeding.

Newly developed analytical techniques disclosed excessive concentrations of fluorine and water in PuO_2 powder that had recently been produced by air oxidation of Pu metal. Several UO_2 - PuO_2 fuel elements now being used in PRTR may contain PuO_2 contaminated by these halides. These elements should be considered candidates for mild cladding failure by a mechanism involving these impurities. An extensive evaluation of archive specimens, fuel currently in process, and development of additional fuel purification processes are proceeding on a high priority basis.

Four hundred pounds of "cold" Salt-Cycle UO_2 was received from the Chemical Development Operation for experiments in the remote fabrication facility.

A rejuvenated fuel element, GEH-4-84, completed the second irradiation cycle in the MTR.

An underwater drill for fuel repairs was fabricated and tested. A small spot welding device of the pliers type was purchased and installed for experiments in underwater resistance joining of a sheath to a wire.

No evidence of nonuniform or excessive corrosion was seen on boroscope inspection of PRTR aluminum shroud Tube 1847. Small dark spots observed randomly on the uniform oxide surface are apparently stains.

Sensing lines from the PRTR calandria to the moderator level servo-manometer in which the normal transmitting medium is dry helium have corroded through in several areas. Examination of aluminum sensing tube sections indicates failure by pitting, apparently the results of periodic flooding with water contaminated by chlorides for the gas drying system.

Hydride analysis of Tube 5679 from PRTR (the fourth tube discharged) indicated that this tube had about 100 ppm hydrogen. This is an increase of 70 ppm.

Zirconium concentrations in the PRTR primary coolant were normal (1-1 ppb), except for two excursions, 1 for 3 days with concentrations greater than 10 ppb, the other for 1 day and 3 ppb. The first excursion may indicate significant fretting.

Testing of a colorimetric procedure for continuous zirconium analysis indicates that concentrations as low as 1 or 2 ppb can be detected.

Preliminary design of the entire shim rod assembly is complete. Fabrication of components for two assemblies is estimated to be 90% complete, exclusive of procurement items.

Installation of the prototypical PRTR process tube in the EDEL-I test pit is complete. Preliminary test runs are in progress. The pressure tube seal testing station is approximately 60% complete.

Preliminary tests show that it may be feasible to enrich UO_2 fuel by inserting plutonium alloy wires into the UO_2 .

The vaporization rate of PuO was considerably greater than that of $\beta\text{-Pu}_2\text{O}_3$. Pneumatically impacted PuN-W cermet were quite stable towards vaporization.

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The melting points of mixed PuC-TaC alloys with greater than 50 wt% PuC were higher than 2400 C.

A plutonium carbide irradiation sample reached a burnup of 37,000 Mwd/ton. Additional capsules containing plutonium oxides and plutonium nitrides are ready for irradiation testing.

Nine capsules containing 1 vol% tungsten spheres randomly dispersed in sintered UO_2 pellets were prepared and shipped to NRTS for irradiation. The test is expected to identify structural features associated with central melting in UO_2 fuel elements that have been irradiated for various periods.

A newly developed computer code (CHEAT) reduced manhours required for computing radial temperature profiles of ceramic fuel rods (needed for irradiation test proposals) by a factor of ten or more.

Autoradiographs of cross sections of the electrical resistivity test element discussed in previous reports revealed a high concentration of fission products at the inner boundary of the columnar grain region.

Additional UO_2 -W cermet irradiations at cladding surface temperatures greater than 2000 C were conducted in the Hanford Snout facility.

The cooperative study with BMI on UO_2 thermal conductivity has been completed. The last samples measured received a four-year irradiation to a burnup of 1.3×10^{19} fissions/cc and showed a further small decrease in low temperature thermal conductivity.

Photographs taken by transmitted light through very thin sections of single crystal UO_2 reveal impurity sites and lattice defects.

A new magnetic force welding chamber was received. The new chamber provides approximately 50% more volume and a much larger viewing window than the present chamber. A nickel-chromium alloy, Incoloy-800, was successfully joined by this process.

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A can containing a layer of nine small containers filled with Be-UO₂ mixtures was pneumatically impacted. Basic compatibility studies of the mixture will be made. Transverse cut EGCR graphite samples irradiated at 800 C were found to expand slightly in the GETR's last irradiation period, thus reversing the trend of contraction observed up to this time. Samples irradiated at lower temperatures continue to contract, contraction being greater and increasing in the parallel cut samples.

Samples of EGCR graphite irradiated under a 800 psi tensile load showed less contraction than unirradiated samples.

Porous graphite processed to 1500 C had a thermal conductivity about twice that of material processed at 1000 C.

Samples of CSF and TSX graphite held at 800 C and subjected to different thermal-cycling treatment for 30 days showed essentially no dimensional change.

Apparatus for coefficient of thermal expansion measurement was tested at temperatures up to 800 C using CSF and pyrolytic graphite.

Lampblack-base graphite exposed incrementally to 2.1×10^{21} nvt, $E > 0.18$ Mev showed a smaller contraction rate during the last irradiation period, thus implying saturation of the contraction effect. Irradiation of high density graphite (2.1 g/cm^3) which was heat treated to 3550 C underwent greater parallel contraction and smaller transverse expansion than material heat treated to 3000 C.

The effect of self-shielding in the irradiation of boronated graphite is being studied by computational methods in the interest of improved experimental capsule design.

Additional boiling burnout data were obtained under fog cooling conditions (flowing mixtures of steam and water) for 19-rod fuel elements. An electrically heated test section was used where the 0.074-inch spacing between rods was maintained by small "warts". Comparison of the results with previous data indicated that the boiling burnout conditions were affected only moderately by

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the method of introducing the fog coolant to the test section and not at all by the use of "warts" instead of wire wraps to maintain spacing.

Testing of the dome seal has been completed with all requirements in the original test outline either met or exceeded. In addition, the dome deformation previously encountered has been eliminated and the seal has proved capable of withstanding a temperature rise of 400 F, in approximately 15 seconds, with no leakage detectable for 5 cycles.

An extension of the economics analysis of the fast oxide reactor - D_2O pressure tube reactor, fuel re-use system has been initiated. The first step is an examination of the fuel costs of the D_2O reactor as a function of the fuel service fee charged by the fast reactor (blanket) operator. The price of Pu is assumed to be \$10/g fissile, the economic interest rate is 12-1/2%, three levels of use charge (4-3/4%, 7%, 12-1/2%) are used, and the fuel service fee is varied from \$10 to \$40/# fuel. A comparison of minimum fuel costs for this fuel re-use cycle, a U^{235} cycle, and normal Pu recycle showed that the fuel re-use cycle is cheaper over the full range studied, implying that the thermal reactor could also pay the total fabrication costs for the blanket of the fast reactor and still generate power at a lower cost than a standard fuel cycle.

The most recent fuel cycle cost estimate for the new reference core design of the Fast Supercritical Pressure Power Reactor is 1.3 mill/kwh. For a similar 1000 Mw(e) reactor, the preliminary estimate of fuel cycle costs is between 0.7-0.8 mill/kwh. Burnup calculations for the 300 Mw(e) core were completed. Results indicate that power distribution can be held within tolerable limits throughout a batch core life. The required control span is about 45 mk for the current reference core, but it appears possible to reduce this by minor adjustments in initial enrichment.

Latest estimates by Advance Fuel Development Operation, CRDO, on fabrication potential of the "waffle grid" honeycomb cermet fuels for a Plutonium Fueled Rocket Reactor indicate that fuel plates up to four inches

thick, and the full diameter of the core (12-14 inches) may be feasible. A sizeable savings in pressure drop through the core should result, and support of individual plates may be made easier. Modified conceptual designs based on the thicker plates are being studied.

Additional multigroup calculations have been started to investigate other moderating materials which can be used in place of hydrogen for the segmented fast core concept. Initial results have been obtained for a steam-cooled fast core in which a graphite region is used to separate two fast regions. As expected, review of the data indicate that graphite may be effectively used but more is required to give the same void effect as a hydrogen moderating material. The multiplication of the reactor with normal coolant conditions is about 4 mk lower for a graphite moderator than a hydrogen moderator for the same void effect. Preliminary design of the HTLTR prototype is approximately 55% complete. Procurement of mockup materials has been started.

A preliminary proposal for a nonnuclear engineering test program on containment effectiveness was prepared at the request of the AEC. This is part of a test program involving construction of a Containment Test Facility (CTF) comprising a small-scale containment vessel and electrically heated reactor coolant system to study the consequences of piping and vessel failures.

The study of the delta-to-gamma isothermal transformations in plutonium was started. Sharp discontinuities in the density-versus-time curves were observed at the start of the transformation of metal known to contain impurities. Seven such discontinuities were observed at 101 C (213.8 F). No discontinuities have been observed in high quality plutonium.

The time-temperature-transformation (T-T-T) curves of the $\gamma \rightarrow \beta \rightarrow \alpha$ transformations of plutonium metal containing less than 250 ppm total impurities by weight have been completed. The pronounced surface distortion referred to as surface veining, commonly observed in unalloyed plutonium, has

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been identified as bands of columnar grain. These bands form during the first 50% of the $\beta \rightarrow \alpha$ transformation and are randomly distributed.

Molybdenum foils with exposures of 10^{19} nvt have been annealed at 750 C (1380 F) and then plastically deformed. Following deformation, the foils were thinned and examined in the electron microscope. Channeling of dislocations has been observed in the foil containing 450 ppm carbon. The channels are similar to those observed in the "as-irradiated" deformed foils. Examination of foils containing < 10 ppm carbon will be forthcoming.

High purity iron foils, obtained from the Johnson-Matthey Company, have been examined in the electron microscope following irradiation to 5×10^{18} and 1×10^{19} nvt ($E > 1$ Mev). Defect structures, due to the irradiation were not observed in any of the foils. However, evidence for dislocation pinning was observed after the lowest exposure. Similar irradiated foils are currently being annealed at 350 C (662 F) in an effort to cluster the defects produced by irradiation.

Electron microscopy of an as-extruded high purity uranium specimen irradiated to 0.03 at. % burnup at 525 C (977 F) revealed crystallographically aligned microtears that are due to the recovery of fission event induced damage. Essentially no fission gas pores are present.

Density measurements of uranium specimens enriched to 1.44 wt% U^{235} and irradiated in a capsule controlled at 625 C (1157 F) to 0.10 at. % burnup indicated volume changes of 8, 13, and 28%. This is somewhat higher than natural uranium specimens irradiated under the same conditions.

Optical microscopy of Th - 1, 4 and 5.4 at. % uranium tensile specimens irradiated up to 1.5 at. % burnup revealed no distinct changes in metallography due to the irradiation.

In-reactor creep tests of Zircaloy-2, 20% cold worked, at test conditions of 350 C (662 F), 25, 000 psi; 310 C (590 F), 35, 000 psi; and 330 C (626 F), 30, 000 psi yielded steady state creep rates during irradiation

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of 2.3×10^{-6} /hr, 4.4×10^{-6} /hr, and 6.4×10^{-6} /hr, respectively. These data combined with previously accumulated data on an Arrhenous plot may be used to predict creep rates of irradiated 20% cold worked Zircaloy-2 for both reactor operating and reactor down conditions. To further verify the description of Zircaloy-2 in-reactor creep, two new capsules were charged this month. Tests will be run in these capsules at 370 C (698 F), 25,000 psi and 320 C (608 F), 35,000 psi creep stress.

Changes in mechanical properties of AISI 304, 348, 410, and AM-350 stainless steels with neutron exposures from 3×10^{19} to 3×10^{20} nvt have been measured. Also, property changes after irradiations at 540 F (282 C) and below 200 F (93 C) have been compared. In general, changes in yield strength occur at a faster rate with neutron exposure at the higher temperature. Out-of-reactor loop exposures for times duplicating those of the G-7 loop irradiations cause significant changes in properties due to thermal effects alone. When thermal effects are accounted for, damage accumulation at 540 F (282 C) is in better agreement with that observed at ambient temperatures.

Flux dosimetry data from the G-7 core position for 7 cycles of irradiation have been obtained, and flux values based on iron, nickel, and titanium wires have been compared. Flux values based on iron and nickel are in close agreement (within 2%), whereas values based on titanium are higher by 7.5% at the peak flux of 10^{14} nv.

Twenty-two notched beam tests were performed on Zircaloy-2 during the month. Variables investigated to date using Zircaloy-2 include temperature, cold work, strain rate, and the direction of crack propagation. Studies to date reveal that plastic deformation strongly controls crack propagation even at liquid nitrogen temperature. It was also observed that brittle crack propagation occurs with greater difficulty at 45 degrees to the rolling direction than at 0 and 90 degrees, showing that fracture behavior is anisotropic.

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Thermal expansion stresses, anchor point reactions, and pipe element reactions were calculated for the latest configuration of primary and shroud piping of the model gas loop.

Pressurized helium at temperatures up to 2100 F (1149 C) in the model ATR high temperature gas loop will be contained by insulating a hot inner pipe from a pressure-containing outer pipe. A castable alumina-silica ceramic fiber insulation (Fiberfrax) has been selected and shape design completed with the aid of a digital computer program. Experiments show that outside coverings of insulating cloth coated with inorganic binder improve durability and internal strengthening members, such as stainless steel mesh, improve strength.

All N-Reactor charging machine testing was temporarily discontinued to ready the machine for shipment and to accomplish all of the punch-list items required for final completion. Preliminary design was started on the mockup which will replace the original charging machine.

A 19-rod Al-Pu fuel element test assembly was completed for the PRCF.

Seven cans, each containing 3.4 lb of 1.5 wt% of $\text{PuO}_2\text{-UO}_2$, were pneumatically impacted at 1200 C and 250,000 psi. Density of the impacted material was 10.92 g/cc (11.5% TD).

The first pneumatically impacted $\text{UO}_2\text{-1.5 wt% PuO}_2$ was processed through the new powder preparation line. Six hundred Zircaloy tubes were received. Vibrational compaction techniques were developed to permit three fuel rods to be simultaneously loaded to at least 86% of theoretical density.

Contract modifications with an off-site vendor for EBWR fuel rod Zircaloy cladding tubing were completed to permit nondestructive testing at Hanford. Preliminary ultrasonic and eddy current test revealed that more than 20% of the tubes may contain defects.

Thirty-six capsules incorporating EBWR fuel of a new composition, UO_2 (depleted)-1.5 wt% PuO_2 , are being fabricated for irradiation tests.

2. Physics and Instruments

The control strength of samarium ball channels, as measured in the exponential pile mockup of N-Reactor, is about 90% of the horizontal control rod strength.

The effect of neutron absorbing splines on the reactivity of N-Reactor was evaluated in a PCTR experiment. The splines' effects were measured in a wet process tube and in the cross coolant channels.

Analyses were completed on exponential pile data for an overbored C-Reactor lattice; the results: the lattice was slightly under-moderated and the vertical safety rod strength was independent of pile flooding or loss of coolant.

Experiments were completed in the PCTR to evaluate k_{∞} for a zirconium tubed K-lattice. Data analysis has not been completed.

Preparation of special instrumentation for the initial fuel loading and physics tests at N-Reactor continued. Specifications for the counting system were completed, bids were received on the neutron detectors, and tests began on the source range process instrumentation.

Simulation studies of the N-Reactor pressurizer are about half completed; controller stability tests compared favorably in the real and accelerated time runs.

Very interesting experimental data were obtained on the effect of neutron reflectors on the critical mass of an unmoderated mixture of PuO_2 and polystyrene. A reduction in critical mass by about a factor of three was observed when a bare system was compared with a system fully reflected with polystyrene. These experiments were conducted in the Critical Mass Laboratory using subcritical neutron multiplication techniques.

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A calculation has been made of the amount of plutonium which when added to natural uranium would make a chain reaction possible in an aqueous homogeneous system. It was estimated that 0.18% plutonium atoms (5% Pu²⁴⁰) in natural uranium would make a chain reaction possible under these conditions.

Satisfactory operation was achieved with the scintillation detector channel used for measurement of dynamic reactor parameters at the Critical Mass Laboratory.

Experimental measurements of the interaction of slow neutrons with light water molecules continued. Data are being obtained using 95 C water with better energy resolution than used previously.

Development work continued on a time-of-flight type neutron spectrometer which will be used for measuring the inelastic scattering of neutrons.

Work on the fast neutron cross section measurements program was directed toward processing the experimental data obtained in June on twenty-two elements.

The evaluation of control rods in plutonium-fueled cores was completed. The principal finding: there is a fairly sizable reduction in control rod worth in plutonium cores as compared to uranium cores. It was established that some of the simpler "approximate" methods of computation are adequate for survey-type calculations.

Code development continued. Alterations to the thermalization code SPECTRE to provide punched card input to other codes were completed. The code INTERSET which computes the interaction effects of arrays of cylinders containing fissile material has been completed. Work continued on the code for computing the bucklings of truncated spheres. A program for solving the thermalization equations, using a modified heavy gas model, in the manner used in the Harwell code SPECTROX, is being written. Program GE-HL Program-E, which computes the fission neutron energy and time spectrum,

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has been written. Work continues on the RBU code development. The testing of HRG (Hanford Revised GAM) has revealed a weakness in the computation of certain resonance integrals. Copies of GE-HL Program S-XI were sent to three other sites during the month; one went to Harwell, England.

Burnup data on three PRTR Pu-Al fuel elements is now complete, except for a few reruns for checking purposes. The analysis of the data continues. Preliminary results show the value of α^{239} to be smaller than the value previously obtained from the test element 5075. An experiment to discover the source of the difference is being run. Work on the burnup of natural UO_2 elements has begun.

Work is underway to evaluate fuel costs in relation to physics parameters for some typical power reactors to the physics parameters. These studies will encompass the cost to the fabricator (through warranty penalties) as well as cost to user (through direct fuel costs).

Data-theory correlation efforts to establish consistent and accurate cross sections and computing methods continued with the computation of effective multiplication factors of several heterogeneous and homogeneous critical experiments. An effort began to apply statistical methods to the results. The sensitivity of critical size and mass to the various cross sections presently available has been explored for "Phoenix"-type cores, and various fast reactor cores have been explored. Effective multiplication factors are shown to vary as much as 7% for a fixed core size (Phoenix-type), and the critical core size by as much as 1.5% in radius for some fast reactors.

Data previously obtained on PuAl (20.6% Pu^{240}) in graphite moderated lattice is being compiled for inclusion in a quarterly report.

Approach-to-critical measurements on a H_2O moderated lattice fueled with PuAl rods (16% Pu^{240}) have been completed for one lattice spacing, 0.66-inch.

The worth of a PuAl fuel element in the center of the PRCF was measured to be 14.7 mk relative to heavy water in the same location.

An experiment is being planned to measure the reactivity of highly irradiated PuAl samples in the PRCF. These samples were irradiated in the MTR.

Isotopic analyses were completed on samples from four fuel elements which had been irradiated in the PRTR.

Work continued on development work in support of the HTLTR. Studies were initiated on the corrosive effects that nitrogen and graphite may have on alloys planned for use in the HTLTR. A study has also begun on a mockup facility for development studies related to the HTLTR. Design of the HTLTR has been started by the Architect-Engineer.

Proper operation was attained on the analog one-node simulation of the Plutonium Recycle Critical Facility. The simulation used six delayed neutron groups and included provisions for scram tests and temperature feedback.

Successful on-line operation was achieved for the three liquid effluent monitors for gamma emitters at PRTR and all electromechanical drawings were completed.

The first regenerating (Phoenix) neutron flux detectors which were irradiated in the KE Reactor magazine facility were discharged and removed from their cans. They are being prepared chemically for mass spectrometer analysis. Satisfactory insulation resistance tests from 20 C to 650 C were completed for one B-11 β current detector.

Development of an improved single frequency eddy current tubing testing instrument was completed, and a prototype was successfully used in field testing of stainless steel tubes in the N-Reactor primary steam generators. Tube sheet region testing of the N-Reactor steam generators was materially improved with the completion of another special solid state eddy current tester.

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Improvement in the mapping of resistivities of samples of N-Reactor process tubes has been achieved with the completion of a special comparator circuit for use in the 450 kc eddy current hydride test instrument.

Satisfactory data comparisons were achieved with the prototype tester for 17-mil wall sheath tubing with the sample tubing both stationary and rotating. The electronic portions of the ultrasonic tester are now completed.

The low- and intermediate-energy neutron detectors for the solid state neutron personnel dosimeter were tested with satisfactory results. Moderated Li^6 and Rh^{103} detecting material is used for the two detectors, respectively.

Operational safety of Biology radioactive aerosol experiments was enhanced with the completion and installation of a special chamber air balance monitor and alarm instrument. The instrument will alarm if the air balance differential exceeds 0.1 inch of water equivalent.

A working experimental model of an instrument for monitoring thoracic air pressures in Biology animals was completed and successfully tested in the laboratory.

Analog computer simulations were developed to aid the study of iodine decay and tellurium activities in the bodies and milk of sheep.

The first reports were received from the field party which is measuring radioactivity of Eskimos in Alaska this summer with the portable whole body counter. Counting at Anaktuvuk Pass was completed. Body burdens of Cs^{137} there have increased about 25% over those measured last summer.

Further evaluation of rain scavenging data continues to show a peak in scavenging efficiency in the region of 0.4 mm drop diameter. It also shows a rapid increase of scavenging efficiency with increasing particle size and density. Both findings are at variance with values currently used to predict "washout" from contaminant clouds in the atmosphere.

In a cooperative effort involving several components of the Hanford Laboratories working in the radiological sciences, the first field release of I^{131} occurred on July 25, 1963. Approximately 100 mc were released over an atmospheric diffusion grid operated by Atmospheric Physics. Early indications are that all test objectives were met and that pretest estimates of dosage were good, marking an important step in the use of radioactive materials for further experimental studies in the environment.

Air filter counting for radioactivity was improved with the completion and delivery of two scintillation coincidence counters--one unit for use in the 325 Building and the second scheduled for use at Fuels Recycle Pilot Plant.

A feasibility study of a conceptual structural analysis machine (SAM) was completed for Metallurgy Operation, and the results were documented. In addition to metal structure studies, the technique employed would be useful to biologists and others interested in gross feature measurements of objects in visual fields.

Successful traversing of two tubes at D Reactor was achieved with the Electrical Readout Traverse Mechanism. Information on distortion was believed accurate to ± 0.2 inch.

Capability for low power pulsing of laser rods has been achieved in Optical Shop experiments. In addition, measurements with phosphor doped glasses have shown one type to be markedly resistant to darkening caused by high level (3×10^6 r/hr) gamma fields.

3. Chemistry

Substitution of aluminum nitrate for aluminum sulfate as the flocculating agent--to minimize the $S^{32}(n, p) P^{32}$ route for production of P^{32} --is under in-reactor evaluation in the experimental water treatment plant. Results show the anticipated reduction in P^{32} levels in the effluent water and also reductions in levels of other effluent radionuclides.

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The new and versatile multidimensional gamma spectrometry technique exhibited significant advantages over former techniques. For example, in analysis of reactor effluent water, As^{76} , Cu^{64} , Ba^{140} , La^{140} , Co^{60} , and Sc^{46} were determined simultaneously in a single measurement on a single sample.

Use of a portable multichannel analyzer and detector system provided field measurement of radionuclides with high sensitivity and accuracy. For example, fallout Zr-Nb^{95} , Ru^{103} and Ru^{106} were measured from a boat in the Snake River, and in the Columbia River upstream of Hanford. In the atmosphere Zr-Nb^{95} , Ru^{103} , Ru^{106} , Ce^{141} , Ce^{144} , and Bi^{214} were measurable up to the maximum altitude (12,000 ft) surveyed.

Laboratory experiments continue to show the applicability of trilaurylamine extraction to the recovery of neptunium and plutonium from Purex waste solutions.

Isotopic analyses of strontium in Purex waste tanks and chemical analyses of Purex essential materials indicate the high natural strontium contamination in certain Purex waste resulted from use of caustic with a high-strontium content. Fortunately, the bulk of the caustic used since the start of strontium production activities had a low strontium content.

A new solvent extraction technique which allows all rare earths to be extracted in trivalent form and then employs an oxidizing stripping solution to retain cerium(IV) in the organic phase while removing the other rare earths to an aqueous phase was tested on laboratory and hot cell scales and is scheduled for testing in the Strontium Semiworks in the near future.

Pressure drop measurements were made with several mesh sizes of Duolite C-3 resin and synthetic Redox alkaline supernatant waste to determine the minimum feasible size for use, based on a balance between pressure drop and kinetics. Both 20 to 50 and 20 to 60 mesh resins appear to be satisfactory.

The concept of using D2EHPA for "total" extraction of dissolved alkaline waste sludge was tested on a pilot plant scale. Although iron extracts nearly quantitatively, it remains in the solvent on stripping out the strontium with 0.03M HNO_3 , giving promising product purities. The iron is removed from the solvent with alkaline washes.

Linde AW-400 zeolite appears to have favorable properties for cesium removal from Purex aqueous sludge leach solution.

Two mechanisms for loss of heat transfer in titanium heat exchangers appear to exist. One of these is the deposition of a thick, silica scale. The other involves the deposition of an invisible film which changes the boiling characteristics on the surface.

Salt cycle experiments indicate that there is no "cut off" concentration which limits the degree of removal of plutonium which can be obtained by electrolytic codeposition of UO_2 and PuO_2 . The plutonium concentration in the melt was reduced to only 0.002 wt% in one experiment and 99.8% removal of plutonium was effected in another experiment (along with removal of 72% of the uranium).

Neutralization of Purex waste with ammonium hydroxide is less effective than is neutralization with sodium hydroxide in reducing the volatilization of ruthenium in subsequent spray calcination. Also, only about 3% of the added ammonia was recoverable in form suitable for re-use, thus voiding one previously assumed advantage for ammonia over caustic.

The capacity of the eighteen-inch radiant-heat spray calciner for water evaporation increased linearly with the square root of atomization pressure, increased with the third power of the absolute temperature of the process, and was 5 to 8% higher with an internal-mix nozzle than with an external-mix nozzle.

Design and fabrication of the auxiliaries of the Waste Solidification Engineering Prototype is well advanced, and current effort is directed toward the detailed design of the solidification machines.

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Results of the first application of high temperature, solid-state electromigration as a means of purifying plutonium metal (without the necessity for melting it) indicate migration velocities in plutonium are substantially lower than in cerium and should make the technique unattractive for routine plutonium purification.

A method was developed to enable the tracing of ground water flow paths by a computer technique which numerically steps along paths of steepest descent with respect to the potential function. Results are being compared with specific closed-form solutions to learn the accuracy of the method.

4. Biology

Since Dowell-F33 (along with trisodium phosphate) may be used at N-Reactor instead of Vertan 690, further toxicity studies were performed. Preliminary results with Dowell-F33, which contains an alkyl phenol as one of its active constituents, showed total mortality of young trout with 60 ppm within 40 minutes (compared with an LD₁₀ for Vertan 690 of 2000 ppm).

The mortality pattern from columnaris appears to be more variable this year with no sharp mortality peak observed in July (as occurred last year). Raising the trough temperature 4 F increased mortality sevenfold, due probably to an aggravated columnaris problem.

In studies using the leukocyte response to intravenously injected endotoxin as a test of the granulocyte reserve, it was noted that miniature swine fed 125 μ C Sr⁹⁰/day for over two years responded subnormally.

Three weeks after stopping the feeding of 2 g stable iodine/day to 2 cows (but continuing daily I¹³¹ feeding), the thyroid concentration of I¹³¹ had increased about tenfold and milk at least twofold.

Dogs on the long-term Pu²³⁹ experiment which have manifested low lymphocyte levels for 2-3 years failed to show any depression in their antibody

response as revealed by antibody titer studies of common canine diseases for which vaccination is accomplished.

Preliminary exposure studies of properly shielded rats in the Physical Constants Test Reactor established the feasibility of using this reactor in studies of neutron effects on the gastrointestinal tract. (Insect irradiation in this facility is also continuing.)

Data from field studies of plants, using dye and isotope traces, suggest greasewood may be useful in studying movement of H^3 through groundwater.

5. Programming

The soft spectrum of the Hanford production reactors is desirable to minimize Pu^{240} formation in Pu^{239} , but it is a disadvantage in making the transplutonium isotopes (e. g., Pu^{238} , Cm^{242} , and Cm^{244}). The spectrum could be hardened, according to recent plutonium recycle calculations, by homogenizing the fuel and water coolant. Although water in the process tubes constitutes additional moderator which normally tends to soften the spectrum further, mixing of the water and fuel in several annular rings also affords generation of additional epithermal neutrons, especially in the energy region to enhance curium production from Pu^{239} . Analyzing this problem properly requires considerable computer effort. Preliminary to this, several nonrigorous assumptions were made to allow estimation of this effect with the ALTHAEA computer code (also nonrigorous). The resulting data support the thesis and it appears that doubling the epithermal flux for a given plutonium load may be possible. This would increase the production of curium isotopes by about one-fourth. Calculations with better codes now appear necessary if it is desired to refine the results and to establish whether severe reactor operation (control) problems may be encountered.

AEC approval was obtained for the use of higher enrichments of PuO_2 - UO_2 fuel elements in the PRTR. The use of 1% plutonium is authorized

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providing the reactivity does not exceed that of a Mark I plutonium-aluminum fuel element containing 330 grams of fissile plutonium. AEC approved the use of long-cooled irradiated fuel in the Plutonium Recycle Critical Facility.

Evaluation of a drop test of a model of the HAPO I-B cask was completed and documented (HW-77962, "Drop Test of an Unbuffered 1/4 Size Model HAPO I-B Cask"). The results indicate that the cask would not rupture or release its contents if dropped during routine operations without the protective buffer. This information will be transmitted to the Technological Hazards Council in response to a recent recommendation.

A formal report, HW-77770, "The Hanford Isotopes Production Plant Engineering Study, " was completed and released to the AEC.

To permit more active conduct of the nuclear engineering standards work at national levels, the following have been completed and distributed:

1. A review of past, present, and proposed nuclear engineering standards activities
2. An up-to-date list of HAPO representatives on such committees
3. A revision of "Guides for HAPO Employees, " first issued in June 1959.

In response to an invitation of the Division of Reactor Development, a preliminary research proposal was transmitted to AEC-RLOO (HW-78354) for a Containment Test Program. The Commission's primary interest is in testing components of containment systems for water-cooled power reactors. Their main objective is to improve reactor siting criteria.

TECHNICAL AND OTHER SERVICES

Three new plutonium deposition cases, each estimated to be less than one percent of the maximum permissible body burden (MPBB plutonium with bone as reference is 0.04 μ c), were confirmed by special bioassay analysis during the month. Two of the incidents occurred in the Weapons Manufacturing Building (234-5) and one in the Critical Mass Laboratory (209-E). The total number of individuals who have received

internal plutonium deposition at Hanford is 321. With the reactivation of one employee, the number of deposition cases currently employed is 233.

Analysis for IPD of frequency of outage work has been completed and submitted to K- and H-Area managers. Initial response has been favorable. The analysis was also presented to the Manager, Finance, N-Reactor, to illustrate a potential aid for N-Reactor control problems.

A systematic sequential sampling plan was developed to determine whether the percentages of defective tubes in the different NPR heat exchangers are unacceptably high. Further assistance was given in establishing confidence limits on these percentages for various heat exchangers on the basis of the sample results.

A study is being made to determine whether there are significant differences between the stud pull results for three cleaning processes on fuel elements canned by the hot-die-sizing process.

A comparison is being made between the pull strength of rails made with the transverse grain and with the longitudinal grain. A test is also being analyzed which compares soap and "3-in-1" oil for cleaning in the ultrasonic welding of rails.

To provide a more meaningful description of actual inspection practices, a sample of current production is being followed through the entire residence path in the inspection facility. This basic information is needed to permit an effective method of recording and processing inspection data, to optimize inspection procedures, and to estimate the reliability of the inspection system.

In response to a request, an overall review of the statistical quality control of sampling and volume measurements by CPD is to be made.

The first experimental magnetic tape made on the 7090 for the purpose of numerically controlling the prototype Sheffield rotary contour gage has

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been completed. This tape is to be tested on the mockup model being assembled at the Schenectady laboratories of Equipment Development Services.

Further work was done on the regression analysis of plutonium homogeneous spherical critical mass experimental data to determine the relative merits of several reactor physics code systems for the theoretical calculation of eigenvalues and critical masses. The regression analysis technique is being extended to a set of Yankee critical mass experimental data and a set of plutonium-aluminum critical mass experimental data.

Preliminary results from the analysis of the spherical experiments indicate that four-group macroscopic cross sections should probably be used in place of two-group cross sections. Current efforts are directed toward preparing the four-group cross section data and entering it into the regression analyses.

Because of the lack of appropriately sized sieves, particle sizes corresponding to those called for by the often successful regular cylindrical packing model have not been available for a current fuel element model. Since difficulty has been experienced in obtaining sufficiently high densities with the available sieve sizes, particle sizes and mass ratios have been based on a recently developed (and incomplete) cylindrical-spiral packing model.

A new input subroutine was written for program GEM and is now being debugged. This input routine will read "unsummed spectra" on cards and do the necessary summing required by the GEM program. The routine will also accept cards of the previous format and print a listing of card input in each case. Primary purpose of the new input routine is to make the GEM program available as a routine research tool to laboratory radiochemists without the complexities of going through the IRA system.

Data were analyzed from $\text{Ce}^{144}\text{-Pr}^{144}$ inhalation experiments performed on dogs. The purpose of the analysis was to express the whole-body retention of $\text{Ce}^{144}\text{-Pr}^{144}$ after inhalation of oxide particles as a function of time and to compare the statistical models for different dogs.

SUPPORTING FUNCTIONS

The Plutonium Recycle Test Reactor's output for July was 929 Mwd, for an experimental time efficiency of 51% and a plant efficiency of 41.6%. There were nine operating periods during the month, of which five were terminated manually and four were terminated by scrams. A summary of the fuel irradiation program as of July 31, 1963, follows:

	<u>Al-Pu</u>		<u>UO₂</u>		<u>PuO₂-UO₂</u>		<u>Other</u>		<u>Program Totals</u>	
	<u>No.</u>	<u>Mwd</u>	<u>No.</u>	<u>Mwd</u>	<u>No.</u>	<u>Mwd</u>	<u>No.</u>	<u>Mwd</u>	<u>No.</u>	<u>Mwd</u>
In-Core	15	1106.9	10	1445.0	60	3865.9			85	6417.8
Maximum		114.3		192.7		103.3				
Average		73.8		144.5		64.1				
In-Basin	28	1850.7	56	4047.1	13	296.3	1	7.3	98	6201.4
Chemical Processing	32	2309.3	1	78.0					33	2387.3
Program Totals	75	5266.9	67	5570.1	73	4162.2	1	7.3	216	15006.5

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

A total of 136 reactor outage hours were charged to repair work.

Main items were:

E-7 valve replacement	- 68 hours
Helium system repairs	- 40 hours
Change out of breakers in MCC-IE	- 9 hours

The July heavy water inventory (for a 28-day period) indicates a loss of 544 pounds.

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The Pu-Al substitution experiment in the Plutonium Recycle Critical Facility was completed without incident. A reactivity measurement for Pu fuel samples was started.

The modifications to facilitate irradiated fuel handling were completed.

Design testing for the Fuel Element Rupture Test Facility was completed during the month. The last test was a measurement of system response to a complete power failure in which the system flow rate decreased from approximately 200 gpm to 20 gpm; during this transient the system decreased from 1800 psig to 1675 psig, which is acceptable.

The blower testing in the Gas-Cooled Loop Facility was completed successfully, with over 200 hours of operation and numerous starts. Parallel operation and successive starting of two blowers were achieved without incidence.

Technical Shops's total productive time for the period was 21,327 hours. Distribution of time was as follows:

	<u>Manhours</u>	<u>Percent of Total</u>
N-Reactor Department	2072	9.75
Irradiation Processing Department	2903	13.61
Chemical Processing Department	248	1.14
Hanford Laboratories	16,097	75.47
Hanford Utilities and Purchasing	7	0.03

Total productive time in Laboratory Maintenance Operation was 15,500 hours of 18,500 hours potentially available. Of the total productive time, 86% was expended in support of Hanford Laboratories components with the remaining 14% used in providing service for other HAPO organizations.

Manpower utilization for July was as follows:

A. Shop Work	2400 hours
B. Maintenance	5800 hours
1. Preventive Maintenance	1700 hours
2. Emergency or Unscheduled Maintenance	1600 hours
3. Normal Scheduled Maintenance	2500 hours
4. Overtime (Included in above figure)	1000 hours
C. R&D Assistance	7300 hours

Heavy water loss during July totaled 564 pounds (\$7,834). The 1,199 pounds of scrap generated during the month resulted in a charge of \$2,621 to operating costs. At the end of July, the total of heavy water scrap accumulated for shipment to SROO was 10,867 pounds (\$135,648). Fifty-five drums (27,445 pounds) of new heavy water were received July 29.

Plant tours decreased markedly in July, but interest in the Visitors Center remained high at an average of 107 visitors per viewing day.

	Number of Visitors	
	<u>In July</u>	<u>Since 6-13-62</u>
Visitors Center	3310	51,090
Plant Tours	75	na

HAPO professional recruiting activity this month is summarized below:

	Plant Visits	Offers Extended	Acceptance Received	Rejections Received	Open Offers at Month End
Ph. D	1	2	1	3	9
BS/MS (Direct Placement)		9	6	6	1
BS/MS (Program)		0	1	3	0

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Seven technical graduates were placed on permanent assignment. Twenty-three additions to the program were accomplished and one member terminated. The total number of graduates on the program at month end was 75.

Authorized funds for ten active projects totaled \$6, 735, 000. The total estimated cost of these projects is \$10, 640, 000 of which \$836, 000 had been spent through June 30, 1963.

Manager, Hanford Laboratories

HM Parker: JEB: mhs

REACTOR AND FUELS LABORATORY MONTHLY REPORTJULY 1963TECHNICAL ACTIVITIESA. FISSIONABLE MATERIALS - 02 PROGRAM1. Metallic Fuel Development

Fluted Fuel Element Irradiations. Two fluted N-Reactor size, single-tube fuel elements are currently undergoing the fifth cycle of irradiation in the ETR M-3 facility. Density measurements made on the elements after four reactor cycles of irradiation (800 MWD/T) indicated that a volume increase of approximately 0.3% had occurred. This corresponds to the expected theoretical minimum volume increase from fission fragment production. These fuel elements are currently operating at maximum specific power of 130 kw/ft with a corresponding maximum fuel temperature of 520 C.

Target Element Development. Irradiation testing of 16 experimental lithium-aluminum target elements in the KER test loops has continued. Early this month a leak in Loop 1 was repaired and additional heater elements were inserted in Loops 1 and 2, which permitted operation at test design levels of 1600 psi coolant pressure and a calculated target element temperature of 300 C.

Evaluation of the "Magneformer" has continued. Satisfactory swaging of 5/8 " OD x 9" long 61-S aluminum alloy tubing onto 2-S aluminum cores has been achieved. X-ray and subsequent metallography indicate that the separation between the core and the clad was less than 0.0001". Studies are also under way to evaluate the use of this machine in developing improved end cap closure techniques for the target elements.

Pure aluminum (99.99%) was melted with pure lithium (99.9%) to obtain a high purity core material for diffusion studies. Metallographic examination indicated that segregation had occurred along the grain boundaries of the second phase, LiAl. The desired composition was 1.3% Li which is well within the solid solution range. A number of heat treatments were made to determine the dissolution time and temperature; however, the metallographic results are not available at this time.

N-Reactor Fuel Support Development. Annealing studies on N-inner supports have shown that it is possible to eliminate the "texturing"

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of the Zircaloy-2 that results in a high degree of orientation in the hydride platelets that form during autoclaving. Supports which were annealed at 750 C and 700 C retained their ductility after autoclaving. Hydrides were visible in the microstructure but were randomly oriented and located at grain boundaries. Additional annealing studies are being conducted to determine the annealing temperatures which will best eliminate the orientation created by forming process while minimizing grain growth during annealing.

Fatigue tests were initiated to establish the effect of hydrides on the fatigue life of inner supports. Initial results indicated that autoclaved supports have a slightly lower fatigue resistance than as-fabricated supports.

N-Reactor Fuel Evaluation. A detailed examination has been made in the Radiometallurgy facility of "buggy spring" inner supports from N-inner fuel components exposed to 1500 MWD/T. The supports, which were observed to be in good mechanical shape, contained 54 ppm hydrogen after irradiation indicating that little or no hydrogen pick-up had occurred during irradiation. Metallography of the Zircaloy-2 supports verified the hydrogen content and showed that hydride platelets in the supports tended to be oriented normal to the surface on the inside bends and tended to be oriented parallel to the surface on the outside bends of the spring supports. No indications of inadequate support performance were evident. Irradiated N-fuel components, including the Be-Zr closures, coextruded bonds, and braze heat-affected zone and Zircaloy-2 clad continue to show good irradiation performance.

Cladding Deformation Studies. Thirty-six NaK capsules containing a total of 94 Zircaloy-2 clad uranium rods have been irradiated to provide data on cladding strain capabilities as a function of cladding thickness uniformity, temperature, and exposure. Nine of these capsules which were discharged at 1100 MWD/T have been opened, and the 25 fuel samples have been photographed, the diameters measured, and the samples visually examined.

The observed cladding strain data agrees with previous data indicating that below a mean cladding temperature of 325-350 C the strain limit before plastic instability or necking of the cladding occurs is approximately 1.5% and above cladding temperatures of 350 C the strain limit increases rapidly.

The unstriated alloy fuel samples of U - 2 w/o Zr exhibited an ability to withstand a greater total cladding strain without necking than the unstriated, unalloyed samples. This may be related to

differences in properties of the alloyed and unalloyed uranium fuel. That is, the fuel itself may influence the strain distributions in the cladding.

The data from samples having striations in the cladding show the strain limit is significantly reduced when the depth of the striation is approximately 0.004" (15% of the nominal clad thickness). The striations in the specimens thus far examined are arcs of a circle with a radius of approximately one-eighth inch. Metallographic study of the necked region of the cladding on one sample shows that the decrease in clad thickness is directly proportional to the depth of the external surface striation developed in the necked area. From measurements made on photomicrographs of transverse sections of a striation, it was found that 15% of the total cladding strain had been concentrated in the striation, with the localized cladding strain being approximately 20 times the average cladding strain. Examination of these fuel samples will continue. The remaining 24 capsules, discharged at an estimated fuel exposure of 1800 MWD/T, are being returned to Radiometallurgy for examination.

N-Reactor Fuel Support Development. A method of analysis has been formulated to determine the temperature rise under fuel element supports as a function of support tab geometry and assumed heat transfer conditions in the support area. The method of evaluation is based on a numerical technique where only the conditions on the surface containing the support must be numerically satisfied. The method of analysis has been programmed and debugging the program has begun.

Fuel Component Development. Laboratory tests to study possible damage to N-Reactor process tubes by charging N-fuel elements are proceeding. A series of tests are being conducted to determine to what extent the reactor process tube is damaged by the repeated corrosion of the bare Zircaloy-2 that is exposed by scratching by the fuel element supports. The test procedure consisted of an autoclaving cycle in 400 C, 1500 psi steam for 14 days between each series of 40 passes of fuel element supports on a process tube section. From these tests the rate of wear was determined to be 0.00035 to 0.00045 inch per 40 passes. This rate of wear extrapolates to 0.018 to 0.023 inch in 2000 passes as compared to a maximum of 0.003 inch wear in a test in which 2000 passes were made on a tube without intermediate autoclaving.

There may be two sources for the difference. The removal of the autoclave film formed between each series of passes may result in deeper total scratch depth in the cyclic scratch autoclave test,

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or the autoclaving cycle at 400 C may cause recovery of some of the cold work thought to be induced in the surface of the tube during the scratching tests.

To determine what the possible annealing effects are in the autoclave tests, a process tube section is being tested by making 500 passes with steel fuel element shoes followed by a heating cycle at 400 C for 14 days in a vacuum. After the first 500 passes, the average scratch depth was 0.001 inch. Two subsequent runs of autoclaving at 400 C for 14 days in vacuum, each followed by a series of 500 passes with steel shoes, has not resulted in any further increase in scratch depth. This indicates that cold-working of the surface of the tube reached a maximum by 500 passes and that annealing at 400 C for 14 days is not causing recovery.

A second similar test is being made on a process tube section in which only 50 passes are made followed by a heating cycle of 400 C for 14 days in vacuum. The data are not sufficiently complete to determine a difference in the scratch depth for this test as against the test where the 400 C, 1500 psi autoclave cycle was used. The data currently appear to be confirming the results obtained in the above annealing study. At this time test data indicate that the accelerated wear rate observed during repeated autoclaving of the tube is associated with the autoclave film.

Hot Heading N-Outer Fuel. Tooling has been fabricated for hot heading short lengths of N-outer tubes and several lengths are being headed, for use in welding development work on the hot headed bonded closure. A different heading technique appears necessary for the larger outer tubes than was used on the N-inner fuel to form the Zircaloy-2 cladding over the shoulder. Otherwise, the cladding folds below the shoulder or shears across the shoulder. A small entrance angle on the heading dies appears to alleviate the problem; however, a subsequent operation will be required to square the ends of the headed tubes for welding.

Strip Forming. Buggy spring strip has been successfully made from one-inch diameter extruded rod. The rod was cold swaged from 1" to 0.300" in diameter with intermediate anneals at 600 C after each 40% reduction in cross section area. These anneals were made in the salt bath at 600 C for 10 minutes and then water quenched.

After swaging, the rods were turk's headed to the finished size strip having a cross section of 0.038" x 0.253". Again, salt bath anneals at 600 C for 10 minutes were made after each 40% reduction. However,

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the final anneal was made in air at 600 C for 10 minutes. This material passed all bend test requirements.

It is now established that a good economical supply of Zr-2 material exists. This material can be formed into strip with less than a 10% scrap loss as compared to the 30% loss entailed in the rolled sheet method. Reported tests indicate that this turk's headed material passes all autoclave tests and does not contain an undue amount of hydrogen.

Closure and Joining. Corrosion tests in 360 C water have continued on the alloy coupons of Zircaloy-2 + beryllium in excess of 1300 hours, on Zircaloy-2 + uranium in excess of 800 hours, and Zircaloy-2 + uranium + beryllium in excess of 200 hours. The following generalizations can be made:

1. Five percent Be - Zr-2 has a weight gain of six times that of Zr-2, and the color continues to turn whiter.
2. After 800 hours Zr-2 + $1\frac{1}{2}\%$ U has about the same weight gain as Zr-2, and Zr-2 + 1% U has a weight gain about $1\frac{1}{2}$ times that of Zr-2. The colors are still glossy black.
3. The Zr-2 + Be + U series shows very little difference in weight gain with uranium content; however, the weight gains increase in proportion to the beryllium content. The 5% Be + Zr-2 - 2500 ppm uranium had a weight gain about 10 times that of Zr-2. In each case the color is white.
4. Another series with oxygen and nitrogen additions to 5% Be + Zr-2 show very high weight gains and had a white color after only 100 hours. Both nitrogen and oxygen have a horrible effect on the corrosion properties resulting in weight gains in excess of 10 times those for Zircaloy-2.

The present tests will continue until the coupons disintegrate or show signs of complete failure. Additional corrosion alloys have been made to test the effect of iron and carbon on the corrosion properties of Zr-2 + 5 Be, and another series will be made to determine the difference in corrosion properties of Zr-2 + 5 Be and zirconium + 5 Be.

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Irradiation of Thorium-Uranium Fuel Elements. Three tubular Zircaloy-2 clad thorium - 2.35 w/o U-235 - 1.0 w/o Zircaloy fuel elements are completing their third cycle of irradiation in the ETR P-7 loop. At the end of the cycle the fuel elements will have achieved an exposure of 2400 MWD/T. Weight measurements made at the end of the second cycle indicated that no detectable volume increases had occurred and no visible changes in the fuel elements had occurred. The test elements continue to operate near design conditions. The maximum specific power is 63 watts/gm (190 kw/ft) with a corresponding maximum heat flux of 72 cal/cm²-sec (950,000 Btu/ft²-hr). The maximum fuel temperature is calculated to be 555 C.

Fabrication of Thermocoupled Fuel. A solid rod, 1.030" OD with 0.028" Zircaloy-2 cladding and a 0.470" diameter Zircaloy core piece with a Th - 1.5% U (fully enriched) - 1.0% Zr fuel annulus between cladding and core, has been successfully coextruded. This material will be used for a long life constant power heat source for heat transfer studies. Pilot stock of the same dimensions and composition (natural uranium) was reported last month.

The extrusion temperature of 760 C was the same as the pilot stock as were all other conditions. A slightly higher extrusion constant of 18.7 TSI was obtained as compared to 17.2 for the pilot stock.

Visual Autoclave Facility. A visual autoclave facility has been installed in the 306 Building to be used in defect testing of small Zircaloy-2 clad thorium fuel samples. The facility has a windowed chamber one-inch in diameter by five inches deep and is capable of 300 C, 1500 psig operation. Direct visual observations as well as time-lapse photography is permissible during exposure of the specimen. Provisions have been made for monitoring the volume of entrained effluent gasses. Gas evaluation is used in the determination of corrosion rate of the core material.

A program has been initiated to compare the rupture behavior of Th - 2.5 w/o U - 1 w/o Zr with that of unalloyed uranium. Rupture specimens consist of 0.525-inch diameter x 2-inch long rods "defected" with a 0.025-inch hole through the Zircaloy-2 cladding.

In-Reactor Generation and Migration of Tritium in Al-Li Alloys. Tests are under way to investigate the migration of tritium in aluminum-lithium alloys at higher temperatures than those typical of Hanford production reactors. An aluminum - 1% lithium target assembly is being irradiated in the K-East Reactor. The assembly includes 16 Al-Li disks, $\frac{1}{4}$ inch thick by 1.4-inch OD, with a

$\frac{1}{2}$ inch center hole. The target assembly is contained in an aluminum capsule, which is connected by an aluminum tube to a vacuum system located on the X-1 level of 105 KE Building. The aluminum capsule includes two concentric tubes, with the target assembly inside the inner tube. The annuli between the inner and outer tubes and between the outer tube and the reactor test hole are pressurized with helium. Gas samples are removed through the vacuum system for mass spectrometer analysis.

At temperatures below 360 C, no tritium release has been observed up to 4.5 GVR*. Raising the temperature to 390 C resulted in tritium evolution which was barely detectable by mass spectrometer analysis (0.03% in a helium-tritium mixture). Calculations to determine the evolution rate are not yet complete.

A replacement capsule has been fabricated with water cooling coils and two sets of heating coils to provide flexible temperature control. Analysis of the capsule gas on heating the capsule with and without the reactor in operation will provide evaluation of the effect of neutron radiation on tritium migration.

2. Corrosion and Water Quality Studies

Rotating Disk Corrosion of Aluminum. To investigate velocity-related factors in the corrosion of aluminum, accelerated tests are being conducted using a rotating disk apparatus. Four-inch diameter disks of 1245 and X-8001 aluminum alloys are rotated at ~ 70 ft/sec (4150 rpm) in water. In three-day exposures at 90-95 C (194-203 F), an attack has been observed which resembles groove pitting seen on aluminum tubes and fuel elements after exposure in-reactor. The pitting attack on the disks shows a definite relationship to velocity; the attack is pronounced at the outer periphery and tapers off toward the center of the disk. Surface scratches on the disks act as initiation sites for pits; rows of pits are observed to follow surface scratches. Pitting has been observed on both aluminum alloys in 300 Area water.

Observations to date indicate a strong dependence of pitting attack on water quality. Tests in deionized water with 10 ppm chloride ion (as NaCl) at pH 6.6 and pH 8.8 failed to produce pitting in three-day tests at 90-95 C (194-203 F). No pitting occurred in a single test in K-West Reactor process water. Pitting in 300 Area water, though unpredictable, has occurred in several tests on both

*Gas Volume Ratio = volume of helium plus tritium at STP per unit volume of alloy.

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alloys. This may be due to periodic variations in 300 Area water quality. Preparation of synthetic waters is being considered to avoid uncontrolled changes in water quality.

Corrosion of Aluminum at High Water Velocities. Plans have been made to use in-reactor facilities on the X-2 level of KW-Reactor to study the corrosion of aluminum coupons in high velocity process water. A coupon holder has been designed and is being fabricated to determine the effect of high velocity on certain forms of nonuniform attack encountered in-reactor. Preliminary runs showed that the X-2 heat exchangers were fouled and flow rates were very low. The exchangers will be cleaned with inhibited sulfuric acid.

Evaluation of NH_4OH as Replacement for LiOH for pH Control.

Operation of KER-1 at pH 10 with ammoniated coolant was continued to evaluate NH_4OH as an alternate for LiOH . During the past month the loop operated in-reactor at minimum temperature approximately 50% of the time and at operating temperature approximately 25%. The loop was operating on reactor bypass during the remainder of the period.

Radiolysis of the coolant (water) was not a significant problem during the low temperature operation; gross amounts of oxygen and hydrogen peroxide were not encountered. However, trace concentrations of hydrazine were observed in the coolant during this period as the result of radiolysis of the ammonium hydroxide. It is possible that the presence of this and/or other reducing chemicals may have inhibited radiolysis of the water. In any case, ammoniated systems appear to offer some significant advantages over lithiated systems for low temperature operation, since appreciable radiolysis of the water is normally encountered in lithiated systems under these conditions.

In general, the coolant samples were slightly turbid during the low temperature operation. However, this turbidity was attributed to the low temperature rather than the ammoniated coolant since turbid coolant conditions are also encountered in lithiated systems during low temperature operation. Some crud deposition may be encountered on the fuel elements as the result of this turbidity. During the relatively brief period of high temperature operation, the coolant samples showed less turbidity than during the low temperature period except for the interim period when the coolant temperature was increasing. High crud release rates during this operation are also encountered with lithiated coolants. During operation at high temperature the coolant quality was satisfactory

except for the high initial turbidity. High temperature operation did not continue long enough to determine the equilibrium turbidity level, however.

Activity levels in the KER-1 and KER-2 loops are also being monitored as part of the program of determining the feasibility of using NH_4OH as the pH conditioning agent in the N-Reactor primary system. The KER-1 Loop (carbon steel) is conditioned to pH 10 with NH_4OH and the KER-2 Loop (stainless steel) with LiOH . Readings are taken 8, 24, and 72 hours after startup and shutdown, and once per week during normal operation. Routine monitoring of the loops to date has demonstrated:

1. No major radiation hazards have been encountered while operating with NH_4OH .
2. During startup the radiation levels in KER-1 (conditioned with NH_4OH) are considerably lower than those in KER-2 (conditioned with LiOH) by an approximate factor of 2-3.
3. During high temperature operation the radiation level in KER-2 is approximately 1.3 times greater than in KER-1.

Crud Formation on Fuel Elements. Five Zircaloy-clad fuel elements from a previous charge in KER-1 exposed to pH 10 NH_4OH were sent to Radiometallurgy for removal and analysis of the crud which had deposited on them. A mixture of sulfuric and oxalic acids was found to be the most effective of several reagents tested to remove the crud. Fax-Film samples of the crud were obtained for x-ray diffraction analysis. Wet chemical and gamma-scan analyses will be performed on the solutions of crud removed to determine what cations are present in the crud, whether the crud adsorbs or concentrates chloride ions, and what radioactivity it contains and how this is distributed along the fuel column.

Ex-Reactor Testing of Target Elements. Two Zr-clad target elements which had been predefected and rupture-tested in the TF-7 Loop at 530 F were examined. Photomicrographs showed the coolant had not contacted the lithium-aluminum alloy core. The corrosion of the aluminum outer layer was very severe. The most severe corrosion occurred between the aluminum spot-welded end cap and the cladding. A concentration cell was formed at the crevice. However, there was a sufficient release of aluminum to the coolant to determine that a rupture had occurred before the lithium-aluminum core would be affected.

N-Reactor Heat Exchanger Studies. Efforts were continued to determine the source of possible contaminants which caused the corrosion problem in the N-Reactor primary heat exchangers. Two corrosion problems exist in the steam generators. One is a stress-corrosion cracking problem which has led to the failure of approximately 50% of the tubes at the tube mouth area in Unit 3-B and 1.5% of the tubes at the tube mouth area in Unit 4-A. The other is an intergranular pitting attack which was detected throughout most of the remaining units except 1-A, 5-A, and 5-B.

The stress-corrosion cracking problem was found after six steam generators had been hydrotested. Inspection of the remaining four untested units has not revealed any contaminant which could have readily caused the cracking. The units were very clean except for small amounts of vapor phase inhibitor (VPI) which had been added for corrosion protection and a very light black smut which presumably was a residue from the chemical cleaning of the units following fabrication. Numerous samples have been taken for analyses. The amounts of materials present are very small, requiring tedious and time-consuming microanalysis. Techniques being used are x-ray fluorescence, x-ray diffraction, gas chromatography, infrared absorption, and wet chemical analyses. To date, none of the analyses has given a definite indication of the exact cause of either type of corrosion. The water used for hydrotesting has been found to be very low in impurities and could not have caused the corrosion.

Corrosion tests are presently being conducted under accelerated conditions to determine whether the VPI could have possibly caused the stress-corrosion cracking. Other tests are being planned which will more accurately duplicate the conditions known to have existed during the history of the steam generator fabrication and installation.

Corrosion Under N-Reactor Fuel Supports. In a previous test severe corrosion of Zircaloy-2 had been encountered in heated crevices under supports exposed to pH 10 LiOH caused by concentration of LiOH in the crevice by boiling. A similar test was started July 10 to determine whether such attack can be prevented by using NH_4OH in place of LiOH. The test sample consists of an electrically heated rod clad in Zircaloy-2 with three N-Reactor type supports spot welded to it. It is being exposed in TF-3 at 315 C bulk water temperature, 2000 psia, pH 10 NH_4OH . The average surface heat flux is about 200,000 Btu/hr-ft². These conditions will produce nucleate boiling on the free surface as well as in the crevices under the supports. The loop was conditioned with

NH_4OH for two weeks prior to the start of the test. Control of pH by passing feed water through NH_4 - form mixed-bed ion exchange resin has been satisfactory.

Chloride Concentration in Gasket. Recent experience with Flexitall gaskets indicates that chloride may tend to concentrate in the asbestos. A stainless-steel gasket sealing a test section in TF-4 during eight decontamination cycles was cracked and the asbestos was found to contain 8.2 ppm chloride. A similar gasket exposed to two cycles and opened once had 1.9 ppm. An unexposed gasket had only 0.3 ppm.

3. Gas-Atmosphere Studies

Kinetics of the Reaction $\text{H}_2\text{O} + \text{CO} \rightarrow \text{CO}_2 + \text{H}_2$. A knowledge of the kinetics of the reaction of water with carbon monoxide and the reverse reaction under reactor conditions is essential in predicting the extent of graphite oxidation by water and the effect of gas atmosphere on the zirconium process tubes.

An apparatus has recently been completed for the study of the thermally-induced reactions. The apparatus consists of a quartz reaction chamber through which the purified gas mixture passes. The chamber is constructed so that graphite samples as large as one inch in diameter and four inches long can be easily introduced or removed. The inlet and outlet tubes for injection and removal of the gases are small diameter to avoid errors due to reaction outside of the reaction chamber. The reaction rates are determined by a chromatographic analysis of the products formed.

At the present time only one run has been completed. The specific rate constant, k , for the homogeneous reaction derived from the expression $\text{rate} = k(\text{CO})^{1/2}(\text{H}_2\text{O})$ was $1.1 (\text{l/mole})^{1/2}\text{sec}^{-1}$ at 850°C . The calculated value from data in the literature is $0.5 (\text{l/mole})^{1/2}\text{sec}^{-1}$. The agreement is satisfactory on the basis of the limited data. Other runs are planned to determine an activation energy for the homogeneous reaction before studying the effect of graphite upon the reaction.

The radiation-induced reaction between water and carbon monoxide has also been investigated. Measured quantities of water and carbon monoxide were expanded into evacuated capsules irradiated in the gamma source to a total dose approximately 3×10^8 r. A G value for hydrogen formation at room temperature by mass spectrometric analysis of the products was determined to be 4.1 ± 0.7 .

The results of a single experiment indicate that the energy absorbed by helium gas in the capsule is also efficient in initiating the reaction of water with carbon monoxide. Attempts to obtain quantitative data in the presence of helium have been unsuccessful due to leakage of helium through the capsule walls. In the future argon will be used for the inert gas to avoid this difficulty. At the present time it appears that the helium-sensitized reaction of water with carbon monoxide may be very significant even for doses of 10^8 r or less when large quantities of rare gas are present. However, more quantitative data are necessary for definitive conclusions.

Zirconium Oxide Oxygen Monitor. It has been reported previously that the electrical resistance of ZrO_2 corrosion films on Zircaloy vary as the oxidizing character of the environment changes. It appears that ZrO_2 may achieve an equilibrium with oxygen in the environment by the reaction $ZrO_2 \rightleftharpoons ZrO_{2-x} + x O$. Since ZrO_{2-x} is more electrically conductive than ZrO_2 , it may be possible to develop an oxygen monitor based on this principle.

A probe made from a piece of ZrO_2 insulated sheathed thermocouple wire has been exposed to high vacuum and oxygen at 400, 500, and 600 C (752, 932, 1112 F). It has been found that the electrical conductivity of the ZrO_2 changes with the oxygen pressure as predicted. Temperature variations also produce conductivity changes. Current work is to define the limits of sensitivity for oxygen and to determine what calibrations can be achieved.

4. Process Tube Development

Irradiation Effects on Tube Properties. Two Zircaloy-2 pressure tubes have been discharged from KER Loops 1 and 2 after residence times of three and four years, respectively. Loop 2 tube has received about twice the integrated fast flux exposure seen by Loop 1 tube. Selected sections of the more highly exposed tube are being moved into the Radiometallurgy facility for examination, with similar sections from the second tube to follow about a week later.

Project CAH-922 Irradiated Burst Test. Construction of the burst test facility for irradiated tubing is estimated to be 31% complete versus a scheduled 32%. All items to be procured by GE are on order and scheduled for delivery within 90 days.

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5. Thermal Hydraulic Studies

N-Reactor Studies. Installation was started of the full scale electrically heated model of a half column of N-Reactor fuel elements in the high pressure heat transfer apparatus. An outlet nozzle and associated fittings are being included so that the test setup will be an accurate model of that portion of an N-Reactor tube and fuel elements from the midpoint in the process tube to the rear header.

The fuel element portion of the test section consists of two co-axial tubes, $17\frac{1}{2}$ feet long. These tubes are made from various metals to produce a decreasing heat distribution along the length and to maintain a proper proportion of heat generation between the two tubes. Small ceramic pieces, shaped and spaced similarly to the support pieces used in the actual fuel elements, are used to maintain both support and electrical resistance between the two tubes and between the outer tube and process tube. The heated section is instrumented with 16 thermocouples to measure surface and core temperatures and with pressure taps to measure pressure drop across each of the three flow channels. The test section was designed for heat generation rates of 2700 KW under steady state conditions and 3800 KW during overload periods of one-half hour.

Plans have been completed for the first series of tests to be run with this test section. Pressure drop data throughout the test section will be obtained at constant heat generation rates ranging up to 2600 KW while the flow is gradually decreased in steps. Among other things, these data will provide information concerning hydraulic stability and will give indications of conditions leading to boiling burnout.

Burnout Data Comparisons. Twenty-six experimental boiling burnout points obtained recently at Columbia University⁽¹⁾ were compared with 31 points obtained previously in the Hanford heat transfer laboratory using similar test sections and experimental conditions. These comparisons were made to determine if agreement was good enough that the two groups of data could be combined, allowing better definition of burnout curves.

(1) Matzner, B., "Basic Experimental Studies of Boiling Fluid Flow and Heat Transfer at Elevated Temperatures," MPR-13-5 63, May 31, 1963.

The Columbia University data were obtained using a 0.504" ID tube, 76 inches in length. In the Hanford experiments, three 0.445" ID tubes were used, with lengths of 32 inches, 64 inches, and 144 inches. These tubes were cooled by internal water flow and were heated by electrical resistance. The experiments were conducted at comparable pressures: 1000 psia at Columbia and 1015 psia at Hanford.

Common mass velocities used in the two sets of experiments ranged from 2×10^6 lb/hr-sq ft to 7×10^6 lb/hr-sq ft.

Plots of burnout heat flux as functions of coolant enthalpy and mass velocity, showed excellent agreement between the two sets of data. Curves could be drawn through the composite plots with no points deviating more than 10%. Coolant conditions in these plots ranged from 22 F subcooled to steam qualities of about 30%.

6. Shielding Studies

MAC Code Calculations and Development. The Barytes concrete calculation was completed, and analytical data are now available to compare with experimental data on Barytes concrete presently in the DR well. Several additional computer runs, changing the water content to simulate Barytes concrete at high temperatures will be required.

Cross-sections are being calculated for 86 elements and isotopes and arranged for direct input into 31 group MAC calculations. When these cross-sections are completed, a simpler group structure will be set up (probably 12 neutron groups) and cross-sections will be calculated for all 86 elements and isotopes in this group structure.

Shielding Experimentation. Experimentation will be reactivated during August with evaluation of Barytes concrete. The present Barytes blocks in DR-Reactor will be evaluated at 100, 200, and 300 C.

7. Reactor Technology

Fast Spectrum Processing Study. A "reference design" of the reactor and fuel subassemblies has been sketched out. The reactor core is $6\frac{1}{2}$ feet in diameter and length with 1-foot thick axial and radial blankets. The core has approximately 300 hex-shaped subassemblies measuring approximately 2.4" side length by 128" over-all length. Each core subassembly contains 127 fuel rods of 0.282" OD, 0.260"

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ID. The radial blanket subassemblies are similar to the core, except for a thinner-walled hexagonal can. There are 19 large diameter rods in each radial blanket subassembly. The upper and lower axial blankets (at the ends of the core subassemblies) contain 19 large diameter rods like the rod blanket.

The reactor vessel is 10 feet in diameter by 41 feet over-all length by one inch wall thickness. There are four (each) 30-inch inlet and outlet nozzles.

An evaluation of the segmented core concept in large sodium cooled, plutonium fueled fast reactors is also being carried out to arrive at a solution to the coolant void problem associated with the 1000 MW fast thorium converter.

8. Graphite Studies

N-Reactor Graphite Irradiations. Irradiations of N-Reactor graphite continue to proceed satisfactorily in the GETR. The H-6-2 capsule successfully completed the fifth cycle of irradiation and is scheduled for discharge and disassembly the week of July 15. The second of the third-generation capsules, H-5-3, will be installed in the reactor during the same week. The H-4-3 capsule has successfully completed the third cycle of irradiation.

B. WEAPONS - 03 PROGRAM

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

C. REACTOR DEVELOPMENT - 04 PROGRAM

1. Plutonium Recycle Program

Fuels Development

PRTR Fuel Element Inspection. The surface condition of a nonautoclaved Zircaloy-clad fuel element appeared excellent after irradiation to 11 MWD in the PRTR. A uniform black zirconium oxide layer was formed on the top three-fourths of the element after 12-day exposure to PRTR coolant. A thinner oxide layer (evidenced by interference colors) was formed on the bottom end of the element because of lower inlet water temperatures. This test shows that properly prepared cladding can be maintained through charging into the reactor and may open the way to increased potential fuel life through reduced initial hydrogen content, reduced fuel element fabrication costs, and simplified remote fabrication techniques.

Irradiation of this element is continuing and two additional nonautoclaved elements were charged into the PRTR.

A swage-compacted $\text{UO}_2\text{-PuO}_2$ element (5114) was examined in the PRTR storage basin and in the FEEF. This element had previously been retired from PRTR service after an exposure of 320 MWD/T on the basis of a questionable mark seen on the surface of one fuel rod during basin examination. Closer examination in the FEEF revealed that the mark was a superficial irregularity that caused unusual light reflections. The element was returned to service and is operating satisfactorily.

PRTR Fuel Element Development. Three $\text{UO}_2 - 1 \text{ w/o PuO}_2$ PRTR swaged fuel elements were completed. These fuel elements were the first to be made from plutonium oxide which had been converted from available metal. The plutonium metal had been prepared by the reduction of plutonium fluoride. Recent analyses reveal that the fluoride and chloride concentrations in oxidized plutonium metal have been relatively high (60-600 ppm F and 100-350 ppm Cl). An analysis of one of the master blends of $\text{UO}_2 - 11 \text{ w/o PuO}_2$ used in preparing the PRTR elements indicated 11 ppm F, 64 ppm Cl, and 320 ppm H_2O . Some of the rods in each of the fuel elements developed pinhole leaks in the cladding, presumably because of the halides. These fuel elements were the first swaged $\text{UO}_2\text{-PuO}_2$ fuel elements to develop cladding leaks. Plutonium dioxide derived from oxalate is being obtained for future fuel element fabrication.

Post-Irradiation Examination of PRTR Fuel Elements. No fuel washout occurred as a result of the failure of a vibrationally compacted $\text{UO}_2\text{-PuO}_2$, Mark-I element (5179) in the PRTR after an exposure of 380 MWD/T.

The reactor was shut down on 7/11/63 after high primary coolant activity indicated a leaker. A raised blister, about 3/16-inch diameter, was observed in the cladding approximately three inches from the bottom end of one fuel rod. No cladding was missing; however, the Zircaloy appears brittle and cracked in the blister region, thus permitting the release of fission gases. The average exposure of the element and the external appearance of the failure are similar to those of the previous Vipac $\text{UO}_2\text{-PuO}_2$ failures, suggesting the same type of failure mechanism. The natures of these failures indicates that the most probable cause is from some impurity (water, fluoride, chloride) in the fuel material. Several other $\text{UO}_2\text{-PuO}_2$ fuel elements now being used in the PRTR are considered to be candidates for mild cladding failure by a mechanism involving these impurities. Evaluation of archive specimens and of fuel currently in process by newly developed and implemented analytical techniques, and development of additional fuel purification processes, are proceeding on a high priority basis.

Fuel Element Rejuvenation. A rejuvenated fuel element (test GEH-4-84) satisfactorily completed a second irradiation cycle in the MTR. The 0.565-inch diameter fuel rod generated a maximum surface heat flux of approximately 405,000 Btu/(hr)(ft²).

After the first cycle in the MTR, the UO_2 fuel element was rejuvenated by opening the cladding and vibrationally compacting enriched UO_2 into a 1/8-inch ID coaxial zirconia tube incorporated into the element during original fabrication. The element was designed to operate with the enriched core in a molten state and should provide important data on the enrichment relocation important to any rejuvenation concept.

Fuel Element Design Studies. Exposure of a UO_2 -Mark I element with special ring type end brackets designed to reduce wear corrosion, was continued in the TF-7 facility at PRTR coolant conditions. The total accumulated exposure is now 131 days, with negligible wear of the bottom end bracket during this time.

Remote Fabrication Studies. Four hundred pounds of "cold" Salt Cycle electrodeposited UO_2 was received from the Chemical Development Operation. Analytical data are being obtained on this material prior to using it for vibrational compaction and hydrogen sintering studies.

Fuel Refurbishing. A versatile underwater drill for fuel repairs was fabricated and tested.

A small spot welding device of the pliers type was installed for experiments in underwater resistance joining of a sheath to a wire. The

purpose of the experiments is to determine whether or not two pieces of wire such as that used to wrap PRTR fuel elements can be joined end to end by resistance welding a sheath to first one wire, then to the other, while the parts are under water.

Irradiation of Uranium-Plutonium Oxide. Microsamples, obtained by drilling, from sharply-defined zones of markedly different radioactivity levels on an irradiated (U,Pu)O₂ capsule (GEH-14-86) were analyzed. The analytical results, corrected to the date of capsule discharge, indicate a relatively constant concentration of Pu, Ce-144-Pr-144, Zr-95-Nb-95, and Sr-90. However, a ten-fold variation in Cs-137 and a 1000-fold variation in Ru-106 was observed. These more volatile fission products are concentrated nearer the fuel rod surface.

Meleager calculations were performed by R. E. Peterson to provide more precise burnup data for 24 irradiated UO₂-PuO₂ capsules. This calculation is needed for evaluation of fission gas release.

Short Duration Irradiations of UO₂ and UO₂-PuO₂. An MTR VH-4 (rabbit) loop fuel capsule and transport unit were shipped to the MTR-ETR for flow tests.

Preparation of PuO₂ Cermets. Development of efficient methods for preparing UO₂ cermet pellets by pneumatic impaction was continued. Initial tests of breakaway die blocks revealed that graphite was ineffective as a parting material - the parts of the die block were cold welded together by the impaction. Other methods of preventing the parts of the die block from cold welding are being investigated. These include (1) depositing a phosphate coating ("phosphatizing") on the surfaces of the dies, (2) coating the die surfaces with Al₂O₃, and (3) coating the die surfaces with ZrO₂.

Corrosion and Water Quality Studies

Moderator Level Servomanometer Sensing Line. The aluminum sensing lines from the calandria to the moderator level servomanometer corroded through in several areas. These lines are normally back flushed with helium through a desiccant bed and should have posed no corrosion problems.

Physical examination of the inside of the tubing revealed extensive pitting and a heavy layer of oxide. The oxide is primarily white but has areas which are pink, orange, violet and black. The black areas were identified as metallic iron particles, many of which were ground into the tube surface indicating that they were introduced during tube fabrication.

Emission spectrography showed the primary constituents of the film to be aluminum, cobalt, magnesium, iron and copper. The aluminum, magnesium, and iron can be accounted for in the alloy and iron particles. Copper evidently was introduced by the use of a copper-containing lubricant on fitting connections. Cobalt chloride is frequently used as the indicating material in desiccants.

It was determined that several times the line and desiccant bed had become flooded with water. Laboratory tests show that the CoCl_2 leaches easily from Drierite. Chloride ion causes severe pitting of aluminum, especially in the presence of other metals such as copper and iron. Preliminary laboratory tests have established the presence of chloride ion in the oxide film and quantitative analysis of the oxide is in progress.

Inspection of Shroud Tubes. Shroud tube 1847 was internally inspected by borescope. Most of the tube was clean with a thin, uniform oxide film. A number of small dark spots, apparently stains, were seen on the tube. No evidence of nonuniform or excessive corrosion was seen. The top and bottom shield sections above and below the shroud tube are covered by a layer of red rust but show no areas of nonuniform attack. The entire traverse of the tube was photographed with a 16 mm movie camera.

Zirconium Concentration in PRTR Primary Coolant. Routine monitoring of the zirconium concentration in the primary coolant of PRTR was continued to detect possible fretting. Grab samples were analyzed by emission spectrography.

During the past month the zirconium concentration was generally within the range considered normal for this system; that is, 0-1 ppb. However, on two separate occasions the concentration increased rapidly to values greater than 10 ppb and 3 ppb, respectively. Both abrupt increases were observed during normal operation at rated power. The first period of high concentration lasted about three days; the second, one day. It is possible that fretting corrosion may have been occurring in the system during the first period. It is likely that the second indication was a spurious signal, or at least did not represent much corrosion since the concentration was not excessively high and returned to normal very rapidly.

A continuous, colorimetric procedure for measurement of ionic zirconium concentrations in water is being investigated using the Technicon Auto-Analyzer. Results obtained to date indicate that concentrations as low as 1-2 ppb can be detected. Studies are being conducted to determine the maximum sensitivity and degree of reproducibility of the procedure.

Reactor Components Development

PRTR Pressure Tubes. The final elevated temperature burst test of a group of six tubes was performed. The data analyzed to date indicate that an increase of 37.8 C (100 F) in test temperature decreases the strength by about 10% within the test temperature range of room temperature to 316 C (600 F).

The pressure-time records of all the irradiated PRTR pressure tube burst tests have been examined. Each of these records show a linear and a nonlinear pressure-time trace for the portion of the test preceding the peak pressure. This may be evidence that these specimens of irradiated Zircaloy-2 tubing displayed a yield strength under biaxial stressing. There are events other than yielding that could occur during a burst test which would account for this departure from linearity. Some examples are: a leak in the pressure system, slipping of the test specimen end fixtures or an increase in the compressibility of water. However, if the existence of a biaxial yield strength can be proven, this would be evidence that these irradiated Zircaloy-2 specimens were not brittle. This point is important particularly with respect to irradiated cold worked material, some of which has displayed such a small amount of plastic deformation during burst testing that it was barely measurable by existing techniques.

The thermal and fast neutron distribution in a PRTR experimental hole has been measured recently. The ratios of fast to thermal neutrons from these data were used to deduce a fast neutron distribution curve for the PRTR using a thermal neutron distribution curve. The result was that the fast neutron boundary in the PRTR was indicated to be at elevations of minus 4.775 meters (15 feet, 8 inches) and minus 2.59 meters (8 feet, 6 inches). In other words, 1.83 meters (6 feet) of the inlet end of the pressure tube and 1.295 meters (4 feet, 3 inches) of the outlet end receive a relatively insignificant neutron dose. This leaves 2.18 meters (7 feet, 2 inches) of the tube that sees fast neutrons. However, there is a transition in the microstructure of about 0.345 meter (1 foot) long between the cold worked and annealed metal that is located somewhere in this fast flux zone. By precise location of the transition zone, better utilization of irradiated PRTR pressure tubes for testing would be possible, i.e., three fast neutron irradiated burst test specimens, each 0.508 meter (20 inches) long and of uniform microstructure per tube, could be realized. A program proposal for the development of an ultrasonic device for location of the transition zone has been obtained from Testing Methods of NRD.

Room temperature burst tests of irradiated PRTR pressure tube specimens from the 9th and 10th tubes discharged were performed. The specimen locations were such that the upper and lower fuel element support marks were included to learn if they had weakened these tubes. Results of the tests show that the strengths of these tubes were not weakened by these flaws.

Pressure Tube Monitoring. Work continued on the assembly of Mark III PRTR pressure tube monitoring equipment. Installation and operational checkout of the new objective lens section and dial gage measuring system were completed. Electrical work of the probe system progressed to the point where it was possible to perform and complete functional checkouts. Instrument change-over to the Mark III from Mark II equipment is under way, but completion and checkout is awaiting delivery of the overdue six-pen strip chart recorder.

Second Generation Mechanical Shim Rod for PRTR. Fabrication of all parts, other than those which are to be procured, is estimated to be 95% complete. The Syncro position readout equipment for the second rod assembly has been received. Procurement of the remaining mechanical hardware has been delayed due to higher priority work.

Shim Rod Environmental Test Facility. Operation of the facility was terminated and the first generation rod was removed from the facility. The test section was disassembled in preparation for the installation of the glass test section.

Fabrication of a prototype of the mechanical braking device was completed. Testing of the device was completed and the results showed that the device performs its function satisfactorily.

Fretting Corrosion Investigation. Installation of the PRTR prototypical process tube in the EDEL-I test pit is completed. While the instrumented fuel element is being completed, all other instruments consisting of five vibration pickups and a differential pressure flow transducer were installed. An initial run, with no fuel element installed, resulted in high vibration amplitude near the center section of the process tube, thus indicating process tube vibration as a potential source of fretting corrosion. Collection of pre-test data is in progress before initial runs using a noninstrumented fuel element.

PRTR Pressure Tube Seal Testing Station. Design is complete and fabrication is 60% complete.

Shroud Tube Mockup. The shroud tube mockup pit is 95% complete. The crates to simulate the calandria are being fabricated by Minor Construction and will be installed early next month.

Connectors. The experimental, uncoated, Inconel seal-ring was received from Gray Tool Company, assembled in the rupture loop Grayloc connector, and installed in the EDEL-II test loop. Spotty leak patterns, ranging from 1.0 cc to 15.4 cc, were measured during the first nine cycles.

Design Analysis

Fuel Re-Use. An extension of the economics analyses of the fast oxide reactor - D₂O pressure tube reactor, fuel re-use system (reported in HW-74549) has been initiated.

The first step is an examination of the fuel costs of the D₂O reactor as a function of the fuel service fee charged by the fast reactor (blanket) operator. The price of Pu is assumed to be \$10/gm fissile, the economic interest rate is 12½%, three levels of use charge - 4-3/4%, 7%, 12½% - are used, and the fuel service fee is varied from \$10 to \$40/lb fuel.

A comparison of minimum fuel costs for this fuel re-use cycle, a U-235 cycle, and normal Pu recycle showed that the fuel re-use cycle is cheaper over the full range studied. Apparently, then, the thermal reactor could pay the total fabrication costs for the blanket of the fast reactor (in addition, of course, to burnup and use charges for the bred Pu) and still generate power at a lower cost than a standard fuel cycle.

2. Plutonium Ceramic Fuels Research

Plutonium Ceramic Irradiations. A plutonium carbide wafer has been successfully irradiated for 44 days in the ETR to an estimated average burnup of 4.2 a/o (37,000 MWD/T or 1×10^{20} fissions/cc). A second PuC wafer has an average exposure of 2 a/o burnup. Additional capsules containing plutonium oxides and pneumatically impacted plutonium nitride wafers await irradiation.

Mixed Carbides. The melting points of a series of PuC-TaC alloys were determined. Melting points decrease rapidly as the PuC content increases, but at 50 w/o PuC the melting point is still above 2400 C.

Plutonium Dioxide Spheres. High density spheres of plutonia with known stoichiometry are useful as basic specimens for oxidation, gas solubility and diffusion studies. Spheres were produced by gas-jet grinding of irregular pieces obtained from crushed high density

$\text{PuO}_{1.7}$ and $\text{PuO}_{2.0}$ arc-melted buttons⁽¹⁾. These spheres were uniform in diameter (0.3 ± 0.005 cm) and 99% TD. Several uniform spheres are produced in one loading of the grinding apparatus⁽²⁾.

During initial grinding operations it was observed that the plutonia fines deposited on the inner surface of the alumina tube that functions as the primary grit, and these abrasive plutonia fines helped to produce spheres faster.

Plutonium Vaporization Studies. Preliminary data show that the vaporization rate of PuO is considerably greater than that of beta- Pu_2O_3 . This conclusion is drawn from the following data:

<u>Material</u>	<u>Temp., °C</u>	<u>Pressure, 10⁻³ torr</u>	<u>Vaporization Rate, mg/cm²/hr</u>
$\beta\text{-Pu}_2\text{O}_3$	1750	0.3	30
PuO_2	1550	40	65

A pneumatically impacted PuN-5C v/o W cermet pellet showed a total weight loss of only 0.03% after heating in vacuum at 1000 C for four hours and at 1200 C for four hours. Density measurements show an increase of 3.5% in specimen volume. Metallographic examinations are in progress to learn the cause of this volumetric expansion.

Plutonium Capabilities Study. A joint AEC-GE study team issued the results of its survey of plutonium fabrication capabilities of five AEC sites.

3. Ceramic (Uranium) Fuel Research

Thermal Conductivity of UO_2 . Thermal conductivity measurements were completed on the final specimen (#57) of the four-year irradiation series of polycrystalline UO_2 in a cooperative program with Battelle. The specimen, having a burnup of 1.3×10^{19} fissions/cc (40% more than the previous specimen, #51) suffered a further small decrease of 7% in low temperature conductivity. Recovery upon annealing followed the pattern shown by all earlier specimens, to the maximum measurement temperature of approximately 300 C.

(1) Evans, E.A., Ceramics Research & Development Operations Quarterly Progress Report, April-June 1963, HW-76302.

(2) Stablein, P.F., "Apparatus for Forming Small Ceramic Spheres," Journal of the American Ceramic Society, October 1962, pp. 505-506.

Additional thermal conductivity measurements on specimen #51 (1.9×10^{19} fissions/cc) at room temperature showed greater than 90% recovery of initial nonirradiated conductivity after annealing at 900 C. Data from this specimen indicate that 100% recovery would result from annealing at about 1200 C.

Earlier irradiation of the large UO_2 single crystal to approximately 10^{14} fissions/cc apparently resulted in significant embrittlement and/or cracking. Despite careful handling, fracturing of the specimen occurred during preparation for measurements. Further measurements on this crystal were discontinued.

The long-term cooperative study with BMI on UO_2 thermal conductivity has been concluded. All specimens have been returned to Hanford for further examination.

Optical Properties of UO_2 . Photographs of very thin sections of single crystal UO_2 taken by transmitted light reveal opaque centers in connecting arrays throughout a brilliant red background. These opaque centers conceivably result from impurity clusters or lattice strains along low angle grain boundaries in single crystal UO_2 . Other opaque centers are related to uranium inclusions in UO_2 .

Analysis of Radial Temperature Profiles. A 7090 computer code (CHEAT) was developed cooperatively by THO and FTAO personnel to enable rapid analysis and plotting of radial temperature profiles of ceramic nuclear fuel rods. A wide variety of fuels, including those having severe self-shielding, can be treated. Significant savings in manpower requirements for preparation of irradiation test proposals were achieved.

Structure Development in UO_2 During Irradiation. Nine capsules containing 1 v/o tungsten spheres randomly dispersed in sintered UO_2 pellets were shipped to the NRTS for irradiation. The object of these irradiation tests is to reveal the structural features associated with central melting in UO_2 fuel elements that have been irradiated for various periods. Previous tests have proved that the limit of central melting can be determined by the position of the tungsten marker spheres after irradiation.

Examination of Electrical Resistivity Element. Autoradiographs of cross sections of the electrical resistivity test element discussed in last month's report show a high concentration of fission products at the inner boundary of the columnar grain region. The dense columnar grains of UO_2 appear as an area of low fission product concentration.

Irradiation of Cermet Fuels. Tungsten- UO_2 cermet irradiations in HAPO Snout facility were continued with the successful irradiation (two hours) of a test capsule under conditions selected to cause a cladding surface temperature greater than 2000 C. This is the third in a series of ultrahigh temperature tests to determine the short term irradiation stability of cermet fuels.

Magnetic Force Resistance Welding Studies. A new magnetic force welding chamber was received. The new chamber provides approximately 50% more volume and a much larger viewing window than the present chamber. In addition, the inside surface has a smooth finish to facilitate removing any internal chamber contamination, such as beryllium, that might accumulate on the walls. The chamber fits existing equipment without requiring modifications.

A nickel-chromium alloy, Incoloy-800, was successfully joined using the magnetic force resistance welding process. Metallographic examination reveals that recrystallization grains in the weld zone are smaller than the grains of the parent material. The weld occurred in one cycle (1/60 sec), using a current of 20,000 amps.

Be- UO_2 Compatibility. A can containing a layer of nine small cans filled with Be- UO_2 mixtures to be used for basic studies on compatibility was pneumatically impacted. The technique probably is applicable to impaction of other materials in small cans. Stainless steel cans, 3/8" OD by 1/2" length and 1/16" wall and end thickness, were loaded with the powder to be impacted, placed upright on a heavy steel disc inside a regular impaction can, surrounded by Al_2O_3 powder, and covered by another heavy steel disc. Several cans may be placed in each layer, permitting a large number of samples to be impacted under the same conditions.

4. Basic Swelling Program

Irradiation Program. One general swelling capsule is continuing to be irradiated at a control temperature of 625 C (1157 F). A second capsule reached its goal exposure and was discharged. This second capsule initially operated at 525 C (977 F), but the temperature was lowered to 180 C (356 F) during the latter stages of irradiation. The specimens in this low temperature capsule will provide information to support or refute theoretical considerations advanced in the recent Metallurgy Research Quarterly Report, HW-77954.

Samples have been prepared and assembly is about 20% completed for two additional general swelling capsules.

Three NaK-filled laboratory capsules containing specimens similar to those included in recent general swelling capsules have completed a one-month anneal at 525 C, 575 C, and 625 C (977 F, 1067 F and 1157 F). In addition to the capsules representative specimens were vacuum annealed at the same time. The specimens will be recovered and processed in the same fashion as irradiated samples in order to obtain greater insight into the mechanisms of irradiation damage in uranium.

Post-Irradiation Examination. Optical and electron metallography have been completed on the two as-extruded high purity uranium specimens from a capsule irradiated to $\sim 2 \times 10^{19}$ fissions/cc (0.03 a/o B.U.) at 525 C. Electron microscopy revealed crystallographically aligned micro tears. The relative absence of small spherical pores indicates that fission gases are not contributing significantly to the volume changes observed in these specimens in this temperature range. The analogous beta-quenched sample is currently being reprocessed.

Specimens were recovered from another capsule irradiated to 6×10^{19} fission/cc (0.10 a/o B.U.) at 625 C. These samples were initially enriched to 1.44 a/o U-235. The density measurements indicate the following volume changes: as-extruded specimen 19A-2, 640 C (1184 F), +8%; as-extruded specimen 19B-2, 625 C (1157 F), +13%; beta-quenched specimen 19C-2, 575 C (1067 F), +28%. Metallography is now in progress.

Post-Irradiation Annealing. The six peripheral rods of a seven-rod cluster of Zircaloy-2 clad uranium have been cut into $\frac{1}{2}$ " thick wafers for pulse annealing studies. The cluster was irradiated at ambient temperature to an estimated burnup of 2×10^{20} fissions/cc (0.37 a/o). The rods are about $\frac{1}{2}$ " in diameter. Wafers will be annealed at various temperatures in the range 550-800 C (1022-1472 F) for times from 10 minutes to 3000 minutes. Each combination of time and temperature will apply to a declad wafer and a clad wafer. A special vacuum furnace constructed for these annealing studies can heat a specimen to the desired temperature in about 10 minutes. Cooling occurs at about the same rate and is hastened by flowing inert gas around the sample. The changes that occur with annealing will be evaluated by metallography, hardness, density, and x-ray diffraction. The as-irradiated material has been processed through metallography. The specimen has the typical cold-worked structure of uranium irradiated at a low temperature. Several cracks are present but they may have been introduced during cutting or processing for metallography. Several pairs of specimens have been annealed and are currently being processed.

Thorium-Uranium. Three tensile specimens (Th-1, 4 and 5.4 w/o U) were vacuum annealed at 900 C (1652 F) for two hours and will be processed in a fashion analogous to the irradiated Th-U tensile specimens to determine the changes that occur due to irradiation. Optical metallography on the four irradiated specimens discussed last month show no evidence of damage due to irradiation.

Restrained Irradiations. The influence of restraint on the swelling of uranium is being investigated with NaK capsule irradiations of Zircaloy-2 clad rods of unalloyed uranium and uranium - 2 w/o zirconium. Nine capsules irradiated to approximately 0.66×10^{20} fissions/cm³ (1100 MWD/ton) have been opened and the samples photographed and measured. Several of the samples contained an intentional striation in the cladding at the cladding-uranium interface. The volume increases calculated from the data obtained are in good agreement with previous experimental results on restrained uranium swelling. The greater diameter increase of the U - 2 w/o zirconium fuel over the unalloyed uranium is also consistent with past experience. There doesn't appear to be a consistent correlation between the original depth of striation in the cladding and the final average diameter increases. This is not unexpected as the surface of the cladding on many of the fuel samples was bumped and only nine diameter measurements were made. The average diameter increase does not therefore represent an accurate indication of the gross changes of the samples. Measurements of density and burnup are first being made on the samples with no intentional cladding striations.

Second Phase Distribution in Dilute Alloys of Uranium. Possible relationships between the swelling resistance of uranium metal and impurity concentration were first considered in FY-1962; active experimental work, however, was subsequently interrupted. This investigation on a series of high purity ingot uranium specimens containing dilute additions of iron, silicon, and aluminum has recently been renewed and has the following objectives: (1) identification of the different types of intermetallic particles which appear in the matrix, (2) determination of the size and distribution of the intermetallic particles as a function of composition and heat treatment, and (3) correlation of the particle size and distribution with the swelling resistance of the specimens during and after irradiation treatments.

Previous work had reached the following state of investigation: a total of 30 samples had been prepared from three different high purity ingot uranium cores. The nominal compositions of these three cores can be classified as (1) unalloyed, (2) alloyed with 86 ppm Fe and 69 ppm Si, and (3) alloyed with 150 ppm Fe and 100 ppm Si. The

Hanford spectrochemical analysis of the three cores, together with the applied series of heat treatments, was reported in the Fuels Development Operation Quarterly Report of January 1962, HW-72347, p. 3.32.

The results of the optical microscopy confirmed the existence of a dispersion of impurity particles, but the magnification (750X) was too low to warrant a particle size and distribution analysis.

Each specimen was subsequently prepared for electron microscopy by an additional cathodic vacuum etch, followed by replication of the surface. Optical and electron micrographs were obtained, but apparent redeposition of the uranium metal during the cathodic vacuum etching produced such a high concentration of artifacts that the true specimen surface was obscured.

An electrolytic etching technique has been developed which provides sufficient surface relief so that replicas can be prepared for electron microscopy. The sample is given a final polish on a rotating wheel with a Linde B abrasive suspended in 2% chromic acid. After electropolishing at 18 v (closed circuit) for 10 seconds, the specimen is electroetched at 5 v (closed circuit) for 50 to 100 seconds. A carbon replica of the polished and etched surface is then prepared from a negative Faxfilm replica by the usual technique.

Samples are currently being prepared for metallography and will be photographed at 15,000X with the electron microscope. A thorough particle size and distribution analysis will be performed.

5. Irradiation Damage to Reactor Metals

Alloy Selection. Procurement of materials to be used as test specimens for the Irradiation Effects on Reactor Structural Materials Program is continuing. Several alloys including 304, A-286, Inconel 600 and Inconel X750, were received this month and quantities of AM350, 406 and Zircaloy-2 have been shipped by vendors. In most cases these have been partial shipments including the 1/8" sheet and 5/8" bar materials. The heavier plates and bars are in the final stages of inspection.

Tensile specimens of R-27, R-235, Inconel 600, Inconel 625, Inconel 702, Inconel 718, Hastelloy N, Hastelloy X-280, Incoloy 800, and T.D. Nickel have been fabricated and are being irradiated to determine the effects of irradiation and environment upon their mechanical and physical properties. Six ETR loop quadrants containing these

alloys have been discharged and shipped to Radiometallurgy for testing. These were irradiated in water at 280 C (536 F) and achieved an estimated exposure of 5.88×10^{19} , 1.3×10^{20} and 5×10^{20} nvt. To date, 24 specimens have been tested. The test data are in the process of being analyzed.

Five alloys are now being examined which are candidate alloys for fuel cladding in the BONUS nuclear superheat reactor designed by General Nuclear Engineering Corporation and now under construction in Puerto Rico. Reactor Metals Research has been asked by the USAEC-DRD to provide irradiation data on Inconel 600, Incoloy 800, Inconel 625, 406 SS, and Hastelloy X-280 to the BONUS nuclear group. Each alloy will be irradiated in 80 C (176 F) and 280 C (536 F) water and at 525 C to 700 C (977-1292 F) gas environments. Exposure levels between 5×10^{19} and 1×10^{21} nvt will be reached with the materials in various metallurgical conditions, including the solution treated, as-received, and cold worked conditions.

In-Reactor Measurements of Mechanical Properties. The following in-reactor creep rates were determined during the month:

<u>Stress</u>	<u>Temperature</u>	<u>Creep Rate/Hour</u>	<u>Time</u>
25,000	350 C (662 F)	2.3×10^{-6}	580 hrs
30,000	330 C (626 F)	6.4×10^{-6}	200 "
35,000	310 C (590 F)	4.9×10^{-6}	600 "

Out-of-reactor rates at 25,000 psi and 350 C (662 F) and at 35,000 psi and 310 C (590 F) are about 2.2×10^{-6} /hr and 2.3×10^{-6} /hr, respectively. Out-of-reactor creep rates for the 330 C (626 F) test are not yet available. Out-of-reactor rates are not actually steady state rates as the creep rate is always either decreasing or increasing with time. The out-of-reactor rates reported here are expected to decrease with time.

The above in-reactor data correlate well with previously obtained in-reactor data when plotted on an Arrhenias diagram. Proper interpolation and extrapolation of the diagram allows in-reactor creep rates to be predicted throughout the stress range of 20,000 to 35,000 psi and temperature range of 275 C to 450 C (527-842 F).

Two additional capsules containing 20% cold worked Zircaloy-2 have been charged. These capsules will be operated at 25,000 psi and 370 C (698 F).

The heater lifetime capsule has accumulated 1200 hours at the condition described in the May monthly report. No heater failure has occurred in this test to date.

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.

With the conclusion of ETR cycles 53, 54, and 55, thirteen quadrants of notched and unnotched sheet tensile specimens are discharged from the ETR G-7 hot water loop. This brings the total of specimens irradiated and discharged from this facility since startup in cycle 39 to 1500. Control specimens for cycles 52 and 53 of the out-of-reactor loop were also discharged.

Flux data from the G-7 loop for ETR cycles 45 through 51 have been obtained and analyzed. The dosimeters employed include iron, nickel, titanium, and aluminum - 0.1 w/o cobalt wires. Flux values based on iron and nickel are in very close agreement, showing less than 2% variation. Flux values based on titanium are generally higher than those for iron and nickel, and the deviation increases with fast flux, reaching a maximum of about 7.5% at the peak flux of 1.1×10^{14} nv.

Tensile specimens of AISI 304, 348, 410, and AM-350 stainless steels irradiated in the ETR G-7 hot water loop to exposures ranging from about 3×10^{19} to 3×10^{20} nvt have been tested. Additional specimens exposed at duplicate times and temperatures in an out-of-reactor loop have also been tested. The purpose of these latter tests was to determine thermal effects alone on mechanical properties.

Changes in yield strength and total elongation with neutron exposure were determined based on material properties in the as-prepared condition. These changes in properties for an irradiation temperature of 540 F (282 C) were compared with previously reported data for irradiations at less than 200 F (93 C).

At neutron exposures greater than 2×10^{20} nvt, the increase in strength and decrease in ductility due to combined environmental and neutron effects were greater for the higher temperature of irradiation. This effect was observed for all of the stainless steels investigated, but the changes were most pronounced for the austenitic steels in the cold worked condition and for AM-350 stainless steel in the L-annealed condition. Time effects were also observed for both

410 and AM-350 stainless steels. This is, specimens irradiated over a longer time to achieve the same integrated exposure exhibited a greater change in properties.

Examinations of data for out-of-reactor exposures revealed that significant property changes are brought about by thermal effects alone. For example, the yield strength of cold worked 304 and 347 stainless steel increased by 11,000 and 16,500 psi, respectively, with exposure times of 300 hours and greater. The yield strength of AM-350 stainless steel in the L-annealed condition increased from about 62,000 to 115,000 psi with 700 hours of hot water exposure. In most cases losses in ductility accompanied the increases in yield strength. When these thermal effects on properties are accounted for in the irradiated data, the changes in properties at 540 F (282 C) are in better agreement with those measured at less than 200 F (93 C).

The use of refractory alloys in various high temperature reactor applications necessitates determining the effects of irradiation and environment on their mechanical and physical properties. Round, buttonhead tensile specimens of Cb, Ta, Mo and W-25Re alloys have been fabricated for irradiation in the ETR ambient water x-basket facilities.

Pre-irradiation characterization of these alloys, including metallographic examination, hardness and tensile strength measurements, and recrystallization temperature determinations, was started during this past month.

Tensile specimens of TZM, Cb-1 Zr, and Cb-752 are presently being irradiated in the ETR G-6 position and will be discharged at the end of cycle 57. Corrosion tests of these alloys in 300 C (572 F) water were completed, and the results indicate that corrosion resistance and scale adherence are sufficient at least for one cycle of irradiation in the ETR hot water loop facility. The Ta-10 W corroded under these conditions at about the same rate as Zircaloy-2. The rates for Cb-752 and Cb-1 Zr were greater by about three times and ten times, respectively.

With the completion of slow bend tests on 22 notched beams of annealed Zircaloy-2, a major phase of the investigation of brittle fracture in Zircaloy-2 has been completed. In this phase the effects of temperature, cold work, loading rate, and anisotropy on the fracture behavior of unirradiated Zircaloy-2 were examined. During the month six annealed notched beams were encapsulated for irradiation in the ETR in order to investigate the added effects of

neutron damage on fracture behavior. The following conclusions have resulted from studies to date:

1. Zircaloy-2 containing a crack or stress riser undergoes a distinct transition from ductile to brittle fracture. This transition is characterized by an abrupt change with decreasing temperature from high ductility and high notch-fracture strength to undetectable ductility and low notch-fracture strength.
2. Above the transition temperature the notch strength is considerably higher than the yield strength; whereas, at sub-transition temperatures the notch strength is sufficiently low that flaw size must be considered in assessing the possibility of catastrophic failure.
3. The transition temperature increases with loading rate and cold work, other variables kept constant. The shift in transition temperature with loading rate can be as high as 500 C (932 F), and the shift caused by 23% cold work is about 50 C (122 F).
4. Fracture toughness in the temperature region of brittle fracture also increases with increase in cold work and loading rate. The increase amounts to 75% with 23% cold work and 70% with a 40-fold increase in loading rate.
5. Brittle crack propagation occurs with equal ease in both the rolling and transverse directions but is substantially more difficult at 45 degrees to the rolling direction (all notches normal to the rolling plane). In the latter case actual propagation occurs either 20 or 70 degrees to the rolling direction and requires from 100 to 200% greater fracture stress. The decrease in fracture strength in going from ductile to brittle fracture is much less abrupt on the 45 degree plane than in the rolling and transverse directions.
6. At sub-transition temperatures crack propagation is accompanied with cleavage planes by prior plastic deformation. This distortion forces the crack to change direction and jump from plane to plane.
7. Mechanical twinning accompanies crack propagation at low temperatures; however, the exact relationship between twinning and fracture is as yet undetermined.

The critical elastic energy release rate for plane strain has been calculated in the brittle fracture range as a function of the variables investigated. This parameter can be applied to the design of structural components to minimize the possibility of catastrophic failure under established operating conditions.

Environmental Effects. To aid the analysis and interpretation of high corrosion rate data resulting from exposure of Zircaloy-2 in the G-7 loop of the ETR, arrangements have been made to obtain daily analysis of loop water pH, conductivity, and oxygen content. A neutron activation analysis of the coolant and a chloride determination will be made once each week.

In addition, the G-7 loop has been equipped with connections in the Sub-Pile Room for attachment of an out-of-flux exposure chamber. This chamber will permit exposure of unirradiated specimens to neutron irradiated loop coolant within one second of coolant discharge from the active zone. Design work on the exposure chamber is in progress. A basket attachment for the top nozzle of the G-7 loop process tube is also being developed to permit exposure of structural metal specimens to coolant out-of-flux but following an appreciable time delay permitting recombination of the radiolysis products of the water and decomposition of hydrogen peroxide that is formed in the core.

A series of out-of-reactor experiments designed to probe the effects of neutron damage within metal specimens on rate of reaction with coolant environments was initiated during the report period. Specimens of Zircaloy-2 irradiated under identified conditions in the ETR in Quadrants 48 and 82 and representing different levels of cold work are being exposed with control specimens to 400 C (752 F), 1500 psig refreshed steam in an autoclave. A number of these specimens have their reactor-gained oxide coatings intact, while others have been stripped of oxide and etched. The unirradiated control specimens were cut from virgin base plate material and from out-of-reactor oxidation test coupons having thicknesses of oxide comparable to the irradiated samples. Weight gain results will be compared at intervals up to 42 days and hydrogen sorption determinations will be made on all specimens at the end of the exposure period.

Gas Loop Insulation Studies. Helium at about 350 psi will pass between some of the model loop components while at a temperature of 1700-2050 F (927-1121 C). Piping for this helium will be a double contained arrangement consisting of a hot superalloy inner pipe, a pressure bearing stainless steel outer pipe and the

separating insulation. The type of separating insulation is being selected by evaluation of available products.

Commercial insulators were compared using the following criteria: insulating efficiency, strength properties, availability, and dimensional and chemical stability. Fiberfrax, composed of an alumina-silica fiber, was selected as the prime candidate while Micro-Quartz (SiO_2), Refrasil (SiO_2), and graphite felt (C) were selected as possible alternates.

A castable or tappable form of Fiberfrax has been obtained for trials in making insulating blocks. Experiments indicate that the surface of the tamped shapes is made harder and stronger by a covering of Refrasil or other insulating cloth and by basting with an inorganic binder. Although considerable strength can be developed in the tamping direction by thorough intermixing of the ceramic fibers, the use of stainless steel hardware cloth or some other structural member may be necessary to insure adequate strength of the insulating blocks.

The first full size Fiberfrax insulating block to fit over the two-inch inner pipe and inside the 10-inch outer pipe has been made with the insulating cloth cover, basting treatments and the stainless steel mesh. Evaluation of the block is under way.

6. Gas-Cooled Reactor Studies

EGCR Graphite Irradiation. The sixth capsule, H-3-6, in the series of irradiations of EGCR graphite has successfully completed three cycles of irradiation in the GETR. All thermocouples except number 5 continue to operate properly.

The length-change results from the samples irradiated in the first five H-3 capsules have been reported in HW-SA-2926, "Effect of Massive Neutron Exposure on the Distortion of Reactor Graphite," which is being submitted for publication in Carbon. Over the temperature range studied, 425-800 C, a definite temperature effect on distortion continues to be evident. The samples irradiated at the higher temperatures contract at a lower rate. At all irradiation temperatures the samples cut parallel to the extrusion axis are contracting at an increasing rate with increasing neutron exposure.

The EGCR transverse sample irradiated at 800 C actually expanded during the last exposure period. The contraction was a maximum of about 0.9% at an exposure of about 1.2×10^{22} nvt, $E > 0.18$ Mev.

The trend toward reduction in contraction rate is also indicated in the CSF transverse samples irradiated at 800 C. However, below 700 C and at lower exposures the contraction rates of the transverse samples are still increasing.

In-Reactor Creep in Graphite. GEH-13-94 was installed for cycle 55 in the ETR and is operating very satisfactorily. At present the test is in cycle 56 and performing as expected with a temperature gradient across the sample of 10 C. This test is presently scheduled for at least three reactor cycles attaining an estimated exposure of 10^{21} nvt, $E > 0.18$ Mev. Analysis of the data of four tests completed to date indicates that a major part of the expected contraction in graphite is restrained by an 800 psi tensile load after exposure to an integrated flux of 5×10^{20} nvt, $E > 0.18$ Mev at a temperature of 475 C.

Thermal Conductivity of Porous Carbon. Low density porous carbon is being considered for thermal insulation in the High Temperature Lattice Test Reactor. Normal commercial grades of porous carbon, processed at 1000 C, are not suitable, since higher temperatures will be encountered in the reactor which would change the properties of the carbon. A candidate material prepared by heating to 1500 C was obtained and its thermal conductivity determined from 25 to 525 C. The thermal conductivity was $0.0032 \text{ cal cm}^{-1} \text{ sec}^{-1} \text{ }^{\circ}\text{C}^{-1}$ at 25 C and 0.0051 at 525 C, or about twice the values for the standard grade processed at temperatures less than 1000 C. The density of the sample was 0.62 g/cm^3 .

Graphite Compression Test. A paper, "Effect of Low Compressive Stresses on Radiation-Induced Dimensional Changes in Graphite," was presented at the Sixth Biennial Conference on Carbon held at Pittsburgh, Pennsylvania. Discussions were held with Mr. H.H.W. Losty (General Electric Company, Ltd., England) who is also examining the effect of stress on graphite dimensional changes. Mr. Losty reported that the initial "creep constants" obtained from "restrained-growth" samples and from irradiated graphite springs were a factor of 10 to 20 higher than the latest results. The results from his latest "restrained-growth" experiments are in excellent agreement with those obtained at Hanford. This is especially encouraging in view of the differences in experimental arrangement, irradiation temperatures, and stress levels employed.

Long-Term Thermal Effects in Graphite. Little is known concerning the effect of prolonged heating and temperature cycling on the physical properties of graphite. Although it is not considered likely that a significant part of the radiation-induced dimensional

change occurring at high temperature is due to thermal effects, no definitive tests have been made at Hanford. Hence, an experiment to measure the changes in length due to the heat treatment was performed.

Six unirradiated graphite samples which had been annealed at 1800 C for one hour in vacuum were examined. These samples were weighed and lengths were measured prior to being sealed as pairs in three quartz tubes. The tubes were evacuated after being flushed with helium. Weight losses due to oxidation were negligible ($< 0.01\%$).

Transverse cut samples of CSF and TSX graphite heated at 800 C for 31 days, and parallel and transverse samples subjected to 40 thermal cycles per day between 415 and 770 C for 31 days showed no dimensional change (less than 0.002%). Samples of NC-7 were held at room temperature about eight hours, heated sharply to 800 C, held for 16 hours and cooled to room temperature. After 30 such cycles a barely measurable expansion was found for the transverse sample. The parallel sample showed no change.

Measurement of Coefficient of Thermal Expansion (CTE). A quartz CTE furnace is being developed to measure the thermal expansion of four graphite specimens simultaneously. Initial measurements up to 800 C have been made using a protective atmosphere of purified argon gas. Results are given in the following table for CSF graphite (Bar #122-60) measured transverse to the extrusion direction and pyrolytic graphite deposited at 2200 C and annealed to 3600 C measured in the c and a directions. The tabulated values include a correction factor of $-0.51 \times 10^{-6}/^{\circ}\text{C}$ to compensate for expansion of the quartz in the furnace. The CTE values are the average between 25 C and the measuring temperature.

Temp., °C	CTE $\times 10^6$		
	CSF (a)	PG a)	PG (c)
200	3.65	-0.91	25.4
400	3.91	-0.61	30.8
600	4.03	-0.26	32.3
800	4.20	-0.07	33.6

The tabulated CTE's agree reasonably well with other experimental data except for the expansion in the c direction of the pyrolytic graphite. This particular value at 400 C and above is higher than reported for nonannealed pyrolytic graphite for reasons that are not known at present.

7. Graphite Radiation Damage Studies

Irradiation of Lampblack-base Graphite. In a continued search for saturation of high temperature radiation-induced contraction, samples made from a lampblack filler and heated to 1400 C are being irradiated at approximately 1000 C in "hot capsules" in the ETR. A total contraction of 3.42% has been accumulated after an estimated total exposure of 2.1×10^{21} nvt, $E > 0.18$ Mev. A length change of 0.42% occurred during the last exposure of 5.5×10^{20} nvt compared to a change of 0.80% during the immediately preceding exposure of 6.5×10^{20} nvt. The contraction rate definitely is decreasing which probably implies an approach to saturation. Evidence for such saturation in contraction has been noted at much lower exposures, approximately 6×10^{20} nvt, and at smaller value of contraction, 1.8%, in conventional Texas coke-based material heat treated to 1370 C.

A Paper, "High Temperature Radiation Effects in Lampblack-based Graphites," was presented at the Sixth Biennial Conference on Carbon and will be published in the journal Carbon.

Irradiation of Controlled-density Graphites. Additional samples of the controlled-density series have received an initial exposure of 2234 MWD/AT at 650 C and results are in good agreement with data obtained on another set at 2639 MWD/AT. New samples included 2.10 g/cm³ density material heated to a maximum temperature of 3550 compared to a normal 3000 C. The parallel contraction was greater and the transverse expansion was smaller in the material heated to higher temperature. Similarly, lampblack-based samples heated to 3550 contracted more than the same type of sample heated to 3000 C. This result was unexpected and as yet no satisfactory explanation is apparent. It is possible that the cooling rate above 2000 C, that is, through the range in which graphite is relatively plastic, may be related to these effects. However, further exposure should delineate the behavior more clearly and supply a better basis for any analysis.

8. Boronated Graphite Studies

Neutron-flux Calculations. Detailed shielding calculations have been performed to predict heating rates and energy deposition within capsule designs currently considered for irradiating boronated-graphite specimens. Transport theory was used to calculate the photon energy flux in 11 energy groups and 40 radial zones. These calculations enable evaluation of gamma heating rates in components of the capsule and provide a firm basis for heat transfer calculations to optimize

design. Absolute values of the heating rate calculations depend upon normalization to the measured value of a flux-dependent process. Relative values, which reflect local perturbation within the system, are self-consistent and accurate to within 10%.

9. Aluminum Corrosion and Alloy Development

In-Reactor Corrosion of Aluminum. Modification of the in-reactor C-1 Loop to enable it to be used for corrosion testing of aluminum-clad fuel elements has been completed except for installation of the in-reactor test section and the tie-in of building services. These items cannot be completed until a design change has been approved by IPD. The ex-reactor portion of the loop was successfully pressure-tested and operated to test all components; it performed well.

The aluminum-clad fuel elements for corrosion testing in C-1 have been completed, except for autoclaving the last batch. Fabrication of a discharge cask was completed. It has nine inches of lead shielding and can accommodate a full 45-inch charge of elements from the C-1 Loop.

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

Fog Cooling of 19-Rod Bundles. Seven boiling burnout experiments were made with an electrically heated 19-rod bundle test section which was cooled with wet steam (fog). The test section consisted of 19 Inconel tubes, each 0.587 inch OD and 19½ inches long. A spacing of 0.074 inch between rods was maintained with ceramic spacers or warts 6¼ inches from each end. An unheated length of 25 inches preceded the heated section. A thermocouple was installed at the downstream end of each rod to measure an averaged temperature of the inside wall.

This test section was rebuilt from one which failed during the course of previous fog cooled boiling burnout experiments. It was identical in all respects to its predecessor except it did not have a flow diverter. The flow diverter used in the previous tests consisted of a ring of Al₂O₃ segments around the bundle about 14.5 inches upstream of the heated section. It blocked the coolant passage between the bundle and pressure tube wall and therefore diverted all of the coolant to the interior of the bundle at its location. One of the primary purposes of these tests was to determine the effect the flow diverter might have on boiling burnout.

Three of the experiments were done at each flow rate of 500,000 lb/hr-sq ft and 1,000,000 lb/hr-sq ft. In each case one experiment used subcooled inlet coolant; the other two used wet steam conditions at the inlet. The other run was at a flow rate of 1,500,000 lb/hr-sq ft and with subcooled inlet coolant conditions.

The raw data show a small effect of the flow diverter. For the subcooled inlet conditions the flow diverter increased the boiling burnout heat flux by 5 to 10%. For the wet steam inlet conditions the use of the flow diverter decreased the boiling burnout heat flux by about 5%.

The test section used in these experiments was nearly identical to one used in earlier tests investigating boiling burnout with subcooled inlet conditions. The test section for the earlier tests differed from the present one in that it had a shorter unheated length (about 6 inches) preceding the heated section, and it used a 0.074-inch OD wire on a 10-inch pitch on twelve of the rods to maintain spacing. These tests and the earlier tests overlapped only slightly, but the few data at comparable conditions for the two test sections show good agreement. This indicates that over the conditions studied the length of the unheated section upstream of the heated section is of small importance and that there is little difference in boiling burnout arising from the use of warts instead of wire wraps to maintain rod spacing.

The experimentation was terminated by physical destruction of the test section. Upon disassembly of the test section it was found that most of the damage was concentrated in a single rod. One of the outer rods had numerous melted patches adjacent to an inner rod

This region was about six inches long and was located approximately midway down the length of the rod. In addition, the rod was separated completely around the circumference at the downstream end as a result of overheating.

The remainder of the bundle appeared to be in good condition but will require further disassembly for a complete assessment of the damage.

Dome Seal Type Nozzle Closures. Testing of the dome seal has been completed with all requirements in the original test outline either met or exceeded. These tests are considered highly successful in establishing the qualities of the dome seal.

The seal successfully completed the following tests with no trace of leakage:

1. Cold hydrostatic tests of 150% adjusted working pressure, 2650 psig.
2. A minimum of 116 hours at steady-state conditions of 570 F and 1500 psig.
3. A total of 50 cycles at 1500 psig while the temperature was varied from 100 F to 570 F over approximately a two-hour cycle.
4. A total of 20 cycles where the dome assembly was removed and replaced between each cycle. A cycle was from ambient conditions to 570 F and 1500 psi and back to ambient.

The collection system was totally enclosed with a water-cooled condenser at the collection bottle. In a test of a different closure device this system collected 0.05 ml in slightly over 78 hours. During dome seal tests no leaks were detected with this system other than a trace collected when excessive dome deformation of 0.135 inch was encountered.

The major problem disclosed by these tests, and eventually resolved, was the flattening or deforming of the dome. Dome deformation is attributed to thermal ratcheting with relation to the holding mechanism. By installing a Belleville spring between the dome stem and retaining lugs, this seal has been found to withstand a rise in temperature of 400 F in approximately 15 seconds with zero leakage.

During the deflection analysis a second dome was fabricated, so two slightly different shapes were investigated. An analytical stress calculation was performed prior to testing. A final test report is being written.

11. Advanced Reactor Concept Studies

Fast Supercritical Pressure Power Reactor. Fuel cycle cost calculations were made for the reference core as revised during June. The fuel cycle costs presently are estimated to be 1.3 mills/kwhr. A preliminary estimate for scaling the reactor up to 1000 Mwe indicates a potential for reduction of fuel cycle costs to 0.7-0.8 mills/kwhr.

Burnup calculations were completed for the 300 MWe reference design for an average goal exposure of approximately 100,000 MWD/T. The gross power distribution shifted outward slightly from beginning to end of core life as follows:

	Percent Total Power	
	<u>Initial</u>	<u>100,000 MWD/T</u>
1st Pass Coolant	53.7	55.9
2nd Pass Coolant	25.4	26.9
3rd Pass Coolant	20.9	17.2

With low initial plutonium concentration in the depleted uranium regions adjacent to the flux trap, the reactor gains substantial reactivity with exposure due to the high conversion ratio and high reactivity value of Pu at this point. With an initial plutonium composition of 2%, a net reactivity gain of about 4.5% $\Delta k/k$ is experienced. It appears desirable to increase the initial concentration in this region and decrease that in the fast core to obtain a flatter long term reactivity characteristic.

Further study was made of the choice of pressure tubing material. Recent irradiation data published for Rene' 41 by NMPO led to a slight reduction in the allowable stress used in tube design.

A review of space requirements for thermal expansion of the coolant pipe jumpers has shown that the diameter of the pipe gallery can be reduced from 30 feet to 21 feet. Consequent reductions in the double-steel-walled-concrete-filled containment vessel will significantly reduce capital costs.

Large fast Reactor Core Design. Additional multigroup calculations have been initiated to investigate the use of moderating materials other than hydrogen for the segmented fast core concept. Results were obtained for a steam cooled core in which graphite replaces a metallic hydride region. These data indicate that a reactivity gain on coolant loss of about 5% $\Delta k/k$ may be reduced to zero with a 5.5 cm graphite region compared to about 3.25 cm of metallic hydride. The multiplication of the reactor with normal coolant flow appears to be reduced only about 0.4% $\Delta k/k$ by replacing the hydride with nearly twice as much graphite.

Plutonium Fueled Rocket Reactor. Latest estimates by Advance Fuel Development Operation on fabrication potential of the "waffle grid" honeycomb cermet fuels indicate that fuel plates up to four inches thick, and the full diameter of the core (12-14 inches),

may be feasible. A sizeable savings in pressure drop through the core should result from use of the thicker plates by reduction of the "end effect" factor (fewer plates will be used). Support of individual plates may be made easier by the use of fewer, thicker plates.

The problem of support of the fuel plates appears to be a major one in this concept. The plates must be firmly held at high operating temperatures, against a heavy pressure load, and thermal expansion must be allowed for over an extremely wide temperature range (on the order of 100 R to 5000 R) without permitting looseness or vibration of the plates. A concentrated effort on fuel support techniques is needed and is planned as part of the conceptual design studies on this reactor.

Alternate control concepts, involving (1) revolving control drums in the reflector and (2) axial movement of one or more fuel plates, are being investigated. Calculations are under way by Physics and Instrument Laboratory personnel of the nuclear behavior of a zoned fuel loading designed to skew power distribution strongly toward the inlet of the core (where hydrogen coolant temperatures are lowest).

12. Reactor Safety Engineering Test Program

Containment Test Facility (CTF). In response to an AEC request, a general outline of a proposed nonnuclear engineering test program to evaluate the effectiveness of various pressure containment and confinement principles was drafted. The program would compare capabilities of the various methods to minimize hazards of pressure, shocks, jets, impulses, and radioactive contamination resulting from reactor piping failure accidents. The proposal suggested preliminary testing of loss-of-coolant behavior on existing equipment in the Thermal Hydraulics Laboratory prior to completion of CTF, followed by suitable studies on the CTF. This proposal, if accepted, will permit immediate participation in experimental work.

Preliminary estimates were also made of equipment and siting requirements to assist in scoping the CTF.

D. DIVISION OF RESEARCH - 05 PROGRAM

1. Radiation Effects on Metals

This program is aimed at 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.

A high temperature x-ray diffraction study of the annealing of irradiated molybdenum foils has been initiated. Preliminary experiments were carried out to calibrate the power supply unit and to determine the magnitude of temperature measurement errors. The measured lattice expansion of molybdenum standards to 700 C (1292 F) and of aluminum to 500 C (932 F) were compared with thermal expansion data in the literature, and an appropriate temperature correction was derived. A sample of molybdenum containing 10-20 ppm carbon, irradiated to 10^{19} nvt ($E > 1$ Mev), is currently being studied. The room temperature lattice parameter of this sample is 3.1479 Å, an increase of 0.025% over the value 3.1471 Å which was obtained for the unirradiated material.

Some difficulty has been encountered in the x-ray measurements because of the large crystallites in the samples. The x-ray powder peaks consist of at most very few reflections from individual grains, and many reflections are absent. Satisfactory lattice parameter values can be obtained, but line breadth measurements may be impractical for this particular sample while in the high temperature mount. If this proves to be the case, the sample will be periodically removed for line breadth measurements in a more suitable fixture.

X-ray diffraction micrographs are being made of the single crystal molybdenum specimens which had previously been found unsuitable for lattice parameter measurements. The few pictures made to date show the extensive substructure of the crystal faces in a striking manner, but the work has not progressed beyond this point. A series of crystals, unirradiated, annealed, and irradiated will be examined to determine the effects of irradiation on the substructure.

The behavior of defect clusters and loops in polycrystalline molybdenum foils containing > 10 ppm and 450 ppm carbon following irradiation and annealing at 750 C (1382 F) has been investigated by transmission electron microscopy and described previously. Following the heat treatment, the small spot defect clusters present in foils containing 450 ppm carbon decrease in number and the size of dislocation loops increases. In order to determine if the large loops do interact with moving dislocations to form channels, foils which had been irradiated to 10^{19} nvt and subsequently vacuum annealed at 750 C (1382 F) were deformed. Upon deformation the foils exhibited considerable brittleness despite the high temperature anneal. Cracking occurred across the diameter of both foils, independent of

carbon content, after a strain of < 0.02 , $\Delta l/l$. Following deformation the foil containing 450 ppm carbon was prepared for transmission electron microscopy. Channels, similar to those observed in the "as-irradiated" deformed specimens, were present. The channels are straight and free of defect clusters and dislocation loops. In regions of the foil where the applied stress was low, dislocation loops are large and easily resolved; however, in regions where channeling has occurred the loops are smaller and could not be resolved. This suggests that defects have diffused from the clusters as a result of the applied stress. Since the interaction of moving dislocations with spot defects, observed in the "as-irradiated" deformed foils, is similar to the interactions of dislocations with the large loops in the annealed foils, it is believed that the spots and loops are due to the same type of defect. It is postulated that the defects involved are vacancies. Examination of a similar foil containing < 10 ppm carbon will be forthcoming.

Efforts to prepare suitable foils for electron microscopy from molybdenum single crystals are continuing. Two 1/8-inch diameter sections were cut from a crystal by spark machining and further thinned by an electrolytic wheel and jet. Foils obtained in this manner were thin enough for electron microscopy on the very edge of the foil. Further development of the final stages of thinning is required.

The polycrystalline molybdenum tensile specimens containing carbon in one of three concentration ranges have been aged after irradiation [10^{18} and 10^{19} nvt ($E > 1$ Mev)]. Aging treatments were: (1) four hours at 175 C (347 F), and (2) 16 hours at 250 C (482 F). The reduced sections of these specimens have all been photoengraved with grid lines to permit photographic recording of the deformation process during tensile testing. Single crystal specimens of molybdenum have been prepared by electron beam zone melting techniques to provide a supply of crystals for future irradiation. Single crystals of rhenium and nickel have also been prepared by electron beam zone refining techniques, and these will be utilized in a study analogous to that being pursued on molybdenum.

High purity iron foils, obtained from the Johnson-Matthey Company, were annealed at 800 C (1472 F) for seven hours and then irradiated to 5×10^{18} and 1×10^{19} nvt ($E > 1$ Mev). After irradiation the specimens were thinned for electron microscopy. Irradiation-induced defects, such as those observed in molybdenum, were not observed in any of the foils. Some spots, 100-200 Å in size, were

observed but these did not exhibit contrast change during tilting of the sample. Dislocation motion can be induced in foils of iron while in the microscope if a large condenser aperture is used. That dislocation pinning occurs in the irradiated foils can be deduced from observations made on dislocation motions. The defects responsible for pinning were not observed. Irradiated foils similar to those examined are being annealed at 350 C (662 F) to permit clustering of the defects produced by the irradiation.

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 these properties. Two areas are under study: mechanisms of phase transformations and mechanisms of deformation and recovery.

Studies of microcracking and physical damage associated with thermal cycling between the low temperature allotropic phase changes have continued. The following observations have been made: (1) no physical damage occurs during either the $\delta \rightleftharpoons \gamma$, $\delta \rightleftharpoons \beta$, or $\gamma \rightleftharpoons \beta$ transformations; (2) a small amount of the δ phase appears to be retained at 190 C (374 F) after the first cycle of the $\delta \rightleftharpoons \gamma$ cycling; (3) the amount of physical damage increases with decreasing $\beta \rightleftharpoons \alpha$ isothermal transformation temperature; (4) the amount of physical damage increases with decreasing $\gamma \rightleftharpoons \alpha$ isothermal transformation temperature; (5) fracturing readily occurs during the $\gamma \rightleftharpoons \alpha$ transformation at subzero temperature; and (6) the $\gamma \rightleftharpoons \alpha$ cyclic damage is greater than $\beta \rightleftharpoons \alpha$ cyclic damage at 26 C (78.8 F), and -23 C (9.4 F), but is much less than the $\beta \rightleftharpoons \alpha$ cyclic damage at 73 C (163.4 F). No cycling has been done at temperatures in which the β phase is the intermediate phase between the γ and α phases.

The study of the $\delta \rightarrow \gamma$ isothermal transformations employed two types of material: (1) plutonium of high quality and (2) plutonium known to contain certain impurities. Sharp discontinuities in the density versus time isothermal transformation curves were observed at the start of the transformation in the metal containing the impurities. Seven such discontinuities were observed at 101 C (213.8 F). Their number and magnitude decreased with increasing temperature. Only two were observed at 144 C (291.2), and only one was observed at 173 C (343.4 F). No discontinuities were observed in the isothermal $\delta \rightarrow \gamma$ transformation of high quality plutonium.

The pronounced surface distortion referred to as surface veining, commonly observed in unalloyed plutonium, has been identified as

bands of columnar grains. These bands form during the first 50% of the $\beta \rightarrow \alpha$ transformation and are randomly distributed throughout the metal.

The XRD-5 x-ray diffraction unit has been installed in a glove box to permit its use in the study of the crystallographic changes resulting from the deformation of plutonium. Prior to sealing the box, considerable effort has been devoted to lining up the instrument in order to make available the maximum resolving power. Although the failure of a transformer in the power supply circuit has delayed the work somewhat, it has been possible to demonstrate the feasibility of completely separating the $K \alpha_1$ and $K \alpha_2$ lines of copper as reflected from the 220 plane of silicon. This represents the resolution of two lines whose 2θ values vary by only 0.06 degrees.

A standard powder pattern for alpha plutonium has been obtained by calculation, using a computer program. This program, obtained from off-site, computes diffracted intensities from single crystal structure data and applies all corrections and multiplicity terms to produce a calculated powder pattern. Input data consisted of the unit cell constants and atomic parameters published by Zachariason and Ellinger, with atomic scattering factors from International Tables for Crystallography. This calculated x-ray pattern will be used as a random standard for preferred orientation studies by the inverse pole figure method.

The computer program has yielded the Miller indices of all possible reflecting planes together with their two theta values and relative intensities. The values show the need for resolving power. For example, with Cu radiation the 020 and the 211 lines with relative intensities of 89.98 and 100.00, respectively, occur at 37.26 and 37.42 degrees two theta. Preliminary cold rolling studies indicate that after about 90% reduction the most intense line appears at about 32.30 degrees two theta. In this region the 11 $\bar{3}$ and 201 lines appear at 32.19 and 32.31 degrees two theta. These lines have intensities of 21.62 and 32.62, respectively. It is apparent therefore that the identification with any degree of certainty of planes which are primarily involved in the deformation process requires a distinct separation of closely spaced reflections.

It has not been possible, as yet, to obtain a satisfactory correlation in line position and intensity between the pattern predicted by the computer and those obtained from cast or worked specimens. This is believed to be the result of orientation effects. In order to obtain an experimental diffraction pattern which can be accepted

as being completely representative of alpha plutonium, it is planned to utilize a sample of powder prepared by the decomposition of plutonium hydride. This should provide the most randomly oriented sample possible and provide a diffraction pattern truly representative of the undistorted alpha structure.

An investigation of fracture interfaces by metallographic examination is being conducted on unalloyed alpha and beta plutonium, unalloyed plutonium partially transformed from beta to alpha, beta stabilized plutonium, and delta stabilized plutonium. In addition to characterizing the fracture appearance as a function of material, fracture temperature, and prior metallurgical history, the investigation will provide basic information concerning the mechanisms of deformation and transformation.

Unalloyed "beta-pressed" plutonium, $\frac{1}{4}$ inch in diameter with a circumferential notch, has been fractured by impact at -196 C, -80 C, 20 C, 100 C, and 165 C (-320.8 F, -112 F, 68 F, 212 F, and 329 F). Cellulose acetate replicas of the fracture surfaces have been made for electron microscope studies. As expected, the electron microfractographs of the 20 C and -80 C (68 F and -112 F) fracture surface reveal the fracture to be primarily brittle; both intergranular and transgranular cleavage are observed. However, regions of ductile shear are also present, as evidenced by dimples elongated in the direction of relative motion of the two fracture faces. Striations and deformation twins on cleavage planes as a rule are curved.

Six tensile tests have been conducted on $\frac{1}{4}$ inch diameter plutonium rods delta stabilized with 1 w/o aluminum. Tests were conducted at -68 C, 50 C, 100 C, 150 C, 200 C, and 250 C (-90.4 F, 122 F, 212 F, 302 F, 392 F, and 482 F), respectively, with a strain rate of $2 \times 10^{-4} \text{ sec}^{-1}$. No evidence of the Portevin-LeChatelier effect was observed. However, at testing temperatures of 50 C and above, several abrupt increases in stress (= 200 psi) were recorded. The occurrence of these discontinuities is most probably explained as machine error. The possibility exists that some transformation during deformation occurs. Density measurements on the strained and unstrained regions of tensile specimens will be made.

E. CUSTOMER WORK

1. Radiometallurgy Laboratory

Examinations and Measurements. Uranium samples from NaK capsule 19 were recovered and photographed.

Samples of scale from both internal and external surfaces of 105-DR Inconel gas loop tube were prepared for x-ray diffraction analysis.

Four UO₂ thermal-conductivity specimens were metallographically prepared for x-ray diffraction analysis.

A mosaic of a complete section of a UO₂ specimen was photographed at 75X.

The ruptured UO₂-PuO₂ rod from PRTR element 5113 was examined metallographically to determine the mode of failure.

UO₂ fuel capsules GEH-14-401 and 402 were examined. The shape of the internal void was determined. Metallography and autoradiography of three sections were completed. A complete mosaic of a cross-section of capsule 402 was made at 75X.

Specimens from rods 5309N, 5307L, 5301C were vacuum impregnated with epoxy resin and sampled for metallographic and micro-chemical analysis. Replicas for electron microscopy and a silicon rubber replica were also made. Twenty-three micro-drilled samples were obtained and dissolved for chemical analysis.

Ten rods from PRTR element 1006 were sectioned and 40 samples taken for burnup analysis, fission product distribution, and metallography.

X-ray diffraction data were obtained from five samples of MgO-PuO₂ capsule experiments. Four sections of PRTR process tubes were prepared for burst testing. The ends were trimmed and length, diameter, and tube wall thicknesses were recorded. From two of the sections silicon rubber replicas were made of fuel hanger and wire wrap fretting corrosion marks on the internal surface.

The samples of cermet material obtained from Lewis Research Center (NASA) were recovered from the concrete-filled bucket in which they had been disposed and are now being prepared for metallographic examination.

Four irradiated Th-U samples were sectioned and mounted for electron microscopic and metallographic examination of the surface transverse and longitudinal to the direction of tensile drawing.

Pulse annealing furnace parameters were established for irradiated uranium specimens and six clad and unclad samples were annealed in the range 550 to 650 C for 30 minutes. One sample was annealed

300 minutes. Several of the samples so annealed were subjected to metallographic and replication analysis. X-ray diffraction data were also obtained.

Eighteen specimens from NaK samples were macrophotographed and dimensionally checked.

Metallographic examination of the first group of Zircaloy-2 tensile specimens from Phase I of the Irradiation Damage to Reactor Metals program has been completed. The first group of stainless steel specimens is under examination.

A specimen of Redox tank SX-1114 cored waste material was recovered from a core drill assembly, photographed, and shipped to the Redox laboratory for analysis.

The last 12 of 25 Hanford production fuel elements were measured to provide data concerning the relative effects of water and oil quenching during heat treatment of the metal.

Three selected Hanford production elements were sectioned and wafers for cryogenic studies were prepared.

The Chemistry Processes group of the Radiometallurgy Laboratory was particularly productive this month: 51 samples were dissolved for burnup analysis, 25 densities were determined, 11 aluminum-clad wafers were declad, 10 PRTR rods were subjected to fission gas collection and five N-Reactor inner elements were defilmed.

Project CGH-857 Physical and Mechanical Properties Testing Cell. The shielding characteristics of the cell were found satisfactory when checked by radiation monitoring personnel. The cell was put into operation on July 10, 1963. Tensile testing and weight measurement were the first procedures used.

Roll Camera Attachment for Remote Metallograph. A new roll strip camera was utilized this month to prepare two photomosaics. The new camera greatly reduces the time required to photograph a mosaic as well as improving the uniform quality of the negatives.

Density Cell. Improved consistency and greater confidence in accuracy has been obtained by new techniques in density determinations. Densities for uranium metal are now accurate to ± 0.01 gm/cc and are quoted at the temperature at which the determination was conducted. Cold standards accurate to 0.001 gm/cc in the range of irradiated uranium are being prepared for density controls.

E-Cell Metallographic Facilities. Bids received on the cell, cell door, castings, viewing plugs, and window were almost \$6,000 less than the estimate.

2. Metallography Laboratories

The amount of work performed by Metallography Laboratories continues to increase. During the month 397 samples were processed, 790 macro- or micrographs were taken, 2596 negatives (a new monthly record) were made, and 9947 prints were processed.

Addition of a "nitrogen-burst" agitation system has improved the quality of negatives and has relieved personnel from a tedious task for more important work.

3. N-Reactor Design Testing

N-Reactor Charging Machine. All testing was temporarily discontinued so that the charging machine could be readied for shipment to N-Reactor by August 5.

Preliminary design was started on a mockup which will replace the N-Reactor charging equipment for Design Test 22.

Two magazines with no taper and a minimum ID of 2.700 have been fabricated. The nozzle end of the polyethylene liner is mechanically restrained with pins.

Major modifications have been initiated to change the control and power functions of the magazine loader from pneumatic to electrical. Most equipment is on hand and the previous controls and drives have been removed.

4. High Temperature Lattice Test Reactor

Design of the mockup has been started with details of the graphite core and associated keys 80% complete; supporting graphite timbers 100% complete; front and rear doors 80% complete; and core thermocouple installations 20% complete.

5. Special Plutonium Fabrications

UO₂-PuO₂ Fuel for the PCTR. Preparation of UO₂-PuO₂ pellets for the PCTR continued. The sintered densities of pellets prepared with polyvinyl alcohol binders were low, necessitating a return to the paraffin base binder previously used. The temperature

distribution in the sintering furnace became more peaked with the nonexplosive argon-hydrogen atmosphere, requiring reduction of the batch size by half.

Al-Pu Fuel for PRCF. A special Al-Pu fuel cluster was prepared for use in the PRTR critical facility for making reactivity measurements on Phoenix fuel capsules.

6. EBWR Fuel Elements

EBWR Plutonium Fuel Elements. The powder preparation line for handling pneumatically impacted UO_2 - PuO_2 mixtures was installed. Twenty-one pounds of pneumatically impacted UO_2 - 1.5 w/o PuO_2 were prepared and processed in the powder preparation line. Seven cans, each containing 3.4 lbs of 1.5 w/o PuO_2 - UO_2 , were pneumatically impacted at 1200 C and 250,000 psi. Density of the impacted material was 10.92 g/cc (99.5% TD). Floor mounting of the 308 Building Dynapak machine used for impaction of EBWR fuel material was insufficient. This condition was corrected by installing larger mounting bolts and eliminating play in the mounting frame by welding. Process equipment includes a can cutter, powder blenders, pulverizer, a series of vibrating screens for particle sizing, and an automatic net weighing machine.

UO_2 fuel densities exceeding 86% of theoretical were obtained by a new vibrational compaction technique, utilizing a horizontal "fishtail" beam attached perpendicular to the shaker unit. Similar densities were obtained both by loading and compacting single EBWR size rods in six different beam positions and by simultaneously loading and compacting three rods at one time. Less than 15 minutes of vibration were required to attain uniform densities with either fused UO_2 or pneumatically impacted UO_2 . Optimization of particle sizing and refinement of loading and compaction techniques is under way.

The first 600 tubes for the EBWR UO_2 - PuO_2 loading were received. Ultrasonic and eddy current nondestructive testing of the tubing will have to be done at Hanford because adequate facilities are not available at the vendor's plant. Another 600 tubes are due by August 1.

Beneficial use of new vibrational compaction is scheduled for August 1, 1963.

Irradiation Testing of EBWR Prototypic Fuel Rods. Cladding components were fabricated for 36 capsules (GEH-14-494 through GEH-14-529) containing EBWR diameter rods incorporating fuel of the new composition, U^{235}O_2 - 1.5 w/o PuO_2 . Tentative arrangements were made to charge 12 of the capsules into the ETR (Cycle 57 or the MTR (Cycle 196)).

EBWR Fuel Rod Tubing. A contract with an off-site vendor for EBWR fuel rod tubing was modified to permit nondestructive testing at HAPO. Six hundred tubes were received and a like number is scheduled to arrive by August 1. Nonavailability of a suitable furnace for the final anneal is holding up the shipment. Preliminary HAPO ultrasonic tests of approximately 50 tubes indicated that 20% of them contained defects. These defect indications have not yet been verified as actual defects. Eddy current tests also are being established. Preliminary tests indicate that the relatively rough surface finish of the tubing make it difficult to discriminate real defects with a high degree of sensitivity. However, preliminary tests have picked up several internal surface defects which were later verified with ultrasonic tests. Also, tubing cleared by the ultrasonic test passed the preliminary eddy current test.

J. W. Woodfield

For Manager

Reactor and Fuels Laboratory

PHYSICS AND INSTRUMENTS LABORATORYMONTHLY REPORTJULY 1963FISSIONABLE MATERIALS - O2 PROGRAMREACTORN-Reactor Exponential Experiments

The analysis of the control strength of samarium ball channels in the wet mockup exponential pile is complete. The strength is roughly 75% of the horizontal control rod strength in the test lattice, but will be about 90% of the HCR strength in the actual lattice.

N-Spline Measurements

PCTR measurements were begun and completed during this month on the reactivity worth of splines in wet N Cross-coolant channels. They will be used at N-startup to measure full-pile control rod strengths. The changes in k_{∞} have been evaluated for the drilling of graphite for the cross coolant tubes, the addition of dry Zr-2 tubes, water, splines, and the removal of water from the splined channels. Finally, the reactivity worth of a spline in a wet process tube has been measured. This information is being written into a report for internal distribution.

Optimization of Re-tubed Lattices

Final analysis of data has been completed and an informal document HW-78320 has been written. The results show that CVIN fuel is slightly undermoderated in the C lattice and that the control strength of the vertical safety rod for this type fuel is constant regardless of pile flooding or loss of coolant.

K-Lattice PCTR Experiments

All experimental work has been completed at the PCTR for obtaining absolute k_{∞} values for KVNS fuel pieces in Zr-2 process tubing in the K lattice.

The foil data for the wet, dry, and splined cases have all been reduced to gold and copper traverse data, and half of this has been further corrected to an equivalent $1/v$ detector traverse.

Multi-Thermal Group Scattering Transfers and Cross Section

Programming and debugging of program SPECTRE S is complete. SPECTRE S calculates group scattering transfers and cross sections and produces punched card output for program S. Provision for punching output for programs HFN and THERMOS is available with the addition of two simple sub-routines. An informal document describing the use of SPECTRE S is in preparation.

Instrumentation

Experimental testing of one gamma energy spectrometer of the NPR Fuel Rupture Monitoring System was continued. Data obtained from tests at KE Reactor show no particular gamma energy peaks in the coolant water; however, it appears that some gamma interference to the spectrometer detector is being obtained from a neighboring cell. Extra shielding will be installed to reduce the interference.

Purchase specifications are being revised for the proposed solid state multi-channel pulse height analyzer for use at NPR.

The project to provide special instrumentation for the initial fuel loading and physics tests at NPR continued actively. A counting system was planned and specifications were prepared. Neutron detector bids were reviewed. Work was started on the experimental testing of the neutron source range process instrumentation.

Progress continued on the experimental Fuels Testing Loop being installed at PRTR. An order for six chart recorders was placed. Because of lack of acceptable bids, the specifications for the two scintillation spectrometer systems must be revised and reissued. A meeting was held to discuss proposed loop tests and experiments. The importance of obtaining quality background data on the system was emphasized. The nature of the first group of planned experiments was established. Consideration is being given to obtain an additional system to monitor particulates in a separator or filter. Emphasis was given the problem of monitoring gaseous fission products. It appears that some 20 to 24 months of specific fuels tests have been planned for the loop to date with other tests following depending on results obtained.

Engineering and physics assistance was rendered to Equipment Development, IPD, regarding the possibility of determining the absolute sensitivity of the gamma chambers under irradiation at KE Reactor. In addition, methods of determining the neutron flux within irradiation facilities and methods of relating the flux to adjacent tube powers using computer techniques were

discussed with Operational Physics, IPD.

Shielding

Specific shielding calculations and informational assistance was rendered: Reactor Physics, NRD, on N Reactor shield material; Operational Physics, IPD, on masonite shielding; and Irradiation Testing, IPD, for an estimate of the curie content of a cask based on the gamma intensity through the cask wall.

System Studies

N Reactor pressurizer simulation studies are approximately 50% complete. Appropriate controller settings as determined by earlier computer runs were used to determine the effects of malfunctions on controller stability. Test runs made in real and accelerated (by a factor of ten) time compare favorably.

N primary loop control studies have been started with the analysis of a system which includes the primary pumps, simplified reactor kinetics, and an elementary model of the steam generator.

Reruns of certain portions of the NPR kinetics studies, which were found necessary after analysis of the initial runs, have been completed and forwarded to personnel in the N Reactor Department.

Discussions were held on the feasibility and desirability of measuring the ratio of delayed neutron fraction to prompt neutron lifetime in the NPR. Discussions are being continued on problems of detector efficiency, large reactor noise theory, and calculation of expected results.

Consultations were provided N-Reactor Department on use of noise techniques for determining reactor parameters and analyzing vibrations, on the development of process tube monitoring equipment, on fuel rupture monitoring equipment, and on determining computing and data processing needs for the Department.

SEPARATIONS

Experiments with Plutonium Oxide-Plastic Mixtures

Subcritical neutron multiplication experiments have continued with the Remote Split-Table Machine using PuO_2 -polystyrene compacts as the fuel. These compacts have a plutonium density of 1.14 g/cc and a hydrogen-to-plutonium ratio of 15.

During the month preliminary critical mass data were obtained with the plastic cubes stacked in the form of rectangular prisms with square base dimensions. The assemblies were reflected to various degrees with 6-in. thick Lucite. The following measurements were performed:

1. For evaluating nuclear safety in hand stacking the RSTM, data were obtained for estimating the reflector worth of a person adjacent to an assembly. An "effective man" was simulated by completely covering the two table halves with Lucite to a depth of six inches on which the array rested.
2. The critical height and mass were determined for an assembly with a 12-in. square base dimension reflected with Lucite on the bottom, and on both the top and bottom surfaces, and with the assembly fully reflected on all surfaces.
3. The critical height and mass were determined for an assembly of 8-in. square base dimension fully reflected with Lucite.

The data obtained to date with the PuO_2 -polystyrene compacts, including the results previously given for the unreflected assembly are summarized in the following table:

Critical Height and Mass of Pu in Rectangular
Prisms of Square Base Dimensions

(PuO_2 -Polystyrene Compacts; $H/\text{Pu} = 15$; Pu Density 1.14 g/cc)

<u>Base Configuration</u>	<u>Reflector (Six Inches of Lucite)</u>	<u>Critical Height (Inches)</u>	<u>Critical Mass (Kg Pu)</u>
12-in. x 12-in.	None	12.5 ± 0.04	33.7 ± 0.1
12-in. x 12-in.	"Effective Man" on Base	9.82 ± 0.1	26.4 ± 0.3
12-in. x 12-in.	Base	10.3 ± 0.05	27.7 ± 0.1
12-in. x 12-in.	Base and Top Surface	8.82 ± 0.04	23.7 ± 0.1
12-in. x 12-in.	All Surfaces	4.94 ± 0.2	13.3 ± 0.6
8-in. x 8-in.	All Surfaces	9.44 ± 0.1	11.3 ± 0.1

The first critical mass value given in the table corresponds rather closely to an unreflected cube, whereas the last value is not expected to depart much from a reflected cube; but note the large reduction in critical mass on reflections of the assembly, a factor of 3. The above results would imply a reduction of about 2.3 for spherical geometry.

The uncertainties shown in the table are standard errors determined from the multiplication measurements with three different neutron detectors during the experiments. The errors do not reflect the uncertainties in individual data points used in the least squares fit analysis of the data, nor do they account for any curvature in the inverse multiplication curve which might have occurred if the arrays had exceeded the 85 percent of critical mass limit permitted in these experiments.

Critical experiments will be performed when approval of the Hazards Summary Report for the RSTM is received from the AEC.

Relative Worth of Plutonium Atoms to U^{235} in a UO_3 -Water Mixture

The minimum enrichment necessary to obtain a k_{∞} of unity in a UO_3 -water mixture is known from experiments in the PCTR with slightly enriched uranium to be 1.034 ± 0.01 weight percent U^{235} . With this enrichment the maximum value of the reproduction factor (unity) is obtained with an H/U atomic ratio of about 4.5.

From a nuclear safety viewpoint it is also desirable to know the amount of plutonium atoms that one would need to add to a natural uranium UO_3 -water mixture to obtain a k_{∞} of unity under optimum conditions of moderation.

The relative worth of plutonium atoms can be defined as the number of U^{235} atoms that must be added to a natural UO_3 -water mixture to give a k_{∞} of unity divided by the number of plutonium atoms that must be added to a similar mixture to give a k_{∞} of unity. Calculations have been made with the GAMTEC code to determine the relative worth of Pu^{239} , and Pu containing 5 w/o Pu^{240} .

It was found that the relative worth of plutonium atoms with no Pu^{240} was 1.99, whereas the relative worth with 5 percent Pu^{240} concentration was 1.80. The ratio of Pu atoms to natural U atoms for which criticality is possible in an aqueous homogeneous system is 0.0016 and 0.0018, respectively.

For a number of years the always safe ratio of plutonium-to-uranium has been based on a number 0.0018. The ratio considered safe has been 0.0017 (1500 g Pu/ton of U). The recent calculations show that criticality would be possible with a ratio of 0.0017 in the absence of Pu^{240} and nitrate, etc., which implies the margin of safety has been smaller than previously thought, i.e., safety would not be ensured by the Pu concentration alone but by other factors. A Monte Carlo calculation will be made to firmly establish the above numbers.

Buckling of Truncated Spheres

Debugging has been completed on a short program which calculates analytically the flux at any point near the top boundary of a truncated sphere by using a double quadratic fit. It was found that an error in the averaging could be eliminated by using a cubic fit rather than a quadratic fit through the longest radial cut. The program which then averages the flux is currently being debugged. Debugging is complicated by the complex integral functions that are encountered.

INTERSET Code

The INTERSET cylindrical interaction code has been debugged, and comparison with experimental data is continuing. A formal report which describes this work has been written for inclusion in the Physics Research Quarterly. A letter has been sent to General Dynamics mentioning the possibilities as used by INTERSET of obtaining analytical integrals for the double cosine solid angle functions for various geometries. General Dynamics has been using numerical integration in a general type computer code.

Instrumentation and System Studies

Safety circuit tests were continued at the Critical Mass Laboratory to determine the delay time required to scram the split-table assembly. The tests include determining time lag from controller trip to the end of rod travel for the two vertical poison rods. Also, each channel of the safety circuit was tested for response time by simulating a step input of pulses in Channels 2 and 3, and generating a positive period input for Channels 1, 4, 5, and 7. A positive period as low as 0.1 second was generated with two operational amplifiers in a rising exponential setup. The simulated inputs were placed at the detector inputs to the channels. The time measured was from the simulated input to the high trip point on the controller for Channels 1 and 4 and from the input to the period trip for Channels 5 and 7. The split-table delay time was determined from the controller trip to the start of table opening.

The scintillation detector channel at the Critical Mass Laboratory was determined to be satisfactory for the measurement of dynamic reactor parameters. A power spectral density analyzer, using the Applied Dynamics analog computer, was assembled and checked with good results. The optimum window and data filters are being investigated. A digital program was written, debugged, and operated on the IBM-7090. The program computes the expected power spectra for reactors as a function of reactivity and lifetime with arbitrary delayed neutron parameters.

Consulting Services on Nuclear Safety - Criticality Hazards1. Nuclear Safety in HL

Revised specification K-1, dated June 4, 1964, was issued for the PRTR Operation.

Nuclear safety limits were submitted to Fabrication Development for Natural UO_2 containing 1.0 w/o PuO_2 .

Development work continued on the new Phase II storage area for the Critical Mass Laboratory. The concept is unique in that the storage compartments will be essentially isolated from the standpoint of neutron interaction. On this basis, the units in the storage array may be positioned closer together than otherwise, and the number of units will not necessarily be limited.

2. Nuclear Safety in CPD

Participation on the Hazards Committee for Recuplex Deactivation continued. Procedure A-47, "Dismantling of J-26A - Phase II", was completed. The top sections of the vessel were cut off, and some of the packing removed. Approximately 9 kg of plutonium were removed. Efforts are now being directed toward the one remaining problem--the removal and storage of the plutonium bearing sludge on the R and B Hood floor.

3. Nuclear Safety Audit of APED Facilities at San Jose and Vallecitos

A nuclear safety audit was made of APED facilities at San Jose and Vallecitos on July 1-3, 1963. Critical mass control procedures were reviewed and processing and storage areas were inspected. This audit was requested by APED through the AEC.

NEUTRON CROSS SECTION PROGRAMScattering-Law Measurements for H_2O at 95°C

Measurements of the doubly-differential scattering of slow neutrons from H_2O at 95°C were continued on the triple-axis spectrometer. The energy resolution of the spectrometer was improved by a factor of two for better-resolution measurements on the scattering with large energy-change. Measurements have been taken with the improved resolution for the five largest values of energy change previously studied with lower resolution. These measurements and the analysis of the results are not completed.

A series of Scattering-Law calculations employing the computer programs LEAP and ADDELT is under way. Preliminary calculations which are fitted to existing data on the Scattering Law for H_2O at $20^\circ C$ show pronounced structure in $S(\alpha, \beta)$ vs α which has not been observed experimentally. The higher-resolution measurements of S' , now in progress, are designed to show this structure if it is real.

Time-of-Flight Spectroscopy for Slow Neutrons

Development work is in progress on the components of shielding, rotor mounts, scattering chamber, etc., to utilize the 4-B hole at 105-KE for slow-neutron inelastic-scattering measurements by time-of-flight. Work was continued on the data-reduction computer program for the time-of-flight system.

Tests continued of the speed and phase stability of test rotors. The Hypersync motors under test show excellent speed stability, but the present motors are not capable of driving a 30-lb. test rotor to full speed. Negotiations are under way to obtain a more powerful model of this motor.

Studies of the characteristics of helium-filled slow-neutron detectors continued for several different gas fillings.

Testing and debugging of the 6144-channel time-of-flight analyzer were completed.

Fast-Neutron Cross Sections

The data from the June measurements of the total cross sections from 3 to 15 MeV of 22 elements have been processed on the data-reduction program. Work is in progress to attempt to correct part of the data for bias and/or beam-monitor changes which occurred during the second week of measurements.

Requests for total cross-section results were received from BNL, KAPL, Grumman Aircraft Engineering Corporation, Stanford University, and the University of Auckland during the month.

Samples of Sc, Eu, Tb, Tm, Lu, Ba, Ce, and Nd were prepared for future measurement of total cross section.

REACTOR DEVELOPMENT - 04 PROGRAMPLUTONIUM RECYCLE PROGRAMHigh Exposure PuAl Lattice Studies

The measurements in the PCTR, using 19-rod clusters of 20.6% Pu²⁴⁰ material in graphite, have been completed. The reactivity and foil activation data for the 6-1/2, 8-3/8 and 10-1/2 inch pitch lattices have been compiled. This data is now being summarized for inclusion in the Physics Research Quarterly Report.

Low Exposure PuO₂UO₂ Lattice Studies

The fabrication of the fuel elements for these studies has stopped pending better control on the pellet sintering process. Planning has begun for the measurements using this fuel in 19-rod clusters on 6-1/2, 8-3/8 and 10-1/2 inch spacings in graphite in the PCTR. These measurements yield the best results when each cell of the 3 x 3 array is poisoned at the cell boundary. Thus drawings of the required graphite are being prepared.

Approach-to-Critical Experiments

Experiments on the 0.66 inch lattice using 2.0% PuAl fuel with 16% Pu²⁴⁰ content have been finished. The total number of rods available was only 72% of the extrapolated critical number of rods which was 1380. The accuracy of this number is probably not much better than 10% because the loading was so far from critical. The data for reflector savings and buckling are still being analyzed.

Drawings of the templates and loading structure for approach-to-critical experiments with PuO₂-UO₂ fuel rods have been made. The drawings have been submitted to the shop with a work order for the fabrication of the equipment. The templates and loading structure are similar to those which will be used in the PRCF so that the templates can be used in the PRCF as well as in the approach-to-critical tank.

Plutonium Recycle Critical Facility

The PuAl substitution experiment, Experiment 21, was completed this month. Preliminary analyses of the data result in a value of 14.7 mk for the worth of the PuAl fuel element in the center of the PRCF relative to heavy water. During the course of the experiment other worths were also determined. The following tentative values were obtained.

<u>Fuel Element</u>	<u>Worth</u>	<u>Positions</u>
UO ₂ in ring 0	1.52 mk	(509)
3 PuAl in ring 8	2.2 mk	(517,905,105)
3 PuAl in ring 8	2.7 mk	(501,113,913)

The difference in the last two values for the PuAl elements results from their different environments in the PRCF. The positions 517, 905 and 105 are near control rods.

Safety rod worths were determined and were found to be different from those measured earlier. However, they are sufficient to satisfy the process specification requirements. Control rod worths were also determined and it was found that they are much less sensitive to changes in environment than the safety rods.

The drawings for the modification of the PRCF for H₂O moderator have been reviewed and approved.

The 1/4" long Co-Al flux monitors from PuAl fuel element 5092 which was irradiated in the PRTR have been received and have been removed from their Zircaloy jackets. This is the last set of the three sets of monitors which are available. Three more sets are still attached to the fuel elements. These monitors will not be removed until after the fuel elements have been used in experiments in the PRCF.

Measurements of the reactivity coefficients in the PRCF of highly irradiated PuAl samples irradiated in the MTR will be interpreted by means of two group perturbation theory. The measurements will be performed in two different environments where the quantities W, the ratio of the sensitivity to fission to the sensitivity to absorption, are as different from each other as feasible. A number of calculations have been made using the SWAP code varying the detailed neutron environment near the samples. For all cases the core of the PRCF was loaded in a two-zone configuration of 21 UO₂ and 27 PuAl fuel elements. The 3 safety rods and 3 control rods were in ring 3 and ring 5, respectively. The results show that the highest value for W was 1.174 and was obtained when the center cell contains the fuel-element thimble, the thimble guide tube and one 1/2" O.D. PuAl rod in the center of the thimble surrounded by light water. The lowest value for W was 0.6 and was obtained with a 19-rod PuAl cluster and no moderator in the thimble. A description of these proposed tests has been written for approval. The 19-rod PuAl fuel element which will allow small samples to be placed in its center has been fabricated. A special hanger for the cluster has been designed and fabricated. The hanger has a hole through it, and will allow the removal of unirradiated samples from the cluster without transferring the

entire thimble out of the PRCF and into the PRTR basin.

Scattering Law for Water

Programming is complete for program 2GUSW1, which produces the scattering law from analytical fit to the spectral function for water. Two resonances are treated exactly for all energy transfers. Debugging of 2GUSW1 is partially completed.

PRTR Burnup Experiments

Analysis of the burnup data from elements 5051, 5095, and 5092 is continuing. The data indicate a lower α for Pu-239 than preliminary data indicated, and an experiment designed to explain the difference is being proposed.

Modified Heavy Gas Equation

A numerical solution of the modified heavy gas equation is being developed using the methods described by Leslie for the SPECTROX 1 program. These methods permit the study of the modified heavy gas equation when applied to uranium, plutonium, or plutonium-uranium systems. Analysis of the methods is complete, and a FORTRAN program for the 7090 computer is currently being written.

Fission-Neutron Energy and Time Spectra

In support of advanced design work on the kinetics of plutonium-fueled power reactors, a flexible computation of fission-neutron energy and time spectra was formulated for the digital computer (GE-HL program E). Input consists of the energies E_1 and E_2 modifying Watt's prompt neutron energy spectrum.

$$S_0(E)dE = \left[(4E_1/\pi E_2)^{1/2} - E_2/4E_1 \right] e^{-E/E_2} \sinh(E/E_1)^{1/2} dE/E_2, \quad (1)$$

the delayed neutron mean energies $\bar{E}_j$ for Bonner's delayed neutron energy spectra

$$S_j(E) dE = 6(1 - E/7 \bar{E}_j)^5 dE/7 \bar{E}_j, \quad (2)$$

the delay fractions β_j needed to assemble the composite prompt-plus-delayed spectrum

$$S(E) dE = \left[\beta_0 S_0(E) + \sum_{j=1}^J \beta_j S_j(E) \right] dE, \quad (3)$$

the output energy lattice, the maximum tolerable fractional error in the output, and the maximum tolerable absolute error in the output.

Output lists and punches the multi-energy band-integrated discrete representation of the prompt spectrum (1), the delayed spectra (2), and the prompt-plus-delayed spectrum (3), ready for use with multi-energy diffusion or transport analyses of reactor kinetics. Spectral data of this sort have been computed for thermal fission, fast fission, and spontaneous fission of the isotopes Th-232, U-233, U-235, U-238, Pu-239, Pu-240, Pu-241, and Cf-252.

The Impact of the Physics Program on Fuel Cycle Analysis Accuracy

The exploration of the impact of uncertainties in physics information upon the economic factors in total fuel costs for power reactors is under way. Two aspects of the penalties are being explored: the effect and cost of under-enriching the reactors, and the effect and cost of over-enriching the reactor. Evaluation is in terms of fuel producer penalties assessed for failure to meet a fuel warranty (burnup), and in terms incremental total fuel cost to the reactor user.

Heterogeneous Experiments - 1.8% Pu-Al

The Pu-Al critical experiments that were analyzed and reported last month have been recalculated utilizing the Nelkin scattering kernel for H₂O in the THERMOS calculation. The results are generally smaller values of k_{eff} , ranging from 1.0117 to 0.9858 as opposed to the previously reported 1.0162 to 1.0023. The spread from maximum to minimum k_{eff} is larger by a factor of 2. The reasons for this and possible remedies are being explored.

Uranium Systems

Two uranyl-nitrate spherical critical experiments have been analyzed as a part of the cross section - calculational model checkout. The GAM-TEMPEST-HFN method was employed. The TEMPEST cross sections were compared with those obtained by a SPECTRUM calculation using Lindenmeier's modified water kernel. The results agree. These are the same two experiments that were analyzed and reported in the May 1963 monthly report. The calculated multiplications agree well with experiment ($k_{eff} \text{ exp} = 1.0$), and those calculated by the RBU model though both calculations yield k_{eff} 's a little low for the boron containing system.

EBWR Demonstration Experiment

The critical loadings in the PRCF for the 1.5% EBWR fuel have been calculated using the Nelkin scattering kernel and compared to the predicted loadings, as reported last month, which were based on the B.S.J. (Brown St. John) scattering model for H_2O . The calculated resonance interaction effects between the uranium and plutonium resonances are significantly different between the two models. The results show the predicted number of rods for critical to be one to four percent lower using the Nelkin kernel except for large water to fuel ratios where the result is approximately ten percent higher.

A calculation of the expected critical loading for the 1.5% enriched fuel in the EBWR environment (i.e., zirconium-spacers, control rod followers, boxes, etc., at the water/fuel = 2.75) has been made. The GAM-THERMOS-TEMPEST-HFN calculational model was again employed with the zirconium and water corresponding to the EBWR core being accounted for in the cell calculations. The k_{eff} of the 36 box configuration (in cylindrical geometry) is found to be 1.13. This is to be compared with 1.18 and 1.20 found for the 2.5% PuO_2 loading under similar conditions during the initial study at ANL, and locally using the MELEAGER burnup code, respectively.

Phoenix Reactor Concept Studies

The control rod calculations for a plutonium-fueled, thermal reactor have been completed. An informal document, HW-78332, "Control Rod Calculations for a Plutonium-Fueled Thermal Reactor," is being issued. The purpose of these calculations was to adapt some of the standard techniques developed for fully enriched uranium reactors to the analysis of plutonium reactors; to examine the effect of certain approximations, and to make qualitative comparisons of control rod effectiveness in various plutonium and uranium environments. An over-all assessment of the applicability of the techniques used in these calculations can really not be made, until at least some experimental verifications are available. However, within the context of the methods used in this study, some of the simpler approximations appear to give results which do not differ materially from those arrived at by somewhat more complicated numerics. The use of the simpler methods is recommended for survey calculations. Finally, a comparison of control rod effectiveness indicates a fair reduction of control rod worth in the Pu cores as compared to similar U-235 cores.

A series of calculations have been started to investigate the effect of Xe-135 and Sm-149 poisoning on plutonium cores, and to make comparisons with similar U-235 cores. The high Pu cross sections should make the Xe-135 poisoning less severe in the Pu core than in the U-235 core.

A study to determine the sensitivity of fuel burnup behavior to cross section changes has been started. A burnup calculation using the updated GAM and TEMPEST libraries reported in the April 1963 monthly report will be compared to a run using the libraries of the previous PHOENIX studies. A Mark I core with zero thickness plates and 70 kg Pu load consisting of 20% Pu-240 was "burned" using constant cross sections calculated at time zero. The initial, first minimum, and peak reactivities are given below. The higher hump is due primarily to the lowering of the capture-to-fission ratio of Pu-241, making the bred-in Pu much more reactive than in the old libraries.

	<u>Initial</u>	<u>Minimum</u>	<u>Maximum</u>
Old	1.048	1.025	1.078
New	1.062	1.029	1.127

The hump in the reactivity curve is expected to drop when the burnup calculation is redone using burnup dependent cross sections rather than the preliminary methods used here. It is clear that the time dependent k_{eff} values for the kind of Phoenix cores considered here are extremely sensitive to the cross section input data.

The use of the MTR as a possible irradiation facility for a complete Phoenix core is being studied. Re-analysis of the plutonium loaded MTR experiment performed in 1958 has been started. Based on this analysis, experiments using Pu fuels more nearly matching Phoenix fuels will be considered within the constraints of the MTR. While such an experiment might not yield optimum neutron spectra or fuel designs for evaluating the Phoenix behavior, an off design burnup experiment might provide, nevertheless, useful data for compact, fully enriched Pu cores.

Measurements of reactivity changes of PuAl samples having 16.33 w/o Pu²⁴⁰ were made in the ARMF as a function of time after irradiation in the MTR. Measurements on samples irradiated to 0.28×10^{21} nvt and 2.0×10^{21} nvt were used to calculate the changes in the poison and fission cross sections of the sample caused by radioactive decay of fission products.

Fast Reactor Plutonium Utilization Studies

As part of the critical mass survey of plutonium- and uranium-fueled compact, cermet reactors, a series of check calculations have been carried out. In the first comparison, reactor core cross sections were generated by means of GAM-I and compared to an HRG ("Hanford Revised GAM") generated set. The resulting k_{eff} values for a Be reflected, 50 v/o void 50 v/o fuel core are shown below.

UO₂ Core

<u>Cross Section Set</u>	<u>Core Radius (cm)</u>	<u>k_{eff}</u>
HRG	13.88	1.000
GAM-I	13.44	1.000
GAM-I	13.88	1.032

PuO₂ Core

<u>Cross Section Set</u>	<u>Core Radius (cm)</u>	<u>k_{eff}</u>
HRG	9.67	1.000
GAM-I	9.86	1.000
HRG	10.0	1.034
GAM-I	10.0	1.014

In a second comparison, group cross sections have been generated assuming an infinite medium; another set was generated with estimated leakage effects taken into consideration. The resulting k_{eff} values for the 50/50 PuO₂ core are tabulated below:

<u>Cross Section Set</u>	<u>Core Radius (cm)</u>	<u>k_{eff}</u>
Infinite Medium	9.86	1.000
Finite Medium	9.90	1.000

Finally, a comparison was made of cross sections generated in the presence and absence of a heavy scatterer. The k_{eff} were calculated for a PuO₂ core containing 50 v/o void, 25 v/o PuO₂, and 25 v/o W.

<u>Cross Section Set</u>	<u>Core Radius (cm)</u>	<u>k_{eff}</u>
Avg'd with Scatterer	17.17	1.000
Avg'd without Scatterer	17.10	1.000

The conclusion to be drawn from this brief study is that the cross section modifications examined here do not have a marked effect on calculated reactor size, and that the simpler calculational methods are generally adequate for survey purposes.

Code DevelopmentALTHAEA-MELEAGER

Thermal flux depression factors for a fuel rod in light water have been obtained with S-4 code and fitted with the formulation used in the ALTHAEA code and described in the April monthly report. The formulation will reproduce the computed values to within $\pm 2.5\%$ for a range of absorption cross section of the fuel from $\Sigma_a = .01$ to 20 and $\Sigma_s = 0$ to 0.5 for the fuel. The values obtained apply to a one-half inch diameter fuel rod with a moderator-to-fuel volume ratio of 1.48. The values recommended for use in ALTHAEA are:

Fuel Σ_s =	0.000	0.084	0.370	0.500
FCONST	1.2046	1.2046	1.2046	1.2046
FEXP	0.4900	0.4745	0.4215	0.3974

RBU

Flow charts of the Monte Carlo portion of RBU are now complete and following a final review will be sent to duplicating for inclusion in the final system documentation. Work is still being done on the system documents, including separate development of the RBU Basic Cross Section Library and its supporting programs.

A meeting sponsored by the ANS Reactor Mathematics and Computations Division designed to establish a master nuclear data library to be available on a nation-wide basis was attended, and the RBU Basic Library offered as a starting point in this work. A subcommittee meeting will be held at Hanford, September 18, 19, and 20 to arrive at final details.

The double-mass approximation to the scattering law for bound scatterers is being refined to produce more accurate estimates of the scattering mass variation with energy. Plans for evaluating the model have been established.

HRG

Two nuclear data tapes have been prepared for HRG, the Hanford Revised GAM code. One of these contains the 127-nuclide GAM-I data transcribed in the format needed by HRG. The second contains data for 175 nuclides derived principally from the RBU Basic Library by means of a specialized version of BARNES. HRG has operated satisfactorily on a test case using the first data tape, but has not run entirely successfully on the same test case using the second tape. The trouble has been traced to that portion of HRG

used in calculating resonance integrals, where some of the new data on the second tape strain the present capabilities of the code.

In closely related work, RE 271, the ANL version of GAM-I, has been obtained to provide cross checking of calculations involved in the joint HL-ANL EBWR program. RE 271 and the associated code for updating the nuclide data tape have been adapted to the 7090, the ANL 131-nuclide data tape has been generated, and a test case run successfully.

Physics Chain

Revisions have been made to programs in the Physics Chain to use the new version of LILLEY2 and the composite LILLEY tape. This arrangement significantly reduces tape manipulation in the standard operating sequence. Test cases are presently being run to insure compatibility between these revisions and the chain system.

Off-Site Distribution of Hanford Machine Analyses

In response to specific requests accrued during the last few months, the latest version of Hanford Laboratories' machine analysis of multi-energy transport theory (GE-HL program S-XI) was shipped to the Phillips Petroleum Company (Idaho Falls, Idaho), to the G. E. Flight Propulsion Laboratory (Evendale, Ohio), and to the United Kingdom Atomic Energy Authority (Harwell, England).

Mass Spectrometry

Isotopic analyses in support of Plutonium Recycle Program studies were continued. Completed isotopic analyses were reported for PRTR fuel element No. 5051 (HW-78047), PRTR Fuel element No. 5092 (HW-78049), and macrodrill samples from PRTR fuel element No. 5095 (HW-78048). Isotopic analyses were also completed for 18 macrodrill samples and 3 burnup samples from PRTR fuel element No. 5108.

Instrumentation and System Studies

The analog, one-node simulation of the Plutonium Recycle Critical Facility was debugged and is now working properly. The simulation uses six delayed neutron groups and includes provisions for scram tests and temperature feedback. Automatic range scaling is provided in two decade steps by means of point storage memory. Circuits are included to represent the use of uranium or plutonium fuel or both. Runs were made representing operation with plutonium fuel only. Information necessary for operation with uranium fuel only was received for subsequent analog computer runs.

Plans were started for tests concerned with the activation of the PRTR controller. An analog simulation of the kinetics will be made on the portable Applied Dynamics analog computer for use with the actual moderator level control system for making initial tests and adjustments. The tests will permit evaluation of procedures for activating the controller, provide training on the use of the controller, and will be useful in determining the dynamic characteristics of the entire system.

Most of the required multiplier phototubes were received for evaluation and experimentation directed toward general improvement of the PRTR Fuel Failure Monitor.

The line filters and one scintillation detector, with attached dynode voltage divider, were received for use in the PRTR Underwater Gamma Scanner for fuel rod and activated wire measurements. The isolation transformers and the second detector have not arrived; however, adequate equipment is available to start operation when the mechanical assembly is made available by PRTR personnel. The three installed liquid effluent monitors for gamma emitters continue to perform correctly at PRTR. All instrument drawings have been completed.

HIGH TEMPERATURE REACTOR LATTICE PHYSICS PROGRAM

In order to evaluate the corrosive effects that nitrogen and graphite may have on alloys planned for use in the HTLTR, some tests are to be carried out in the Fuel Design Operation laboratory. The tests will include heating of samples of Hastelloy-X, nickel, inconel 625, TD-nickel, and molybdenum to 1000°C in nitrogen and in contact with graphite. Metallographic, chemical, and stress analyses will be made after a heating period of 200 hours and again on additional samples after 1000 hours.

An initial study, for cost and scoping purposes, is being made for a proposed mockup of the HTLTR. The structure will contain a graphite block 10 feet long and about 22 inches square surrounded by thermal insulating layers comparable to that planned for the HTLTR. Provision will be made for inserting graphite heating elements, control and safety rods, and thermocouples that match the proposed design for the HTLTR. The purpose of the mockup will be to test the performance of these items up to temperatures somewhat in excess of the temperatures to which they will be subjected in the HTLTR. These tests will be carried out throughout the design period.

Design work on the HTLTR and facility has been started by the Architect-Engineer.

NEUTRON FLUX MONITORS

Analysis work continued in an effort to derive the relationship between the cadmium ratio and the spectral parameter, r , to confirm the validity of the equation used. The experimental evaluation of r , T , and ϕ is completed. A report is presently being prepared as a classified informal document.

The first group of regenerating (Phoenix) detectors, irradiated in the KE Reactor magazine facility, was discharged and de-canned. The units are presently undergoing chemical separation in preparation for mass spectrometer analysis. A computer program was run to calculate the anticipated behavior of the samples which contained 28.4 percent Pu-239 and 71.6 percent Pu-240. When the mass spectrometer data are obtained, another calculation will be made to include the effect of Pu-241 content, if any, in the detectors. Work was started on a formal report regarding the regenerating detectors; however, questions regarding the proper security classification of some pertinent data have not been resolved.

Because of inability to secure a suitable water-cooled reactor test facility, one Boron-11 beta-current detector unit was oven tested for insulation resistance changes from 20°C to + 650°C. The end point resistances were one teraohm at 20°C and two kilohms at 650°C. The 650°C value, although low, will be adequate if the measuring circuitry has an input resistance of less than one hectohm. Further testing was planned in an uncooled facility and will be conducted when a suitable low resistance measuring circuit for one nanoampere can be obtained. Two similar Boron-11 units are also ready to test. In addition, experimental fabrication was continued on two other Boron-11 detectors to determine optimum assembly techniques. The initial attempt resulted in a short circuit due to diffusion of weld scaling material back into the magnesium oxide insulation. A second assembly was successful where open-end fabrication was carried out and an outgassing hole provided. Due to some moisture accumulation, the unit with its solid sheath cable was heated until the resistance stabilized at about 300 megohms; the outgas hole was then welded.

The configuration for the third test loop for the microwave in-core neutron flux monitor was analyzed. Calculations indicate that a circular waveguide using standard 3/8 inch O.D. tubing is feasible; the tubing was ordered.

NONDESTRUCTIVE TESTING RESEARCHElectromagnetic Testing

Development of an improved single-frequency eddy current tubing testing device was completed and a prototype was operated for field testing of stainless steel tubes. The instrument utilizes a graphical nulling device

UNCLASSIFIED

and the recording of both the in-phase and quadrature components of the test signals. The use of graphical nulling facilitates equipment adjustment and the two-channel recording provides more information than in conventional eddy current tests. A two-channel direct current output makes possible the use of a cathode ray oscilloscope as a vectorscope to aid in identifying and interpreting eddy current signal characteristics for different tubing discontinuities. The instrument was applied in the field with a developed internal test probe in the testing of 0.520 inch I.D. 0.054 inch wall, Type 304 stainless steel tubes. The test was complicated by the presence of bends in the tubing and carbon steel tube supports at intervals along the tube, and by the necessity of using an internal probe. Signals found in some tubes exceeded considerably those due to an artificial 0.013 inch electro-machined notch 1/8 inch long by .004 inch wide on the inside surface of a standard section of a tube. The noted signals were caused by areas of intergranular corrosion.

A solid state eddy current tester was designed, breadboarded, and fabricated for use in examining the tubes of steam generators in the tube sheet region. The test probe is mounted on a flexible shaft which protrudes from the front of the tester. Indication is provided by a meter and provision is made to connect a cable for a remote meter or a recorder. The tester operates on a frequency band from 1.25 mc to 1.75 mc. A solid state amplifier was fabricated for use with the tester. The amplifier input is balanced and remote from ground. The output circuit has a common ground.

A Dermatron eddy current thickness measuring instrument also was modified and used with an auxiliary amplifier in the testing of the tube sheet region of some steam generator tubes.

Zircaloy-2 Hydride Detection

Both long and short term stability of the 450 kc laboratory prototype eddy current hydride test instrument were improved through the use of a comparator circuit. A prototype temperature compensated probe for use at a frequency of 450 kc was fabricated. Modification of a small lathe for use as a scanning device is complete. The equipment is now ready for final adjustment and initial mapping of the resistivities of samples of N Reactor process tubes.

Stability tests on the new chopper-type comparator eddy current circuit show the one hour period drift in base level to be less than ten percent of the signal from a 2000 ppm (estimated) hydrided area in a Zircaloy-2 bench standard. This is a factor of 20 in improved stability over that obtained using ordinary oscillator-bridge-detector circuits, even with highly stabilized power supplies and a crystal controlled oscillator. Drift in the

ordinary circuits was primarily due to cathode emission changes when vacuum diode detectors were used and solid state diode detectors gave less stability than did vacuum diodes.

The base output level of the comparator circuit is almost independent of changes in oscillator level, filament and plate supply voltages, and changes in amplification of the a.c. amplifier. Experiments showed that excellent simultaneous compensation for the effect of probe lift off, probe tilt, and sample surface curvature has been achieved by the off-balance bridge method in the present instrument, which operates at 450 kc to give an indication of Zircaloy-2 resistivity changes at depths up to about 0.025 inch. Design of an eddy current circuit which operated on the comparator principle at 40 kc is partially complete.

Fundamental Ultrasonic Studies

Previous experiments indicated that a small transducer used with a lens system would provide the best plane, constant amplitude, ultrasonic waves. Several lens designs were tried using Lucite and aniline which have favorable acoustic properties and are readily available materials for solid and liquid lenses. In order to obtain plane waves from the spherical waves transmitted by the small transducers, single lens f-values in the range of $f/1$ to $f/3$ were necessary when either of these materials is used. Since spherical aberrations are notable in this range of f-values, a three lens system with an f-value greater than $f/8$ was designed. Special lens materials are being obtained for continuing experiments.

When continuous ultrasonic beams are incident on liquid-metal interface at the critical angle for Rayleigh waves, a lateral displacement occurs between the incident and reflected beams. In addition, the amplitude distribution in the reflected beam is altered in the vicinity of the Rayleigh angle. The beam displacement and the change in beam amplitude distribution are caused by the presence of inhomogeneous boundary waves which are similar to the more conventional Rayleigh waves. An invention disclosure has been filed on possible methods of utilizing these phenomena in nondestructive testing.

It has been reported in recent literature that detectable ultrasonic waves are generated by electromagnetic wave absorption of particle bombardment. These waves are believed to be the result of thermal stresses developing at the energy absorption site. Since the energy absorption can be confined to small regions of a metal surface, the resulting sources of ultrasound may prove to be valuable, high energy point sources for fundamental wave propagation studies. Also the detection and measurement of these ultrasonic waves may provide methods for studying the process by which the energy

itself is absorbed. For example, it may be possible to study thermal stress cycling by ultrasonically monitoring the thermal stress waves. It was also considered that the thermal stress waves from fission energy spikes might be detectable ultrasonically leading to new ways for detecting fission damage. An initial experiment was conducted using an aluminum-uranium alloy foil which was made to fission by both test reactor and neutron pulse irradiation. A 20 mc quartz transducer was bonded to the foil for the ultrasonic detector. Neither irradiation experiment produced detectable ultrasonic signals. Since the fission energy absorption is very fast, it may be necessary to use very high frequency, broad-band techniques in order to permit the detection of any ultrasonic waves that are generated.

A possible new method for controlling the transmitting and receiving properties of ultrasonic transducers was conceived and filed as an invention disclosure.

USAFEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Design and fabrication of the prototype 17 mil wall sheath tubing test continued. The electronic portion of the tester was fabricated and tests in conjunction with a dual beam crystal were started. Preliminary checks show the defect gate capable of following the tube surface as the tubing is rotated and crystal-to-tubing distance varies. With the tubing stationary, notch depth versus amplitude response data were obtained. The tubing was then rotated and translated past the crystal with the recorder plotting the amplitude response. Comparison of the data taken under stationary and rotating conditions showed the tester response to be identical under both conditions.

A plot of the notch depth versus amplitude response data showed the tester to possess a linear relationship for notches with depths of 5% to 50% of the 17 mil tubing wall. The electronic portion of the tester is operating satisfactorily, and upon completion of prints and parts lists, fabrication of the tubing tester will begin.

Crystal characteristic measurements are continuing. Permanent equipment for performing these measurements has been made operational. In addition to tests which measure transducer acoustic and electrical characteristics, techniques have been developed to ensure that the crystal is accurately aligned with respect to its housing. Tests have shown that as-received crystals are for the most part positioned in their housing with exceptionally close tolerance. This being true, it becomes possible to fix the

position of the crystal housing with respect to the test specimen, a step which appears to vastly simplify both fabrication and operation of the ultrasonic test station. Initial crystal measurements show this technique to be feasible and a test tank is being designed and fabricated on that basis. The tank is approximately a three-inch cube. Crystals are permanently mounted in two side walls while the remaining side walls house bearings which accept the tubing. The tank is fully enclosed with a Lucite top for easy viewing and ports are provided in both top and bottom for rapid tank filling or emptying. The tubing may then be inserted without plugs in the end to keep the inside of the tubing dry while testing. This tank should also be quite versatile in the ways it may be integrated into a test system.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

In a cooperative effort involving several components of the Hanford Laboratories working in the radiological sciences, the first field release of Iodine-131 occurred on July 25, 1963. Approximately 100 millicuries were released over the horizontal and vertical instrumented atmospheric diffusion grid starting at 0506 PDT and continuing uniformly for 40 minutes. The objectives were the following: 1) to relate the dispersion of Iodine-131 to the concurrent meteorological conditions to provide a verification of current prediction models, 2) to delineate the gaseous and particulate fractions of the tracer and determine changes that might occur with time of travel which would affect the form of the prediction model, 3) to determine the rate at which Iodine-131 deposits on natural pasture grass, cheat grass, and desert sage from which values of the deposition velocity can be determined, and 4) to compare analytical methods for determining the amount deposited.

The above objectives and those from other components form the basis for a broad study embracing the physical and biological behavior of Iodine-131 for short travel times. Analysis of the vegetation and air samples was in progress at month end. Early indications show that the dosages predicted for the experiment (HW-78312) were essentially correct and that the plume was contained within the experimental grid area. Significant counts were found at two miles' distance from the source.

The microscopic work portion of our precipitation scavenging research has now reached a point where the final results may be anticipated. The data consistently point to a peak scavenging efficiency for raindrops of about 0.4 mm diameter, and suggest a minimum scavenging efficiency for raindrops of 0.7-1.0 mm diameter. This is a deviation from currently used theoretical

calculations, which predict a rather flat peak in efficiency for drops larger than 1.0 mm.

The efficiency of collection increases with particle size also, as expected from theoretical considerations, but apparently at a much greater rate than current theory allows. Thus, our final data evaluation may show scavenging efficiencies considerably greater than 1.0 for values of $a^2 \rho > 50$, where a is the particle radius in microns, and ρ its density in grams per centimeter cubed.

Actual efficiencies await some further microscopic work and evaluation of the anisokinetic error of our air sampling system. Pending the completion of these, the following table is intended to give the presently indicated relative magnitudes of the scavenging efficiency of different sizes of rain-drops for various values of the area-density parameter, $a^2 \rho$.

Drop Diameter (mm)	$a^2 \rho$ (micron ² gram/cm ³)				
	12	25	50	100	200
0.4	0.5	0.9	1.7	3.0	5.2
0.6	0.3	0.5	1.2	2.1	4.3
0.8	0.4	0.5	1.0	1.6	3.6
1.0	0.4	0.6	1.0	1.6	2.8
1.2	0.4	0.6	1.0	1.7	2.3

Dosimetry

Whole body counting was started to determine radioactivity in Alaskan Eskimos. Counting was completed at Anaktuvuk Pass and Fort Yukon. The field party is now at Barrow. Comparison of Cs-137 body burdens at Anaktuvuk Pass in the cases of individuals who were also counted last year shows considerable individual variation but an average increase of about 25%.

Most of the whole body counting equipment was in Alaska, but it was possible to do a few experiments with the shadow shield. Some measurements were made with the slit system used in differential scanning and the system was calibrated for K-40. In both cases 3x3 inch NaI scintillators were used. This information was collected for the benefit of those, particularly medical users, who could use the shield to advantage but do not require such a large crystal as we do.

Fabrication was completed of the containers and most of the supports for the plutonium X-ray counter.

It was necessary to shut down the positive ion Van de Graaff twice during the month. Once was to replenish the deuterium gas supply. The other time was to replace oscillator tubes burned out by a high voltage surge. A transistorized nuclear counting system was received and is slowly being integrated into the activities of the laboratory.

Four BF_3 counter tubes that were ordered over a year ago were received. They are made of oxygen-free copper and were expected to show very little deterioration in counting characteristics due to gamma ray irradiation. Initial tests indicate that they behave as expected.

In spite of not being able to confirm the calibration of other laboratories for the spherical moderator-lithium iodide scintillation counter for low energy fast neutrons, the device was used at the PRTR. The indications were that much of the neutron dose was from low energy fast neutrons. The double moderator confirmed this by indicating an average energy for the neutron spectrum of about 0.3 MeV. Effort is continuing to calibrate the device and to improve the interpretation of the data obtained from it.

AEC Radiological Physics Fellows who are spending the summer with Radiological Physics began experiments to obtain knowledge needed for the design of the facility for the University of Washington spermatogenesis study. The first measurements were of the gamma rays accompanying the neutrons from reactions produced with the Van de Graaff and the increase in gamma radiation due to neutron capture in shielding materials.

Investigation continued of methods for increasing the sensitivity of radiation calorimetry. Results with pyroelectric materials have been disappointing so far; we do not seem to have reproduced phenomena reported in the literature. A simple gas thermometer was constructed. It had a sensitivity (measured as a capacitance change of a diaphragm) of 100 picofarads per degree centigrade, which appears promising for further development.

Instrumentation

The three individual detectors of the experimental solid state neutron dosimeter were tested with neutrons varying in energy from 12 KeV to 5 MeV. The measured responses of the moderated Li-6 and Rh-103 detectors were as predicted theoretically. Difficulties with the U-238 cover originally planned for fast neutron experiments necessitated the use of a Np-237 cover; however, due to pile-up caused by alpha activity, no valid fast neutron data were obtained. To circumvent this problem, five mil thick foils of 20-fold depleted uranium were obtained and attached to the detector cover just above the surface of the detector. Further tests will be carried out when the Van de Graaff becomes available.

Experiments for the telemetry of animal temperature, heart rate, and breathing rate information were carried out. Temperature variations were used to cause frequency modulation of the carrier and a pulse count-rate meter discriminator was used for detection. Satisfactory sensitivity was achieved with detectable limits imposed by oscillator stability. Amplitude modulation was used for heart beat rate and breathing rate experiments to permit continuous recording of both parameters. Work is progressing on a breadboard receiver to separate all variables for specific display.

A special glove box air balance monitor and alarm unit was completed for use at the Biology Animal Inhalation Laboratory. The device will serve to improve safety of radioactive aerosol experiments requiring a negative air pressure differential with respect to the outside atmosphere. The solid state manometer instrument provides an alarm signal if the air balance differential exceeds 0.1 inch of water equivalent.

An experimental working model of an instrument for monitoring thoracic air pressure in Biology animals was assembled and is being tested in the laboratory. Animal tests will be conducted by Biology personnel.

Design of a special valve was completed for use in studies of radionuclide deposition in animal lungs. The valve, which is now being fabricated, will control the path of air inhaled by the animal. Solid state control circuitry is being developed for automatic operation of the system. Additionally, modifications were carried out on an earlier model inhaled air volume monitor to improve operational characteristics.

A satisfactory location with exhaust hood was finally obtained for use in carrying out the filter-trapped beryllium measurements. The hood will be used to enclose the alpha source and complete detector head assembly. The Po-210 source is scheduled for delivery shortly.

Fabrication was completed on an experimental miniature solid state integrator to be used with portable neutron dose rate instruments. "Cordwood" type high packing density layout was used to conserve space. A total of 12 binaries, 2 amplifiers, a multivibrator, and an electromechanical register are incorporated. Initial testing appeared satisfactory.

Attempts are being continued to improve operation of the HAP0 Radiotelemetry System with the development of a new method of obtaining radiation level measurements. Developed circuits employ three linear decades ranges with provision for call-in information from all three. Satisfactory breadboard tests were completed and a final model is being fabricated. The specific three decades to be monitored will be determined depending upon requirements.

Necessary parts are being obtained to fabricate one production-type personnel, rechargeable, miniature, signaling dose meter with a single trip point at either 50 mr or 100 mr. This unit will serve as the model from which additional dose meters fabricated commercially may be patterned.

Experimental work continued on the scintillation CsI detector alpha energy analysis air filter monitor. Tests to date have been gratifying; however, the instrumentation is being continuously tested and recorded to secure specific radon-thoron background interference information during extreme atmospheric inversions. Although this approach, if it proves satisfactory, should result in a relatively inexpensive continuous monitor for airborne Pu-239, the previously developed scintillation coincidence-count continuous alpha air monitor has now operated satisfactorily in production use for about five months. Thus, both approaches to the problem seem to have considerable merit and specific application conditions will probably determine which type should be employed. Solid state surface barrier alpha detectors are also being investigated for use in alpha air filter monitors; however, the present relatively high detector cost makes this approach somewhat unattractive.

Experimental circuit development was continued regarding scintillation building radiation monitoring. The major emphasis involves using a 500 cps vibrating applied to the phototube in proportion to the radiation field intensity such that logarithmic response over a number of decades is achieved. Test chassis operation indicates that satisfactory performance can be obtained from 10°C to 55°C with five or six decades of coverage.

The magnetic core memory for the 400 channel analyzer was wired to the associated electronics. Minor modifications are being made to the memory drivers to provide pulse uniformity.

Assistance was given to Radiation Protection Operation in the programming of the Donner 3500 analog computer to simulate the decay of iodine and tellurium in the body and milk of sheep. The simulation was checked by programming the same problem on the GEDA computer.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses of program samples were provided at a normal rate during the month. Bids have been received on the fabrication of alumina parts for a vacuum-lock sample changer for the mass spectrometer. Testing has begun on the Ion-Test Bench of a scintillation-type ion detector which was designed for possible installation on the mass spectrometer. Several features

of assembly and operation which caused high counting-rate backgrounds and instabilities have been identified and corrected.

TEST REACTOR OPERATION

Operation of the PCTR continued routinely during the month. There were no unscheduled shutdowns. The k lattice experiments - k_{∞} KVNS fuel, dry were completed during the month. Groups of rats were irradiated in a predominantly fast neutron flux for varying irradiation times. The poison spline experiment for NRD was completed. Three samples of Hastelloy containing gadolinium were measured for gadolinium content. The aural monitor system, used during reactor fuel loading, was revised and made automatic. Operation of the Core Load Switch, which is necessary before entry to the reactor room, now also turns on the aural monitor.

The TTR was not operated during the month.

Negotiations were under way towards the excess Preliminary File Assembly machine and materials from KAPL.

CUSTOMER WORK

Weather Forecasting and Meteorological Services

Consultation continued on the meteorological and climatological aspects of oxides of nitrogen releases in the 300 Area. A weather watch for cross-arm changes activities continued for Electrical Distribution. Numerous requests for river data were processed.

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	93	87.3
24-Hour General	62	86.4
Special	182	92.0

The past month was the coolest July since the beginning of records in the Hanford Area in 1912. The average temperature was 72.4 as compared to 72.8 in the July months of 1915 and 1948, which previously were the coolest of record.

Precipitation during the past month totaled 0.31 inch, practically all of which occurred on the 7th. This was nearly twice the normal amount for July.

Lightning strikes on the evening of the 1st started 24 grass fires, which burned 4000 acres of cheat grass on the reservation. One of the fires was less than a mile from the meteorology tower.

Instrumentation and System Studies

A special soil moisture content measuring probe being developed for Advanced Technical Planning, CPD, and Chemical Effluent Technology, HL, has been partly assembled. A dual low-noise solid state preamplifier and pulse mixer was designed and tested. Five standard HAP0 aluminum wall BF₃ proportional counters will be employed as slow neutron detectors. The required Pu-Be neutron source was received.

Engineering assistance was rendered to Chemical Effluent Technology, HL, regarding the adjustment and employment of several inverters.

Detection probes for a plutonium waste container measurement system being developed for Plutonium Process Engineering, CPD, were tested. One NaI crystal was determined to have poor resolution, and it was returned to the manufacturer. A high voltage balance control was designed to drive the two probes and provide the same gain from each.

Improved mechanical features were incorporated into a special sample changer for large liquid radioactive samples for Analytical Laboratories, HL. The modifications will improve the ruggedness of the sample moving portion.

Assembly, testing, and delivery was achieved on two scintillation coincidence counters for air filter samples; one unit was for use in 325 Building by Radiation Protection, HL, and the other for use at the Fuels Recycle Pilot Plant.

Assistance was given Research and Engineering, IPD, in the selection of equipment for automatic readout of the sector probe. Meetings were held with two prospective suppliers of applicable equipment.

An accelerometer switching device was designed for Engineering Development, RAFL. The main problem was proper handling of low level vibration signals in an electrically noisy environment.

A feasibility study of a conceptual structural analysis machine has been completed and the results documented. The study was done under a work order from Metallurgy Operation. The machine (SAM) is a device to read metallog-

raphic photographs and make quantitative measurements electronically. In the system described in the memo, the photograph, or microscope sample, is scanned with a flying spot in a TV-like raster. The resultant light variations are detected with a photo tube. This video signal is processed in a suitable logic network which includes two serial memories. As illustrations of the machine's versatility, area fractions of a certain metal phase can be measured, as can the area distribution of the grains, the number of grains, or the angle between grain boundaries. The advantages are greater precision due to constancy of judgment and much greater speed. The greater speed means that many more sample areas may be scanned, thus improving the measurement statistics. The technique is useful to biologists, bacteriologists, or in fact anyone who must count or measure the gross features of objects in a visual field.

A component failure in the creep capsule instrumentation prototype micro-positioner control unit on-line at the in-reactor test facility at 105-KW caused the termination of three creep tests which were started in February 1963. The unit is being modified to incorporate high reliability components and circuit redundancy at critical points.

Design of a high temperature, microformer interrogation and readout system for the creep capsule tests is progressing. The core of the interrogation system is a magnetic circuit which employs the Hall effect to synchronously demodulate carrier frequencies to a d-c error signal. The system will be used in high temperature metals research.

Fabrication of the creep program 96 point data logger has been delayed due to the design modification of the micropositioner control unit.

Several combinations of circuits and scale factors for the C-column simulation were tested before it was determined that equation solutions were unstable. It was found, however, that the system can be represented to good approximation by neglecting organic backmixing diffusion term. This deletion has a further advantage in that the reduction in unknown terms simplifies optimization studies.

Considerable interest has been shown by IPD and NRD with respect to methods of measuring the operating characteristics of ultrasonic transducers. These tests help to insure uniform performance of production line nondestructive inspection equipment by reducing variations in the transducers used. Recently, seven internal bond test transducers for the I&E fuel elements were evaluated. Large variations were present in this lot. Although the results are not yet conclusive, transducers with the most uniform beam characteristics produced the most reproducible test results.

At the request of Plutonium Metal Development, HL, equipment is being designed and assembled for measuring the elastic properties of sample metal rods under highly transient conditions. Resonant frequencies induced in the rod as a result of thermal stresses which occur as the temperature of the sample is heated some 400°C in a few microseconds will be measured. The heating is to be accomplished by discharging a 10,000 joule capacitor bank through the sample. Present efforts directed toward design of the capacitor bank, switching methods and safety circuitry.

Optics

The Electrical Readout Traverse Mechanism was used to traverse two tubes in D Reactor with electromicrometer readout. No operational difficulties were experienced. The calculated distortions are believed to be accurate to ± 0.2 inch. Four runs were made on each tube and averaged.

Work has begun on the design of a borescope for use in inspecting VSR channels. The instrument will be designed to provide flexibility without loss of illumination or image, ease of disassembly and assembly for service purposes, wide angle objective lenses and uncemented relay lenses.

The use of optical projection methods and photocell readout to measure rail heights of fuel elements yields readings accurate to ± 0.0002 inch but the measurement is too slow for production line application. A circuit which trips a relay to indicate acceptable rail height has been tested. It is sufficiently fast and responds repeatably with an accuracy of ± 0.0002 inch.

A fuel inspection viewer has been designed and fabricated to assist in inspecting N fuel element spacer slips for cracks.

Capability for low power pulsing of laser rods has been achieved in the Optical Shop. Work is under way to obtain measurements of the threshold powers required to pump the two available rods at 77°K . These measurements will be compared to similar measurements made after the rods have been irradiated in a 3×10^6 R/hr gamma field. Experiments completed to date with the gamma field include relative brightness measurements of terphenyl scintillation material in various configurations and for different irradiation times. Brightness measurements were taken while the samples were in the field. Fluorescence from phosphors and phosphor doped glasses were also observed in this manner. Two terphenyl blocks irradiated for a total of 67.75 hours showed a noticeable decrease in output brightness after 45 minute irradiation and were not measurable at the end of the test. The blocks were darkened and attempts to bleach the darkening were not successful. The effect on output brightness of optical coupling between two terphenyl blocks was investigated by varying the spacing between the two blocks and by using

mineral oil as the interface medium. No appreciable increase in output brightness was noted as a result of increased coupling. The effect on output brightness of increased mass was investigated by increasing the number of terphenyl blocks in the field from one to six. The increased brightness was slight. One block, covered with reflecting silver tape, was measured and these measurements were comparable to those of six blocks.

Measurements on phosphors and phosphor doped glasses indicated that much greater brightness can be obtained with phosphor powder than with doped glasses. The most brilliant phosphor, P-20, seems to offer promise for pumping Nd^{3+} since the fluorescence should offer good pumping efficiency. One of the phosphor doped glasses obtained from Owens-Illinois Company exhibited a marked resistance to darkening in the gamma field. It was only slightly yellowed after 1.5 hours irradiation, while nine other samples were darkened after only five to ten minutes in the field. Should phosphor doping prove desirable, this glass would be a good host.

During the four week period (June 16-July 14) covered by this report, a total of 370 man-hours shop work was done. The work included:

1. Fabrication of ten glass bearings for CPD.
2. Fabrication of two rotatable camera backs for use on HL Metallographs.
3. Repair of one camera shutter.
4. Resurfacing of two pump seals.
5. Fabrication of laser rod grinding jig.
6. Fabrication of fuel inspection viewer.
7. Fabrication of two ultrasonic crystal holders.

Physical Testing

A continuing major effort has been applied to the nondestructive inspection of N Reactor steam generators on a continuous basis. To date, fluorescent penetrant has been applied on a statistical sampling basis to the welded tube ends in each of the ten steam generators, each of which contains some 4000 welded tube ends. In addition, eddy current tests have been applied to a number of the 5/8 inch O.D., 60 feet long, 304 stainless steel tubes in each steam generator. The tests successfully resolved conditions of stress corrosion cracking, and the eddy current tests have detected intergranular corrosion attacks in isolated regions along the tube lengths.

It is significant to note that these tests, applied on an emergency basis under difficult field conditions, have contributed substantially to an assessment of the magnitude of the problem. The eddy current effort, in particular, has been of major benefit as a result of new inspection

techniques which were successful in signal to background noise ratio improvement where the background noise arose from tubing kinks, bends, and dimensional variations.

Sufficient data were obtained to provide a confident estimate of each steam generator over-all metallurgical condition. It is expected that further testing will be performed on a normal work schedule. It appears likely that the eddy current test will be extended to other reactor components prior to startup.

Other field testing applications involved magnetic particle inspection of several crane hooks and associated equipment at the request of IPD. These tests successfully isolated defective hooks and thereby contributed to safe operation.

An ultrasonic thickness measurement was performed on Crib Waste Trailer No. 5472 in the 300 Area. Spot radiography was also conducted at various locations to detect possible corrosion; none was found.

ANALOG COMPUTER FACILITY OPERATION

Analog computer problems considered during the month were:

1. C-Column.
2. NPR Heat Exchanger and Control System.
3. Iodine and Tellurium in Milk.
4. D Reactor.
5. NPR Pressurizer.
6. PRCF Hazards.
7. Meteorology Study.

Computer utilization was as follows:

<u>GEDA</u>	<u>EASE</u>	
236	252	Hours Up
36	20	Hours Scheduled Downtime
48	48	Hours Unscheduled Downtime
<u>0</u>	<u>0</u>	Hours Idle
320	320	Hours Total

An analog technique was developed which causes the computer to automatically search for the optimum value of two system parameters. The technique, useful for optimizing system performance and for determining system parameters, may

have an application to the C-column study which is presently in progress. In this study it is required to find the values of certain parameters which produce a computed curve similar to a measured curve.

A digital program that allows the programming of the IBM-7090 computer in much the same way as an analog computer was secured from the Martin Company. The digital-analog simulator, which allows simulation of the basic building blocks of the analog computer such as integrators, differential relays, dividers, summers, etc., is being investigated as a possible means to aid in the scaling and programming of analog computers. It is believed that much time can be saved by running sample runs on a digital computer to indicate the range of outputs to be expected from various parts of the analog simulation.

Equipment consisting of a return signal analyzer and a power supply was received from Boonshaft and Fuchs, Inc. This second return signal analyzer is designed to slave with the return signal analyzer already present in the Model 711-A transfer function analyzer. The equipment, now being checked out, provides capacity for analysis of two signals simultaneously.

An unscheduled computer downtime occurred as a result of an air conditioner failure. A second air conditioner is now being installed in order to facilitate routine maintenance.

INSTRUMENT EVALUATION

The experimental gamma compensated scintillation beta-gamma hand and shoe counter is now under field test at 105-B. Performance to date has been satisfactory. The counting efficiencies, for standard HAP0 beta-gamma sources, were determined to be 7.5 percent for the hand probes and 4 percent for the shoe probes. The efficiencies for present HAP0 "Five-Fold" hand and shoe counters are about one percent.

Considerable testing progress was achieved regarding the seven scintillation solid state combined alpha-beta-gamma hand and shoe counters being fabricated by Instrument Laboratories, Inc. of Seattle. The units are completed and need only final inspection before shipment. As performance examples, the counting efficiencies were determined to be 12 percent for alpha detection by the hand probes and 4 percent for beta-gamma detection by the hand probes.

Evaluation testing continued on a Victoreen Model 440 beta-gamma dose rate meter portable instrument. Useful battery life was determined to be about 55 hours under continuous use for a calibration change of less than 5 percent.

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Calibration, evaluation, and final testing was completed on a scintillation logarithmic and linear response dose rate monitor fabricated for Analytical Laboratory, HL. The four linear ranges are 0-10 mr/hr up to 0-10 R/hr with essentially similar coverage for the logarithmic response range.

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CHEMICAL LABORATORY

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Film Adsorption Studies

Analyses were completed on the adsorption of five additional radioisotopes on aluminum coupons coated with various agents. These coupons were exposed for 30 days in reactor tubes downstream from the reactor zone. The average results from duplicate tests are tabulated below:

RADIONUCLIDE ADSORPTION ON TREATED COUPONS

<u>Coating</u>	<u>Radioactivity Relative to Non-Treated Coupon</u>				
	<u>Sc⁴⁶</u>	<u>Cr⁵¹</u>	<u>Zn⁶⁵</u>	<u>La¹⁴⁰</u>	<u>Sm¹⁵³</u>
Black Flo-Master Ink	.48	.84	.25	.60	.58
Black Marks-a-Lot Ink	.69	.72	.34	.78	.74
Silicone Resin 82	.58	.26	.32	.79	.72
Silicone Resin 112	.66	.28	.33	.80	.76

Of interest is the reduction of Zn⁶⁵ by factors of 3 and 4, and of Cr⁵¹ by a factor of 4 by the silicones. These coatings are the first to show substantial Zn⁶⁵ reductions. Significant reductions of each of these radioisotopes were observed which increases the interest in determining the effectiveness of these coatings under the conditions of the reactor flux zone.

Water Treatment Pilot Plant Studies

Since June 7 the Water Treatment Pilot Plant has been operating at a pH of 7.0 with aluminum nitrate as the flocculating agent, added at a rate designed to provide a zeta potential of zero. This flocculating agent was chosen to provide a test of the effect of reducing the sulfate ion concentration of process water. Excellent water quality has resulted with an average turbidity of 0.001 and filter runs of 8-12 hours. For all radionuclides except Cu⁶⁴ the effluent concentrations are lower in the tubes cooled with pilot plant water than in those cooled with process water. Although complete data have not been obtained as yet, P³² values are significantly lower than those obtained with aluminum sulfate as the flocculating agent. Another interesting observation on this test is that the radionuclide concentrations in the effluent from the aluminum tubes are lower than those from the zirconium tubes.

Deionized Water Coolant Studies

The regeneration operation of the deionizer was revised and resulted in higher purity water. Water with a specific resistivity of 3-4 megohms is currently being produced. Use of this water in two reactor tubes since June 7 has resulted in a reduction in concentration of Na^{24} in the effluent by a factor of 2 and of Ga^{72} by a factor of 5. These concentrations appear to be low enough and constant enough for salt addition studies to be started and investigation in detail of the origin of effluent radioisotopes of particular interest.

SEPARATIONS PROCESSESDisposal to Ground

The gross beta activity in ground water samples from wells monitoring the 216-A-10 Purex process condensate crib showed a slight increase during the current month; the maximum value reported was $4 \times 10^{-3} \mu\text{B/cc}$. Strontium-90 concentrations in May and June, 1963, samples from these wells were $< 7 \times 10^{-8} \mu\text{B/cc}$.

Gamma scintillation probings were made recently in wells adjacent to the 216-A-10 crib to determine the relative distribution of radionuclides in the ground water. The readings obtained indicated the highest concentrations of contaminants were at a depth of 10-15 feet below the water table. The results of depth samples obtained in wells at the north and south ends of the crib confirmed the earlier probe readings. In well 299-E-24-2 (north end) the total beta concentration at the water table was $4 \times 10^{-3} \mu\text{B/cc}$; at a depth of fifteen feet the concentration was $1.3 \times 10^{-2} \mu\text{B/cc}$. In well 299-E-17-1 (south end) a similar but less pronounced difference was noted. All of the water samples collected over the forty feet of depth are being analyzed for Sr-90. The distribution of radionuclide concentrations in the ground water under this site is probably due to the potential distribution pattern in the ground water zone, rather than to geological characteristics. Well logs and drilling samples indicate fairly uniform material (predominantly sands and gravels) to a depth of fifty feet below the water table.

Attempts were made to take core samples of the Redox 114-SX tank sludge by the rotary method. A special brass bit was used that was designed to collapse for core retention. No solid heel could be felt during drill rod placement, however, and drill operation revealed that the bit was resting on the tank bottom. Examination of the retrieved core barrel after the bit was collapsed showed only a small amount of solid material, thus supporting the belief that almost all of the solids had been dissolved and sluiced out of the tank.

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Plutonium Electrowinning

The appearance of a third phase just above the plutonium metal cathode in the latest electrowinning experiments prompted a brief study of the role of plutonium metal in the formation of this third phase and the effect of PuO_2 additions to an operating electrolysis cell.

It was observed that plutonium metal which has come into contact with the undesirable third phase material (not yet positively identified) develops a thin, metallic-appearing film which adheres very tenaciously to the metal. If this metal is subsequently used as a cathode, without prior removal of the film, it will initiate the formation of a new third phase, even though the melt itself has been rigorously purified. In attempts to remove the film from the metal mechanically, by treating with chlorine at 800 C in a LiCl-KCl flux, or by treatment with 8N HNO_3 , only the use of nitric acid was successful.

It was also observed that PuO_2 could be added in small quantity to the laboratory-scale electrowinning cell without detriment, provided the cell was operating at the time. It appears, then, from this and other information, that oxide impurity in the cell prior to start-up of the electrolysis may cause the formation of the undesirable third phase and the skin on the plutonium metal cathode, which in turn leads to the formation of additional third phase material as electrolysis progresses.

Solid State Electrolysis

Application (by J. R. Morrey) of absolute rate theory to interpretation of data acquired in study of electromigration of impurities in cerium yielded some rather anomalous findings, notably an apparent negative activation energy. Re-examination of the experimental data then revealed an unsuspected phenomenon - in the 242-hour electrolysis run with cerium, iron had migrated completely out of the specimen and into the tantalum clip which protects the nickel electrodes from contact with the cerium specimen. Applying corrections based upon measurements of the amount of iron which had diffused out of the cathode end of the cerium specimen, a self-consistent pattern of transport numbers and activation energies for electromigration of impurities in cerium, uranium, and plutonium is obtained.

Data on electromigration of iron in cerium, uranium, and plutonium yield values as follows for the transport numbers and activation energies.

<u>System</u>	<u>Transport Number</u> (g. atom. cm. sec ⁻¹ volt ⁻¹)	<u>Activation Energy</u> (Kcal/g. atom)
Cerium at 500-650 C	3.70×10^{-6} (Avg.)	4.07 (Avg.)
Uranium at 900 C	1.2×10^{-8}	20
Plutonium at 500 C	2.14×10^{-8}	14.9

Data on electromigration of nickel in cerium indicate a transport number about one percent of that for iron and an activation energy of 10.5 Kcal/g. atom.

Difficulties have been experienced with the spectrographic analysis of the samples from the plutonium run. A partial spill of some of the loaded electrodes occurred, and the samples will have to be re-run. Though not very satisfactory, the results reported on the first analysis indicate migration of nickel, chromium, silicon, and tin also occur in plutonium. The extent of migration of these elements is in about the same range as that of iron.

Carbon analyses have been completed on samples from the plutonium run. The results resemble those obtained with cerium rather than those obtained with uranium; no preferential migration appears to occur in plutonium. Both anode and cathode ends are higher in carbon than the center of the specimen. This may result from oxidation of carbon and evaporation of carbon dioxide as was postulated earlier for the cerium case.

The foregoing data show little promise for practical application of solid state electromigration in purification of either plutonium or uranium except possibly in small-scale, special purpose applications.

Use of Uranium(IV) in the Purex Partition Column

Two alternate procedures for preparing uranium(IV) nitrate solution from uranyl nitrate are being studied. A catalytic procedure (proposed by DeLeone, et al., Industrial Chemistry Laboratory, National Committee of Atomic Energy, Rome, Italy) involves contact of a uranyl nitrate-formic acid-urea solution with platinum coated aluminum pellets at room temperature. In the most promising operating conditions found to date, a 0.5M UNH-2.5M formic acid - 1.8M HNO_3 - 0.2M urea solution is passed through a bed of catalyst (1/8-inch alumina pellets containing 0.5 w/o platinum) at a rate of one liter per hour per 15 kilograms of catalyst. Under these conditions, 80 percent reduction of uranium was obtained.

In the second procedure examined, sodium sulfoxalate formaldehyde ($\text{NaHSO}_2 \cdot \text{CH}_2\text{O} \cdot 2\text{H}_2\text{O}$) is used as reducing agent. Complete reduction of uranium(VI) to uranium(IV) was obtained when equal volumes of 0.8M UNH - 3.0M HNO_3 and 0.9M sodium sulfoxalate were stirred together at 25 C. Uranium(IV) is reported to be stable for several months in solutions prepared in this manner. Possibilities of formaldehyde-nitric acid reactions in this system and in the use of the product solution in the Purex plant are recognized and being evaluated.

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Trilaurylamine Extraction of Plutonium and Neptunium from Purex Plant 1WW

Further single batch contacts of Purex plant 1WW solution (produced in March, 1963) with an equal volume of 0.3M TLA-Soltrol were made to determine effects of such variables as hydrazine concentration, reduction time and contact time. In all cases, greater than 95 percent of both neptunium and plutonium were extracted. Decontamination factors across the extraction step ranged from 1400 to 4800 for zirconium-niobium, 12 to 22 for ruthenium and 100 to 350 for cerium.

In a different type experiment a portion of 0.3M TLA-Soltrol was contacted alternately with fresh equal volume portions of plant 1WW (with hydrazine added to reduce neptunium) and fresh one-half volume portions of strip solution (0.1M oxalic acid). Six extraction-strip cycles were completed. Plutonium and neptunium losses were below five percent in all extraction contacts. Approximately 80 percent of the neptunium and plutonium present in the organic at the time was removed in each strip contact. Fission product extraction and strip behavior remained constant throughout the six cycles.

Ascorbic acid was tested as a reducing strip solution for removing plutonium from TLA-Soltrol solvent. Over 99 percent of the plutonium present in a 0.3M TLA-Soltrol solution was removed when the organic was contacted (30 - 60 minutes at 25 C) with an equal volume of ascorbic acid.

WASTE MANAGEMENT AND FISSION PRODUCT EXTRACTION

CSREX Process - Laboratory Studies

Tests to determine loss of BAMBP to alkaline solvent washes were initiated. When 0.475M BAMBP-Soltrol solution was contacted at 25 C with an equal volume of 1.0M NaOH previously equilibrated with Soltrol, the aqueous phase was about 0.01M in BAMBP (about two percent loss of BAMBP to the aqueous). Concentration of BAMBP in the aqueous was determined by extracting it into CCl_4 and measuring the absorbance at 279 m μ . The result given is the average of nine determinations. Similar measurements on the BAMBP-D2EHPA-Soltrol system are in progress.

D2EHPA Process - Pilot Plant Studies

The use of a separate scrub column, as recommended last month, gave considerably lower sodium and nickel concentrations in the product solvent. Sodium and nickel contamination in the product were reduced ten- and two-fold, respectively, over previous experience. An additional two-fold reduction in the nickel content was achieved by adding 0.005M EDTA to the scrub solution.

The concept of "total" extraction of dissolved alkaline waste sludge was briefly tested. The feed was prepared from a 1M HNO_3 solution containing initially 0.4M Fe^{+3} and 0.35M Al^{+3} (plus small quantities of other cations such as Cr, Mn, Ni, etc.). The solution was adjusted to a pH of 3.6 by adding 2.5M Na_2CO_3 , with acetate ion added to prevent iron and aluminum precipitation. A typical final solution contained 0.14M Fe^{+3} , 0.12M Al^{+3} , and 0.35M acetate. The feed was stable over a two-day period but began to precipitate as a gel on continued aging. In these runs, strontium and cerium losses in the 1A column ranged between 0.2 and 1.2 percent and 0.4 to 5 percent, respectively. These losses were obtained at volume velocities from 500 to 900 gph/ft^2 with excellent 1A column stability. No analyses of iron or aluminum in the organic extract were obtained, but qualitative evidence based on color indicated near quantitative extraction of the iron.

Attempts to strip the strontium from the solvent with 0.03M HNO_3 were only moderately successful due to inadequate pH control (no scrub column was used). In two runs, however, stripping losses of one percent or less were attained to produce pH's of 2.7 to 3.4. The strip flow was equal to the solvent flow in the runs; the volume velocity was 720 gph/ft^2 . Analyses of the strip product showed $<0.001 \text{ g Fe}^{+3}/1$ and $0.004 \text{ g Al}^{+3}/1$.

The solvent was washed in a "10" column with both 2M NaOH and 1M sodium tartrate tested. Both runs were made at 50 C with the organic phase continuous, and both washes were quite effective in removing the iron from the solvent. The NaOH produced a considerable amount of precipitate but without interference with column operation. No precipitate was observed with the tartrate wash.

In the course of the solvent washing studies, it was discovered that the solvent recipe (0.3M D2EHPA, 0.15M TBP) did not prevent "third phase" formation whenever the sodium concentration in the basic wash solution was less than 0.6 M. An unusual third phase situation was also discovered when an attempt was made to reacidify the washed solvent in a second column by contacting it countercurrently with 1M HNO_3 (organic phase continuous). After a few minutes' operation, a very milky dispersion was observed which eventually formed a "plug" through which the acid could not penetrate. At this point, a second organic phase was observed below the normal interface.

Cerium-Promethium Separation

The separation of cerium from the trivalent rare earths (promethium fraction) is crucial to the recovery of either cerium or promethium. Of immediate Hanford concern is the severe limitation which the intense heat-generating rate of cerium would place on the accumulation of a promethium inventory if the two were stored together without separation. Recent Hot Semiworks tests of the ORNL permanganate-D2EHPA process were largely unsuccessful and substitution of NaBiO_3 or PbO_2 for KMnO_4 was also not fully

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satisfactory. Search has accordingly been initiated for a more efficient process, either solvent extraction or precipitation, for accomplishing separation of cerium from the other rare earths. Promising preliminary results have been obtained with a scheme which involves extracting all the rare earths (including cerium) into an organic phase followed by stripping under oxidizing conditions, such that the trivalents are removed while cerium remains in the organic (for later reduction and removal). Advantages include 1) a great reduction in radiation damage to the oxidant and 2) a much purer and more concentrated promethium fraction, better suited for subsequent ion-exchange purification. Laboratory experiments to date have featured use of D2EHPA as extractant and silver-catalyzed persulfate as oxidant; however, a variety of other extractants and oxidants will be evaluated for this use. A quantity (70 gallons) of "carbonate product" - a crude strontium, rare earth concentrate - has been received from Purex for use in hot-cell testing of promising processes.

Cesium Removal from Wastes

Since kinetics are limiting in the removal of cesium from Redox alkaline supernatant wastes with Duolite C-3 resin, pressure drop measurements were made to determine the practicality of using finer mesh resins. Pressure drop data were obtained for 20 to 40 mesh, 40 to 60 mesh, and 20 to 60 mesh resin for water and synthetic "SX" farm waste at flow rates up to 15 column volumes per hour. Results indicated that 40 to 60 mesh resin with SX waste had a prohibitively high pressure drop, but all other combinations were in a favorable range. For example, the 20 to 60 mesh resin with SX waste at 2 column volumes per hour flow rate had a pressure drop of 0.6 foot of water per foot of bed. Cesium breakthrough data indicated similar kinetics for 20 to 60, 40 to 60, and the previously used 20 to 50 mesh resin while the 20 to 40 mesh material had considerably poorer kinetics. Thus, the 20 to 50 and 20 to 60 mesh resins are most acceptable from both kinetic and pressure drop standpoints.

Cesium removal from Purex aqueous sludge leach solution by ion exchange was investigated. The synthetic waste used was equivalent to five volumes of waste per volume of sludge and contained about 3M sodium and 5 percent as much cesium as the tank supernatant solution. The cesium breakthrough volumes for Linde AW-400 zeolite were very favorable, 300 column volumes to 50 percent breakthrough and 120 column volumes to 1 percent at a flow rate of 4 column volumes per hour.

Storage of Strontium on Zeolites

Further observations and experiments with the B-Cell strontium-loaded zeolite cartridge has revealed 1) that the gas or water vapor absorption-desorption reaction with the zeolite is fast compared to the rate of recombination of radiolytic gas and 2) that the B-Cell unit now has a

small leak. Results of the zeolite study to date and proposed future studies were summarized in an informal memo.

Strontium-Lead Sulfate Carrier Precipitation

Pilot plant testing was initiated on recently recommended improvement for carrying out the lead sulfate carrier strontium precipitation process currently used in the Purex plant. Techniques are being sought to adapt this process for strontium removal from acidified waste sludges. The procedure involves adding sulfate ion to $1M$, neutralizing with sodium carbonate to pH 1, digesting one hour at 80 C, adding lead nitrate at this time to a concentration $0.02M$ and centrifuging.

The procedure was tested in the pilot plant 26-inch solid-bowl centrifuge operated at 1800 rpm and developing an acceleration of 1200 G's. The bowl residence time was set at five minutes. Under these conditions, the strontium loss to the supernate in two duplicate runs was three percent. A third run was made with $0.43M$ hydroxyacetic acid added (as in the current Purex flowsheet) and with the lead nitrate added slowly over a 15-minute period while agitating (in the earlier runs, the lead nitrate was added essentially instantaneously with the tank agitator off). The loss in this run was 1.5 percent. The feed, approximating that which may, in part, be expected with acidified stored alkaline sludge, for all of these runs (after final volume adjustment) contained $0.05M$ Fe^{+3} , $0.14M$ Al^{+3} , 0.002 to $0.005M$ Ce^{+3} (as a stand-in for the rare earths), and 0.0004 to $0.001M$ Sr^{+2} . No difficulty with foaming was encountered during pH adjustment with carbonate, nor was there any sign of iron precipitation.

The centrifuged precipitate had the consistency and appearance of firm toothpaste and was very difficult to slurry in the centrifuge bowl. Attempts to metathesize the cake with carbonate in the bowl by adding $2.5M$ Na_2CO_3 at a rate of 2.5 moles/mole lead were not successful, either at 25 or 60 C. The lead sulfate precipitate was readily soluble in $0.6M$ Na_4 EDTA added at 20 percent stoichiometric excess and the resulting solution was stable at pH 4.

Except for the difficulties with cake slurrying and dissolution, these runs provided no clue for the cause of current Purex strontium losses of about 20 percent.

Strontium Extraction from Dissolved Sludge

A precipitation approach was reported in the May monthly progress report for the treatment of dissolved waste-tank sludges to reduce their iron content prior to processing in the B-Plant Waste Management facility. A separation of strontium and iron is desirable in order to reduce the volume of feed to the B-Plant solvent extraction battery and eliminate or reduce the quantity of expensive complexant required to maintain the iron

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in solution during solvent extraction. Treatment of acidified sludge waste with sodium carbonate, to about pH 1, followed by carrier precipitation of strontium with lead sulfate continues to show much promise. Numerous laboratory experiments have been performed to investigate the effect of pertinent variables and to better define optimum conditions and a successful liter-scale hot cell experiment has also been completed. In the latter, two-year-old LWV was adjusted with caustic and nitric acid to acidified sludge composition, $\text{Pb}(\text{NO}_3)_2$ equivalent to 0.02M added, and digested at 80 - 90 °C for one hour. The precipitate was compact, crystalline, and fast settling, and carried 96.4 percent of the strontium. Other experiments showed that the digestion period (at > 80 °C) should be at least 15 minutes when rare earths are also present and that final solution volume, starting with LWV, should be not less than twice the initial volume in order to insure freedom from iron precipitation.

In associated experiments, in support of the Purex Head-End strontium production effort, a pH 1-1 M Na_2SO_4 solution was found to be the preferred wash for removing interstitial solution from the strontium-product centrifuge cake. Washing at 80 °C was slightly more effective than at 25 °C.

Quality of Strontium-90 in Purex Waste Tanks

Isotopic analyses of the strontium in the Purex waste tanks and analyses for "cold" strontium in the various Purex essential materials were recently completed and appear to have resolved the mystery of why some tanks contain "good" strontium (close to theoretical Sr^{90} content) and some "bad" (diluted with natural strontium). Explanation is the level of natural strontium impurity in caustic from the two vendors who supply most of Hanford's needs. One caustic runs typically 200-400 ppm (100% NaOH basis) versus 12 ppm for the other. With one exception (Tank 102A), calculations based on filling schedules and purchase records yield tank isotopic compositions in excellent agreement with mass spectroscopic analysis. Tank 102A will be re-analyzed. It is fortunate most of the caustic requirements since the start of the strontium program have been supplied from the low-strontium source.

Treatment of Purex Plant Stored Waste Sludge

Pertinent to possible neutralization of the carbonate present in Purex plant stored waste sludge, the evolution of nitrogen dioxide from nitrite-nitrate-carbonate solutions as a function of pH and temperature was determined. Nitrite-nitrate-carbonate solutions were brought to a selected pH with nitric acid and, while at a selected temperature, were sparged with argon gas. The sparge gas was scrubbed with permanganate solution and the O_2 present was determined by a titration procedure. Consistent values were obtained at temperatures of 25 and 57 °C in the pH range 3 to 5.5. At higher pH values, the NO_2 in the vapor phase was too low for accurate

measurement. At 87 C values obtained were less consistent due to changes in pH during sparging. As expected, NO₂ evolution increased with increasing temperature and decreasing pH.

Thermal Conductivity of Soils

Advantage was taken of the steep moisture profile of the model crib near 200 North Area in field studies to gain more data on the relationship between soil moisture and thermal conductivity. The latter is important in determining allowable heat generation rates in bulk solidified salt cakes. The line heat source thermal conductivity probe was used in conjunction with a neutron moisture probe to obtain corresponding in-situ measurement of moisture and conductivity to depths of 10 feet in the model crib.

The test results indicate that thermal conductivity of the soil increases rapidly (0.07 to 0.2 Btu/hr ft² °F ft) with moisture content over a range of from 5.4 to 10 volume percent. From 10 percent to 30 percent moisture the thermal conductivity increases more slowly from 0.2 to 0.35 Btu/hr ft² °F ft.

Waste Tank Sample Pump

In support of the Waste Management program improved methods of waste tank sampling are being sought. The use of weighted sample bottles is time-consuming and results in high personnel exposure rates. An alternate sampling method has been devised using an air-operated, deep well pump to permit sampling from selected tank levels and to deliver the samples to an enclosed chamber at ground level. From this chamber laboratory size samples may be remotely withdrawn by means of a small jet or by gravity. The entire assembly can be remotely installed through existing four-inch pipe nozzles. The sampler is adaptable to either continuous or intermittent sampling.

EQUIPMENT AND MATERIALS

Non-Metallic Materials

Samples of an acetal polymer (Delrin), a polycarbonate (Lexan) and a styrene-acrylonitrile copolymer were irradiated (gamma) and electrical resistivity measurements were made to evaluate radiation induced changes. The styrene-acrylonitrile polymer had the greatest radiation tolerance. No significant change in either surface or volume resistivity occurred during an exposure to 10⁹ R. The sample also had remarkably good mechanical strength after 10⁹ R exposure. The polycarbonate was too brittle to test after 10⁹ R and the acetal polymer could not be tested after 10⁸ R. Tests of the chemical compatibility of these three materials with several typical plant solutions are in progress.

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Scaling of Titanium and Stainless Steel in Concentrators

Scaling studies involving exposure of titanium and 304-L stainless steel bayonet heat exchangers in simulated Purex concentrated waste (1WW, F-6) and waste backcycle (3WB, H-4) streams were continued. A scale about 1/16-inch thick has formed on a titanium bayonet after 14 days' exposure to simulated 3WB solution which was periodically butted with colloidal silica. No appreciable loss in heat transfer has been noted to date; the internal temperature of the bayonet has remained nearly constant (155-158 C). In contrast to a scale formed when the bayonet was exposed in a 10M HNO_3 - SiO_2 medium (no iron or uranium present) the scale formed in simulated 3WB solution appears quite adherent and has shown, to date, no spalling tendency. No scale has formed on a 304-L stainless steel bayonet exposed to simulated 3WB solution presumably because of corrosion undercutting.

No visible scale has formed on a titanium bayonet boiling simulated 1WW solution for ten days. However, the internal temperature of the bayonet has increased linearly with time from 150 to 187 C, perhaps due to a decrease in the number of boiling nucleating sites on the surface with consequent change from nucleate to film boiling. Boiling of the solution has nearly stopped after ten days' exposure.

Other work* indicates that the heavy scale deposit present on the titanium tube bundle recently removed from H-4(3WB) service is largely silica. Among several reagents tested by McKenzie, only 40 percent HF completely dissolved samples of this scale. In attempts to define practical descaling reagents for use on the bundle, the corrosion of A-55 Ti by solutions containing 1 - 5 M HF and various additives designed to inhibit corrosive attack was studied. Only one of the combinations tried, 1M HF-1M- CrO_3 , corroded A-55 Ti at a rate which might not be considered prohibitive. These studies are continuing.

A visual examination of a titanium tube bundle from the Purex F-6 (1WW) concentrator was made. The condensate return line was visible with the crane optical system but the ends of the tube bundle were not. Heavy scale deposits, such as those in the H-4 bundle, were not apparent although a thin, grey scale appears to be present on portions of the bundle submerged during normal operation. Several pictures, in color, were taken with the remote camera. These are not yet available for study.

Effects of Welding Parameters on Corrosion of 304-L Stainless Steel Weldments

A systematic study is in progress to determine if variations in welding parameters (interpass temperature, stringer vs. weave deposit, number of

*By T. R. McKenzie, Separations Chemistry Laboratory,
Chemical Processing Department

passes, heat input, etc.) have a significant effect on corrosion resistance of weld metal in welded 304-L stainless steel. Two methods of welding, tungsten inert gas (TIG) and shielded metal arc (SMA), are included in the study. All welds have been made using material from a single sheet of base metal whose chemistry and history are well known. Samples of all welds will be corrosion tested in 1M HCl, 65 w/o HNO₃, and simulated Purex LWV solution. All TIG-welded samples have been corrosion tested at this time. There do not appear to be any significant differences among the different weldments in their corrosion behavior in a given reagent. All SMA-welded samples have been tested in 1M HCl. No significant differences in corrosion rate or mode of attack were found among the SMA-welded samples. There is a significant difference between SMA-welded and TIG-welded samples in 1M HCl which will be explored further. Testing of SMA-welded samples in 65 w/o HNO₃ and simulated LWV has not been completed.

PROCESS CONTROL DEVELOPMENT

Air Pulser Control System

A prototype of the air pulser control system for the new Plutonium Reclamation Facility has been obtained for testing and evaluation of operating characteristics prior to plant startup.

Development of a simpler, less expensive alternate for pulser control continued with the design and testing of a centrifugal flowmeter as a key component. This device operates on the principle that the pressure differential between the inner and outer peripheries of a flow loop is related to the velocity through the loop, and is independent of flow direction. The meter is thus the mechanical analog of an AC to DC converter, and provides a signal directly related to the pulse frequency-amplitude product. On the basis of the test results, specific designs for use with three- and four-inch diameter columns have been determined. This system represents the lowest investment in equipment of several systems so far investigated.

Advanced Process Control Development

A paper has been drafted describing the derivation of the equations representing the mathematical model of the C-column. Fitting of experimental data to these model equations is proceeding with the help of analog computer runs to obtain the range of values of the equation parameters. Initial difficulties with instabilities in the analog solution have been overcome by mathematical manipulations; and approximation to K (mass transfer coefficient), B, and D (aqueous and organic backmixing coefficients) are being determined using a direct search method. The next step is to determine the best values of K, B, D for each run using a digital solution

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to the two point non-linear boundary value problem. This problem has not analytic solution but can be solved through a stepwise linearization and iterative process. The programming for this step is now complete and debugging is in progress.

The program which simulates the dynamic characteristics of the C-column has been completed. This program will be used to test adaptive and optimal control schemes in conjunction with process control computer systems. De-bugging is proceeding using test cases which have appeared in the literature.

REACTOR DEVELOPMENT - O4 PROGRAMPLUTONIUM RECYCLE PROGRAMSalt Cycle Process

Flowsheet Studies - Laboratory studies of the recovery of plutonium from molten chloride salt solutions by electro-codeposition of $\text{UO}_2\text{-PuO}_2$ solid solutions have continued. In attempts to reach the current process goal of removing over 99 percent of the plutonium with not more than 5 percent of the uranium, the following has been accomplished:

1. Up to 94 percent of the plutonium has been codeposited with about 10 percent of the uranium from a LiCl-KCl melt initially containing 20 weight percent uranium and 0.2 weight percent plutonium. About 20 percent of this deposit broke loose from the cathode during the electrolysis.
2. Starting with a melt containing only 0.02 weight percent plutonium and codepositing under similar conditions, more than 90 percent of the plutonium was removed, along with 15 percent of the uranium. Thus, the final plutonium concentrations in the melt were only 0.002 weight percent, indicating that plutonium removal by codeposition techniques does not have a "cut off" plutonium concentration. Further evidence in support of this conclusion came from another experiment in which 99.8 percent of the plutonium was removed, along with 72 percent of the uranium.

Chloride Contamination in Electrolytic UO_2 - The following results have been achieved in the program to investigate the causes and cures for chloride contamination in electrolytic UO_2 :

1. A technique for chloride determination, using an amperometric titration with coulometrically-generated silver ion, was worked out and placed in service in the Analytical Laboratories.
2. Analysis of samples of UO_2 prepared in Chemical Development pilot plant equipment showed reduction in chloride content from about 1000 ppm to 180-200 ppm upon washing in distilled water. Treatment of two washed samples by pyrohydrolysis techniques (exposure to an atmosphere containing 5 percent H_2 , 4 percent H_2O , 91 percent N_2 at 1100 C for six hours) reduced the chloride content to 98 and 55 ppm.
3. Analysis of washed samples of UO_2 produced in a cell using a pyrolytic graphite anode gave chloride contents of about 50 ppm.

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Pyrohydrolytic treatment of this material, under the conditions cited above, had little effect upon chloride content. These results offer support to the idea that occluded micro carbon particles (from the anode) act as chloride contamination sites in electrodeposited UO_2 .

Electrochemistry of Uranium in Molten Chloride Salt Solutions - EMF data and the thermodynamic properties derived therefrom for the chlorination of UO_2 in equimolar KCl-LiCl at 570 C are summarized below:

<u>Conc.,</u> <u>Molal</u>	<u>E,</u> <u>volts</u>	<u>$\Delta G,$</u> <u>Kcal/mole</u>	<u>$\Delta S,$</u> <u>Kcal/degree-mole</u>	<u>$\Delta H,$</u> <u>Kcal/mole</u>
0.034	0.697	-32.1	- 8.3	-40.4
0.108	0.656	-30.2	- 9.7	-39.9
0.394	0.605	-27.9	-12.5	-40.4
1.130	0.549	-25.3	-15.5	-40.8

Precipitation Studies - Six cerium (stand-in for plutonium) precipitation runs were made in a 20-liter salt bath containing 2.5:1 LiCl-KCl molten salt. Various recirculation devices which were operated with a $\text{Cl}_2\text{-O}_2$ sparge and designed to collect the CeO_2 on quartz surfaces were employed in the runs. The design that gave the best recovery (65 percent) was simply a bed of quartz chips through which the molten salt was circulated. An overall analysis of the six runs showed that 248 g of cerium were added, 106 g cerium were recovered, and 32 g cerium remained in solution. Cerium losses probably resulted from plating of the cerium on the quartz pot above the surface of the molten salt solution. A large quantity of black material (CeO_2 with UO_2) was visible at this location. This problem will probably not be encountered if work is done in the 1:1 KCl-LiCl system.

Hypochlorite Destruction Tests - The hypochlorite formed during scrubbing of chlorine with a 15 percent NaOH solution must be destroyed in order to permit temporary storage in available stainless steel tanks of wastes that will be generated during the projected full level demonstration. Methods represented by the following equations have been suggested.

1. $2\text{NH}_3 + 3\text{NaOCl} \rightarrow 3\text{NaCl} + 3\text{H}_2\text{O} + \text{N}_2$
2. $(\text{NH}_2)_2\text{CO} + 3\text{NaOCl} + 2\text{NaOH} \rightarrow 3\text{NaCl} + \text{Na}_2\text{CO}_3 + 3\text{H}_2\text{O} + \text{N}_2$
3. $\text{H}_2\text{O}_2 + \text{NaOCl} \rightarrow \text{NaCl} + \text{H}_2\text{O} + \text{O}_2$

The hydrolysis of urea in the NaOH solution resulted in the ultimate formation of a sub-micron mist of NH_4Cl which plugged the tower off-gas lines leading to the abandonment of method 2. Complete destruction of the hypochlorite was achieved without adverse side effects by adding a 10 percent

excess of 15 percent hydrogen peroxide at hourly intervals. The evolution of oxygen was not a serious problem.

Chlorine Liquefaction Studies - A 50 percent chlorine - 50 percent air mixture was passed at four liters per minute through a modified 15 lb. capacity chlorine cylinder which was immersed in a liquid nitrogen-trichloromono-fluoromethane bath (-84 C to -70 C) resulting in the liquefaction of 95 percent of the chlorine as determined by analysis of downstream caustic scrub solutions. The run was terminated after two hours because of difficulty in keeping the bath cooled to the desired temperature of -100 C, which was actually never reached.

Container Materials - A graphite crucible which had been coated on the inside with pyrolytic graphite was tested as a possible container for fused salts. Molten equimolar LiCl-KCl penetrated the lower portion of the vessel (nitrogen blanketed) in less than 24 hours. The coating delaminated and the substrate graphite deteriorated near the top.

RADIOACTIVE RESIDUE PROCESSING

Full Level Calcination Studies

Preliminary observations on an A-Cell spray calciner run (#17) with ammonia-neutralized Purex waste were reported last month. Analytical results now available verify that the extent of ruthenium volatilization was intermediate between the very low value (<0.1 percent) with caustic and that with acidic waste (about 20 percent). The cumulative fraction of the feed ruthenium evolved during the run and collected in the condensate was 4 percent, whereas that of the other fission products (Ce-144, Cs-137, and Zr-Nb-95) was about 0.1 percent, i.e., decontamination factors of 1000. Overall decontamination factors through the off-gas train were in the normal range of $>10^8$.

Since the caustic run involved the use of sugar, as well as the alkaline conditions, a second ammonia-neutralized run (#18) was made identical to #17 but with 30 g of sugar added per liter of waste. A higher concentration of sugar was not used because runs in the "cold" spray calciner indicated that a non-meltable powder would result. With 30 g sugar/l and ammonia, 1 percent of the ruthenium was evolved, with instantaneous samples ranging from 0.6 to 4.0 percent. It thus appears that the presence of sugar decreased the loss of ruthenium somewhat, but not to as low a value as in the caustic run. Only 3 percent of the ammonia added to the feed was found in the condensate. Hence, very little of the ammonia could be recycled, one of the previously assumed advantages of ammonia. Ammonia's only apparent advantage over caustic would appear to be in the somewhat smaller volume of melt or calcine produced; however, even this advantage would be small with FTW (formaldehyde treated waste) or STW (sugar treated waste) as contrasted to LW.

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Eighteen-Inch Radiant-Heat Spray Calciner

Extensive runs were made to determine the capacity of the Cold Semiworks reactor with the new draft-tube arrangement. The adjustable tube and bottom cone were extended to their maximum lengths of 105 inches and 131 inches, respectively, for all runs. Capacity for water evaporation, defined as the maximum rate at which no water droplets reached a centrally located thermocouple 4-1/4 feet below the nozzle, range from 17 gph at 650 C wall temperature and 20 psig atomization pressure to 36 gph at 800 C wall temperature and 60 psig atomization pressure. The reactor capacity at a constant furnace temperature increased approximately linearly with the square root of the atomization pressure (absolute) over the range of 20 to 60 psig.

Reactor capacity with an internal-mix nozzle (Spraying System Co. No. 72) operating at a given atomization pressure was from five to eight percent higher than with an external-mix nozzle of comparable size (Spraying Systems Co. No. 70); and atomization gas requirements were generally about 30 percent less. However, feed line pressure is higher with the internal-mix nozzle (40 - 80 percent of atomization gas pressure versus zero line pressure).

At constant atomization quality the reactor capacity increased with approximately the third power of the absolute temperature of the furnace. A crude analysis of the heat flux as a function of the reactor wall temperature and the mean gas temperature in the draft tube annulus indicated that about 50 percent of the heat flux was due to thermal radiation at a wall temperature of 760 C.

Heat transfer to the evaporating spray in the center section of the reactor is essentially completely by conduction-convection from the hot gas recyded to the top of the reactor through the annulus.

Waste Solidification Engineering Prototype

Approximately one year ago, the Hanford Laboratories were authorized by the Division of Reactor Development, USAEC, to initiate design, development, procurement and fabrication of a Waste Solidification Engineering Prototype. The prototype is to be installed in a new facility, the Fuels Recycle Pilot Plant, now under construction. The prototype is to be a flexible array of equipment capable, with modest additions and alterations, of providing engineering scale demonstration of promising solidification processes with actual high-level wastes.

Design and fabrication activities to date have been principally concerned with the auxiliaries (feed system, off-gas treatment, viewing and manipulation aids) required. Current efforts are on the engineering

scoping and detailed design of the solidification machines. Three solidification devices, pot calciner, radiant-heat spray calciner with or without continuous melting, and a continuous phosphate glass device will be provided. Cold development of the first and last processes is being conducted at the Oak Ridge National Laboratory and Brookhaven National Laboratory, respectively.

Current schedules are targeted and indicate that fabrication of the glass prototype parts and assemblages with the possible exception of the glass machine will be complete at the same time (June 1965) construction of the Fuels Recycle Pilot Plant is scheduled for completion.

Calciner Pot Level Measurement

Following completion of powder and melt level instrumentation studies using powder calcined without phosphate addition, further experimentation was initiated with phosphate-containing material. A five-inch-diameter induction-heated pot was used to melt the pre-heated powder. During an experimental run to study the performance of melt level detection instrumentation, the pot erupted, spewing its contents inside the hood. The eruption is attributed to the sudden release of pressure built up under a crust of solidified melt. The crust apparently formed over a previously melted partial charge in the pot. New experimental apparatus and revised operating procedures are being prepared to enable continuation of this work.

Filter Blowback Study

Instrumentation has been assembled and calibrated for measuring gas pressures at selected points along a porous stainless steel filter during blowback. The instrumentation consists of a bridge balance unit, a power supply, and a memory-type oscilloscope which will read out the dynamic pressures of up to ± 14 inches of water generated during the pulsed application of air pressure to the filter during the blowback cycle.

Electrostatic Bubble Scrubber

A mock-up was assembled which essentially duplicates the important dimensions and geometry of the electrostatic bubble scrubber used in high level waste calcining studies. The deposition of 0.2μ particles in the various stages of the scrubber was measured at a flow of one cfm and for two operating voltages. In Table I are shown the stage efficiencies for the conditions employed. The stage efficiency is the ratio of the retained weight of particles to the weight entering the stage.

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TABLE I

Ionizer Voltage	-7700 V	-7200 V
2nd Stage Voltage	-2600 V	-2400 V
Corona Current	60 μ a	36 μ a
Stage Efficiency		
Ionizer	91%	77%
2nd Stage	49%	40%
(passage to scrubber)		
Scrubber	47%	42%
Total Percent	97.5%	92%

The voltages and currents used are typical of the presently used operating conditions of the calciner electrostatic bubble scrubber.

For the 0.006-inch wire used for the corona discharge a large fraction of particles was deposited in the charging section.

There are some differences in the mock-up and the actual scrubber but these are believed to have a minor influence on the overall results.

Thermodynamic Properties of Zeolites

Rational thermodynamic equilibrium constants and standard Gibb's free-energies were determined for several synthetic and natural zeolites with the exchange reaction, $\text{Na}_{\text{zeolite}} + \text{Sr}_{\text{solution}} \rightarrow \text{Sr}_{\text{zeolite}} + \text{Na}_{\text{solution}}$.

Rational equilibrium constants are representative of an average cation selectively over the whole equilibrium curve, and for that reason are more useful for comparative purposes than uncorrected mass action quotients. Only the selectivity at a given point on the equilibrium curve is reflected in the mass action quotient. The larger the equilibrium constant, the greater the zeolite selectivity, in this case, for strontium. Equilibrium constant relationships in several cation systems also allow determination of the accuracy of the equilibrium data.

Rational thermodynamic equilibrium constants for Linde 4AXW, 13X, AW-400, AW-500, Norton Zeolon and Hector clinoptilolite were 42.3, 15.7, 0.339, 0.295, 0.220 and 1.09, respectively. Associated standard Gibb's free-energies for the reaction, expressed to the nearest 100 cal/mole, were -2200, +600, +700, +900, and -100 cal/mole for 4AXW, AW-400, AW-500, Zeolon and Hector clinoptilolite, respectively. In general, there is an inverse relationship between SiO_2 to Al_2O_3 ratio in zeolites and zeolite strontium selectivity.

Ion Exchange Properties of Ferrocyanides

Ferrocyanides of cobalt, nickel, copper, zinc, cadmium, tin, lead and cerium were prepared and shown to have ion exchange properties. The exact composition of the resulting salts was not determined, but the compounds can be expressed in the form, $(K_2, Co) Fe (CN)_6$. Distribution coefficients (K_d) were determined for each compound and favorable values were obtained on the nickel, copper, cadmium, lead, and cerium salts. The copper and nickel salts were stable in acid solution and were further evaluated in column experiments. These exchangers were loaded with cesium from a synthetic 1965 FTW (85 gal/ton) solution for comparison with other exchangers.

The 50 percent cesium breakthrough points were at 490 and 295 column volumes for the copper and nickel salts at a flow rate of 7.6 column volumes per hour. This is equivalent to 0.66 to 0.35 meq. Cs-137 per gram of exchanger.

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BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

A method was obtained which holds promise of enabling ground water flow paths to be found from the potential distribution. This is an important step in determining the permeability distribution for use in the electrical ground-water analog, since the permeability integral (described in the May Monthly Report) must be evaluated along a flow path. The flow path method, which is easily adaptable for computer use, numerically steps along paths of steepest descent with respect to the potential function. Such a path is specified by simultaneously satisfying the two ordinary subsidiary differential equations defining a streamline. Results from the method are currently being compared with closed-form solutions for specific stream functions to learn the accuracy which can be obtained.

The extent and character of the Prosser interbed and the overlying Saddle Mountain No. 3 basalt flow were further clarified. That basalt flow, which forms the surface of the Columbia River basalt series beneath much of the Hanford Works area, is not continuous beneath all the area. It either eroded from the part now beneath the Separations plants areas or was never laid down there. This fact and the high permeability of the Prosser interbed to wells adjacent to the Columbia River indicate that the Saddle Mountain No. 3 basalt surface may not realistically be used as the lower boundary for electrical analog simulation of ground water flow.

South of Gable Mountain but north of 200-East Area other aquifers in the basalt series are also subject to recharge by unconfined ground waters and potentially by separations plants effluents. Well 699-53-55 encountered a channel in the basalt eroded through what is now interpreted as the Saddle Mountain No. 2 basalt flow, and that bottoms in the Beverly Member, an extensive interbed. Data from that well and considerable other evidence indicate that the Priest Rapids No. 4 basalt flow, that lies beneath the Beverly Member and much of south-central Washington, is probably the optimum stratigraphic unit for use as the lower boundary for electrical analog simulation of ground water flow. Burial of that Priest Rapids flow by later flows that protected it from subsequent erosion probably makes its characterization by mathematical expressions far simpler than if its structural features had been modified by stream erosion.

RADIOLOGICAL AND HEALTH CHEMISTRY

Multidimensional Gamma Spectrometry

Some preliminary data have been obtained using the new multidimensional analyzer in conjunction with our large three-crystal detector. For

those radionuclides for which this system is applicable, significant increases in sensitivity and selectivity are possible. In reactor effluent water analysis As^{76} , Cu^{64} , Ba^{140} , La^{140} , Co^{60} and Sc^{46} can be determined without prior chemical separation, and measurements of Na^{24} and Mn^{56} at short decay times can be made more accurately.

Environmental Gamma Spectrometry

A 9-3/8-inch diameter by 4-inch thick sodium iodide detector with a portable multichannel analyzer was used to determine the feasibility of high sensitivity field measurements of radionuclides. In a boat it was possible to measure ZrNb^{95} , Ru^{103} and Ru^{106} due to fallout in the Columbia River above the Hanford reactors and in the Snake River. In the Columbia River just below the reactors Na^{24} , Cl^{38} and Mn^{56} could be measured. In the McNary Dam reservoir Na^{24} , Zn^{65} , ZrNb^{95} , Ru^{103} , Ru^{106} and Cr^{51} could be measured. In the atmosphere up to 12,000 feet altitude, ZrNb^{95} , Ru^{103} - Ru^{106} , Ce^{141} - Ce^{144} , and Bi^{214} were measurable.

Iodine-131 Release Test

During this report period a special air sampling head was designed, tested, procured from an off-site fabricator, and prepared for the proposed iodine release test. The polyethylene sampler is designed to include a membrane filter to remove particulate material, a four-inch bed of silver-coated copper turnings to remove I_2 , a charcoal-impregnated filter and a 1-1/2-inch bed of activated charcoal to remove the remaining volatile iodine compounds. About 600 of these will be used during the test.

Iodine-131 Studies

In studies concerned with I^{131} in fallout, some of the I^{131} is found in a volatile form not as readily collected by sampling devices as is I_2 . These materials are collected by activated charcoal beds. Tests were made on the gas released from the Purex stack using a caustic scrubber followed by a charcoal trap. Since water vapor interferes with the collection of iodine compounds on charcoal, air was introduced into the line between the caustic scrubber and the charcoal to reduce the humidity of the stack gas sample. Results obtained to date indicate that less than one-half of the total iodine of the sample was collected by the caustic scrubber. This scrubber is of the same design as those used to monitor the I^{131} output. Although the chemical form of this volatile compound containing I^{131} has not yet been determined it appears to be an organic iodide.

Radiation Chemistry

In irradiation bleaching of erioglaucine solutions at dose levels above about 10^4 rads a post-irradiation bleaching is noticed with most solutions.

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This may be due to secondary "long-lived" free radicals formed in the processes following irradiation. Study of the kinetics of this post-irradiation effect should provide information on the nature of these radicals and their reactions. This effect should be understood before continuing the studies of the variation of protection index with temperature. Preliminary data show, however, that thiourea, which is an excellent protective agent, appears to be equally effective at 4.5, 25, 45 and 65 C. These data support the idea that thiourea is the center of an extensive water structure even at high temperatures. Preliminary nuclear magnetic resonance spectra on thiourea solutions show a broadened water proton peak which also supports the "structured water" concept of these solutions.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Aerosol Sampling Studies

The errors associated with sampling a coarser grade ZnS powder* at sampling flows much lower than isokinetic were measured at a wind tunnel speed of about 7 miles per hour. The total fluorescence was used as a measure of the ZnS on the filters. When the sampling rate was 1/14 the isokinetic rate, the apparent concentration based on this flow was 2.5 times the actual concentration. At flows nearer isokinetic, this error was progressively lowered.

Optical counting of particles 25 μ and larger showed that for a flow ratio of 1:14 an apparent dust concentration larger than the actual by a factor of about four would be indicated. At an even lower flow ratio (1:20) the error appears to be less as reported in last month's report, a result requiring verification, since there seems to be no satisfactory explanation for this observation.

A precision anemometer received recently revealed a source of error in previous wind tunnel velocity measurements which will require some adjustment of earlier data.

Gravitational Particle Settling in a Horizontal Circular Tube

An expression was derived for the gravity deposition of particles initially of uniform concentration in a horizontal circular conduit, and moving along the conduit. An earlier derivation required the arbitrary assumption that the concentration of particles in those portions of the conduit containing particles did not change. In the recent derivation this assumption was not necessary. Identical expressions were arrived at either with or without this assumption. Thus, it appears that the somewhat arbitrary assumption was valid.

*U. S. Radium Corp. No. 2330

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Aerosol Generation

Considerable operating experience was obtained with the spinning disc aerosol generator. A more powerful and sturdy air motor was installed to minimize disc wobble.

Reasonably uniform aerosols were generated having selectable particle sizes from 3 to about 12 microns. Insuring that dry particles are always produced is a problem requiring further study.

ISOTOPES DEVELOPMENT - O8 PROGRAM

Radiochemistry of Promethium

Arrangements are progressing for the irradiation of U-235 to measure the independent fission yield of Pm-146. The expected rare earth activities after a three-hour irradiation have been calculated. Although only about 1000 d/m of Pm-146 will be formed, several curies of other rare earth activities will be present. A literature search has been conducted to determine the decay schemes of the nuclides expected in appreciable quantities so that shielding requirements may be estimated, and unfamiliar activities may be identified by gamma spectrometry. Analytical methods, including ion-exchange chromatography, for separating and purifying the promethium sample are being tested in the laboratory. Elution with alpha-hydroxyisobutyrate from an ammonium-form Dowex-50 bed at ca. 80 C appears particularly attractive.

Consideration has been given to counting techniques which will permit analysis for Pm-146 in the presence of large amounts of Pm-148. Use of beta anticoincidence, Compton anticoincidence, and gamma-gamma coincidence in combination may permit analysis of short-cooled (less than one year) promethium. Availability and capability of instrumentation for this specialized counting is being investigated.

Pyrochemical Separation of Cerium and Promethium

Due to the intense heat generating rate of Ce-144 and the radiation damage problems associated with aqueous solutions and organic solvents, a pyrochemical process is under study for the separation of cerium from the other fission product rare earths. The process is based on earlier Ames Laboratory work with macro quantities of non-radioactive rare earths and involves heating a mixture of rare earth nitrates (including cerium) in a molten magnesium nitrate matrix to oxidize and preferentially precipitate the cerium (as CeO₂). Highlights of these studies are as follows:

1. At a 2:1 mole ratio of magnesium (nitrate flux) to cerium and reacting for one hour, CeO₂ recoveries were 90 percent and 98 percent for reaction temperature of 240 C and 275 C, respectively.

2. Cerium product purities of 79 percent and 73 percent resulted at temperatures of 240 C and 275 C.
3. Cerium product purity is slightly improved (83 percent) at lower temperatures (200 - 220 C) but recovery falls to 50 to 65 percent.
4. Increasing reaction time from one hour to two hours had no appreciable effect on recovery or purity.
5. With a 1:1 magnesium to cerium mole ratio, the molten salt media became a semi-solid mastic during the course of reaction. Mole ratios of 2:1 resulted in liquids at all temperature studies.
6. Analysis of cerium product indicated the following mole percent impurities: La 0.2, Pr 0.2, Nd 4.0, Sm 0.3, Gd 0.3. Mixed rare earth feed consisted of 0.9, 19.0, 33.9, 4.1, 36.2, and 2.4 mole percent Y, La, Ce, Pr, Nd, and Sm, respectively.

Preliminary results, from studies with magnesium nitrate-sodium nitrate mixtures, indicate that basicity plays an important role in the reaction and accounts for the effectiveness of magnesium. The basicity theory also suggested the similar use of ammonium carbonate, and this was confirmed. With ammonium carbonate, only insoluble CeO_2 and soluble tri-valent rare earth nitrates remain after reaction, since excess ammonium carbonate is volatile. However, unlike the magnesium nitrate process, the carbonate process passes through a solid phase and has the further potential objection that small amounts of ammonium nitrate are formed.

Fission Product Encapsulation

A new high temperature furnace has been installed and experimental work initiated on the preparation of strontium oxide heat sources. Strontium oxide cermet, and the Dynapak process for making them, are being explored with Ceramic Fuels Development Operation. A new Dynapak can has been designed (with V. L. Hammond) which will take advantage of design improvements of previous cans.

E. R. Smith for

Manager
Chemical Laboratory

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BIOLOGY LABORATORY

A. ORGANIZATION AND PERSONNEL

Dr. W. J. Billings, Duke University, who was assigned on the Summer Professorship Program to Radioecology terminated on July 26.

GENERAL

The Cary recording spectrophotometer was received during the month and placed in operation. From preliminary checks, it appears to be functioning properly.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - O2 PROGRAM

Vertan 690

The receipt of the second lot of Vertan 690 permitted a second series of bioassay on the acute toxicity of this proprietary material on some 800 young rainbow trout. The estimate of LD_{50} for the second lot of Vertan 690 was 2000 ppm which agrees closely with the previous results obtained for the first lot of Vertan 690 tested. The effect of the addition of a surfactant, F-44, was also tested. The presence or absence of 0.4% F-44 had little effect on the toxicity of Vertan 690. Another surfactant (Dowell F-33) of interest to NRD with an alkyl phenol as one of its active constituents is currently under acute toxicity tests. Preliminary results on zero-age rainbow trout (average weight about 2 g) showed total mortality at 60 ppm or greater within 40 minutes. Tests continue at lower concentrations.

Columnaris

Last year the sharp peak mortality from columnaris of zero-age rainbow trout held in river water at 146-FR was observed during July in 64 F water. (The maximum river temperature in 1962 was about 69 F and did not occur until late August.) This year the mortality pattern appears to be more variable than last year, and no sharp peak is apparent at present. The summary of the total mortality for the last five-week period from all causes for groups of fish held at various conditions are:

<u>Treatment Groups</u>	<u>Average Percent Mortality of Duplicate Troughs</u>	<u>Avg. Body Wt. (g)</u>
River water	1.1	1.81
River water heated 4 F above river temperatures	7.1	1.61
River water chilled 4 F below river temperatures	0.8	1.64
River water "sterilized" with ultra violet	0.5	1.84

Although the mortalities expressed are total rather than mortalities specifically assigned to columnaris, past studies in our hatchery indicate a strong correlation between the two. An increase of 4 F above river temperatures appears to aggravate the columnaris problem in hatchery troughs.

Deaths in laboratory fish resulting from columnaris were first noted approximately three weeks after peak concentrations of the myxobacterium were noted in the Yakima. Mortality has been greatest among both the youngest and most crowded fish.

Concentration of columnaris in the Yakima River near Prosser reached a maximum during the middle of June, with a successive decline, even though water temperatures remained essentially constant at 21 C. The cause of this "burst" of organisms will be sought.

Strains of the myxobacterium obtained from widely separate sites during 1962 were tested for their effectiveness in killing fingerling salmon and trout. Widely varying virulence was noted in strains from all locations.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Strontium

Two clinical tests are being developed so that they may be evaluated with regard to being sensitive indicators of radiation damage to the hematopoietic system of swine fed Sr^{90} . The first of these tests is used to determine the granulocyte reserve of the animals. Purified bacterial endotoxin is administered intravenously and the hematological response followed over a 48-hr period. In the normal animal, when an appropriate dose of endotoxin is given the leukocyte count is increased to approximately twice pre-injection levels. Preliminary experiments indicate that a lesser response is seen in miniature swine fed 125 μC Sr^{90} /day for over 2 years. (Further studies utilizing additional animals are now in progress.)

The second test currently being developed for use in the miniature swine determines the fragility of the erythrocyte. This test is done in vitro by placing fresh erythrocytes in decreasing strengths of hypotonic saline solutions and then determining the degree of hemolysis produced. Normally, the erythrocyte lysis is not observed until the solutions contain 0.6% NaCl or less. Hemolysis in the solutions closer to isotonicity indicates increased fragility of the erythrocytes. "Base-line" values are being established for control miniature swine, and the test will be extended to miniature swine ingesting Sr^{90} daily.

Iodine

Stable iodine feeding in cows being fed 5 μC I^{131} /day was discontinued. An abrupt increase in thyroid and milk concentration of I^{131} was noted in two cows that had been fed 2 g stable iodine per day. A third cow formerly fed 5 mg stable iodine per day showed no increase in milk I^{131} and a slight increase in thyroid I^{131} concentration. After 5 days, milk concentration of I^{131} in the two cows (previously fed 2 g I^{127} /day) reached a plateau at about two times former levels. Thyroid concentration of I^{131} in these two cows leveled off (at a value ten or more times their lowest concentration) about 3 weeks after cessation of stable iodine feeding and approached that of a control cow.

Lambs born to hypothyroid ewes appeared less mature than control lambs as determined by weight gain and skeletal radiographs. This immaturity is probably a reflection of the low milk production of the hypothyroid ewe.

Four lambs were exposed at 210 meters downwind from a release tower in the meteorology network. The exposure time was about 15 minutes following the release of 120 mc of I^{131} . The majority of the I^{131} was on the wool. It appeared that about 0.015 - 0.03 μC of I^{131} was on the wool of each sheep (1 meter² area), following exposure to what was estimated as 1.5×10^{-5} mc of $\text{I}^{131}/\text{m}^3/120$ mc release. No significant thyroid uptake was detected. These values are estimates and subject to modification when all the data are evaluated from this cooperative study.

Neptunium

Serum from female rats injected with 2 mg Np^{237} /kg body weight was assayed for glutamic-pyruvic transaminase, isocitric dehydrogenase, and quinine oxidase. Levels of all these enzymes were very substantially elevated at 24 and 48 hours following neptunium injection. Only isocitric dehydrogenase showed a significant elevation at six hours following neptunium injection. These effects are indicative of severe liver damage. It will be interesting to determine whether similar effects are noted in male animals for whom neptunium is much less toxic.

Plutonium

Comparative hematological data were obtained from rats injected with 20 μC Pu^{238} or Pu^{239} . Similar white blood cell depression was noted with both isotopes about the fourth day post-injection. Red cell depression was not evident until after the second week, at about which time the Pu^{239} animals expired. Red cell counts in the Pu^{238} animals decreased to about 65% of control values and remained at this level despite evidence of essentially complete bone marrow destruction. It would appear that the spleen or other accessory hematopoietic tissues maintained the red blood cell level in these animals. The greater toxicity of Pu^{239} may be attributed to the fact that the greater mass of this nuclide causes greater deposition of it in spleen and other soft tissues which prevents this ectopic hematopoiesis.

Inhalation Studies

Promethium-147 in 0.067 N perchloric acid was given two dogs by inhalation and two by intravenous injection. Preliminary results indicate high levels of Pm^{147} in liver, skeleton, and lungs 30 days after administration by both routes.

Three dogs, exposed to I^{131} released from the meteorology tower, were apparently outside the path of the release.

One dog of the six exposed to Pu^{239} released at the Tonapah test range was sacrificed for tissue analysis. Preliminary results of excreta analyses show daily excretion of nanocurie quantities of plutonium during the first month after exposure.

Several agents were tested in rats for removal of inhaled plutonium. All PuO_2 experiments were negative. DTPA appeared to be more effective than diethyldithiocarbamate in removing inhaled plutonium nitrate; however, analytical results are not complete.

All of the dogs on the long-term Pu^{239} experiment were tested for distemper, hepatitis, and leptospira canicula antibody titers. All dogs have normal titers despite their low circulating lymphocyte levels. It was suspected that the low lymphocyte levels of these dogs which have persisted for 2 to 3 years in some cases would have depressed their antibody response. Angiocardiography was performed on one dog exposed almost four years ago to $\text{Pu}^{239}\text{O}_2$ and showing severe respiratory symptoms. No vascular changes other than right heart enlargement were seen.

Preliminary bronchoscopic work was completed using radio-opaque materials to test procedures for washing inhaled radioactive particles from the lung and to develop a technique for exposing individual lobes of the lung to radioactive aerosols. The latter is proposed as a method for extending our studies of the biological response to lung-deposited radioactive particles.

A commercially available highly absorbant, dust-free animal bedding material (Cedar-oma) has proved to be acceptable for the dog runs. Bedding material is required to absorb moisture from the runs which causes severe dermatitis. However, most materials such as sawdust, wood shavings, etc. are dusty and therefore undesirable for dogs on inhalation studies.

A second uranium ore exposure was completed with the serial sacrifice of the rats. No results are available from either experiment.

Gastrointestinal Radiation Injury

Thirty-five rats were exposed to total-body fast neutron irradiation in the Physical Constants Test Reactor. The animals were shielded with 2 inches of lead covered with U^{235} foil, which resulted in a neutron spectrum similar

to that obtained from the Godiva Critical Assembly at Los Alamos. A maximum neutron energy of 2 mev was observed, with the majority of neutrons falling in a range around 800 kev. At exposures of 1300 watt minutes, death occurred after four days, clearly due to intestinal radiation damage. The latest death observed was at 16 days after 900 watt minutes and was attributable to bone marrow damage. No deaths have occurred from exposures of 800 watt minutes or less. From these results, the median lethal level was estimated at about 300 rads of neutrons. The gamma contaminant at this level amounted to about 140 rads. Assuming additivity of gamma and neutron effects an RBE of approximately 2 is indicated for these fission neutrons, in agreement with considerable data previously reported by others. These results establish the feasibility of using the PCTR in studies of neutron effects on the gastrointestinal tract.

Cell Proliferation in Fish

Forty-four yearling silver salmon were acclimated to 10 C Columbia River water for a week. Each fish was then given an intraperitoneal injection of 1.8 μ c tritiated thymidine per g body weight. The fish were starved throughout the acclimation and sampling period with no apparent harm from fasting. Fish were killed at 11 different time intervals ranging from one hour to 14 days post administration. About ten different tissues which are often associated with radiation damage were fixed and sectioned for autoradiographs. The final results on this study of cell proliferation in fish will not be available for some months; however, the successful technical procedures set the pattern for the second and third experiments in which fish will be subjected to environmental stress of salt water and X-irradiation.

Secondary Disease Studies

Experiments employing rat bone marrow to protect various strains of irradiated mice have indicated that the marrow from female rats produces lower and more delayed antibody titers than previously observed in studies employing bone marrow from male rats.

Hepatic Tumor Studies

Laparotomies were performed upon a number of rats after 16 weeks on a diet containing the carcinogen N, N-dimethyl-p-phenylazoaniline (DMAB). Some of these animals, at the start of carcinogenic feeding, had received injections of Pu²³⁸ or Ce¹⁴⁴. Liver damage, possibly cancerous, was apparent in about half of the animals examined. No conclusions can be drawn at this time as to the effects of the injected radionuclides.

Microbiology

Prolonged cultures of yeast on D₂O produced a progressive decline in the ability of cells to utilize a carbohydrate energy source. In such cells there was 1.5 times as much DNA but only half as much RNA per unit mass as in cells cultured in H₂O. This observation may be a clue to the observed inhibition of mitosis by D₂O as observed in many cells.

Radiation Effects on Insects

Using matings of different aged males and females of T. castaneum, it was shown that males mature on the second and females on the fourth day of adult life when the temperature is held at 29 C. Reproductive productivity over a limited time was doubled by increasing the temperature at which the beetles were grown from 24 to 32 C.

Adult beetles were exposed to fast neutrons in the Physical Constants Test Reactor to determine effects on reproduction. Neither the physical characteristics of the exposure nor the reproductive effects are known as yet.

Plant Nutrition

In an attempt to quantitate the potential transpiration rate of plants at any given time, leaves were exposed to a solution of the fluorochrome sodium 3 hydroxy-5, 8, 10 pyrenetrisulfonate. No penetration was observed through surfaces not having stomatal openings. However, the amount of dye penetration could not be repeated in successive exposures even when environmental conditions were carefully maintained. Thus this, as well as the surface replication method, fails to define the extent of stomatal opening.

Plant Ecology

Selenium-75 placement at 0.5 meters, within the depth of annual moisture penetration, was detected in the leaves of hopsage. Placements at 1 meter, below the seasonal penetration of soil moisture, were not detected in the leaves of hopsage. These data suggest that hopsage is not dependent upon subsurface moisture for existence in this semi-arid region.

Using moisture stress analysis and plant uptake of isotopic and dye tracers it was possible to predict and test the water source from which plants derive their moisture. Thus it was clear that greasewood uses groundwater whereas many other perennials, as well as the annuals, derive their moisture from the annual moisture deposited at depths of 50 to 150 cm, just below the roots of the annual grasses. Knowledge about greasewood might be useful in studying movement of H³ through groundwater in some areas of the reservation.

Columbia River Limnology

Periphyton samplers were placed in the F-1 riffle area to initiate comparative studies of productivity using biomass, chlorophyll extraction, and C^{14} uptake as indices. Some samples will also be analyzed for Zn^{65} accumulation.

Preliminary samples for P^{32} and Zn^{65} accumulation by plankton were taken at several points across the stream below Hanford to investigate the "plume" effect of the effluent. A definite plume was found on the west side of the river and will be considered in future sampling.

Rattlesnake Springs Limnology

With completion of the dam and stabilization of the water level in the new impoundment, C^{14} uptake studies were initiated to investigate the seasonal and sequential aspects of productivity in a newly formed impoundment.

HA Kornberg/es
Manager
BIOLOGY LABORATORY

HA Kornberg:es

TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORY

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposiums

McClellan, R. O. and L. K. Bustad. Toxicity of significant radionuclides in large animals. Conference on Veterinary Toxicology sponsored by the New York Academy of Sciences and the American College of Veterinary Toxicology, New York, N.Y. July 25, 1963.

McClellan, R. O., H. W. Casey, H. A. Ragan, and L. K. Bustad. Diagnostic radioisotopes in veterinary medicine - present and future. Annual Meeting of the American Veterinary Radiology Society, New York, N.Y. July 29, 1963.

b. Seminars (Off-Site and Local)

Hungate, F. P. Genetics. University of Washington Summer Institute for Radiation Biology, Seattle, Washington. July 1, 1963.

Smith, V. H. Radiation protection by chemical means. University of Utah Summer Institute, Salt Lake City, Utah. July 25, 1963.

c. Seminars (Biology)

None

d. Miscellaneous

Stuart, B.O. Biological effects of ionizing radiations. Presented to radiation monitors on 7/5 and 7/11 - 100-D Area.

II. Articles Published

a. HW Documents

None

b. Open Literature

Cushing, C. E. and P. A. Olson. 1963. Effects of weed burning on stream conditions. American Fisheries Society 92: 303-305.

Uyeki, E. M. 1963. Liver and spleen phosphatase activity in bone-marrow transplantation. Intern. J. Radiation Biology 6: 257-269.

Rickard, W. H., and L. M. Shields. 1963. An early stage in the plant recolonization of a nuclear target area. Radiation Botany 3: 41-44.
(Work performed elsewhere.)

III. Visits and Visitors

a. Visits to Hanford

- 7/11/63 - 9 High School Students, Columbia Basin College, Pasco. Tour.
- 7/11/63 - Dr. C. E. Ballou, Univ. of California, Berkeley. Discuss research with R. C. Thompson and RT O'Brien.
- 7/16/63 - 3 USPHS officials and 2 from Washington State Department of Health toured Biology, escorted by HG Hicks, local AEC office and R. F. Palmer.
- 7/16/63 - R. F. Steadman, Donald Bradford, G. W. Johnson, A. Lazure, A. Flax, and J. Davis from the Department of Defense, Washington, D. C. toured Biology and discussed research with HA Kornberg.
- 7/18/63 - 14 Members of a graduate level course in Public Health in Richland toured Biology and were oriented in the research program by R. O. McClellan.
- 7/19/63 - Robert Dodson, Idaho State School System. Discussed radiation effects on the nerve system with H. E. Erdman.
- 7/26/63 - J. F. Stara, USPHS, Cincinnati. Discuss toxicity studies with LK Bustad, WJ Bair, JF Park.
- 7/26/63 - Capt. W. R. Godden, Kirtland Air Force Base, New Mexico. Discuss Project Roller Coaster with W. J. Bair, C. E. Breckenridge, JF Park.
- 7/28/63 - Dr. D. J. LeTourneau, Ag. Chem. Dept., University of Idaho, Moscow, Ida. Consult on research with Dr. D. D. Mahlum.
- 7/29/63 - Dr. Jack Hsieh, Seattle, Wash. Interview and tour with FP Hungate.

b. Visits Off-Site

- 6/20-9/1 - W. C. Hanson will be conducting research in Alaska.
- 7/1/63 - F. P. Hungate presented a lecture at the Summer Institute for Radiation Biology at the University of Washington, Seattle.
- 7/2/63 - M. P. Fujihara, obtained Yakima River water samples at Prosser.
- 7/10-11 - J. F. Park attended an AEC Meeting on biological hazards of plutonium at the Lovelace Foundation, Albuquerque, N.M.
- 7/11/63 - R. E. Nakatani attended a meeting at the Oregon State University, Corvallis, Oregon.
- 7/15-9/15 - L. L. Eberhardt and D. G. Watson will be conducting research on large mammal food chain studies in Alaska.
- 7/25/63 - C. E. Cushing and W. H. Rickard conducted fallout sampling of water and vegetation at the Wooten Game Range, Dayton.
- 7/29-31 - H. A. Kornberg attended the Symposium on Closed Ecological Systems sponsored by Lockheed in Palo Alto, California.
- 8/1-2 - H. A. Kornberg visited Ames Research Center at Moffett Field to discuss space research and USNRDL in San Francisco to discuss radiation research with Dr. Vaughan.
- 7/11 & 15 - V. G. Horstman inspected feed pellets at Othello, Wash.
- 7/20-8/6 - R. O. McClellan discussed research on large animals with Dr. Earl, FDA; Maj. Seigneur and Capt. Persing in Wash. D.C. He presented papers at the Conference on Veterinary Toxicology and American Veterinary Medical Association Mtg. in New York City.

IV. Achievements

None

V. Honors and Recognitions

None

VI. Professional Group or Organization Assignments

None

APPLIED MATHEMATICS OPERATIONMONTHLY REPORT - JULY, 1963ORGANIZATION AND PERSONNEL

A. D. Wiggins terminated to accept a position as Associate Research Biostatistician at the School of Public Health, University of California, Berkeley, California. R. J. Wood joined the group on July 24 as a Statistician.

OPERATIONS RESEARCH ACTIVITIES

Analysis for IPD of frequency of outage work has been completed and submitted to K- and H-Area managers. Initial response has been favorable. The analysis was also presented to the Manager, Finance, N-Reactor, as illustrative of potential aids for N-Reactor control problems.

Work on the Tri-City Area economic model continues. Work is now being done to project base period financial data (incomes and expenditures) to the current year (average values for 1963).

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTSN-Reactor Department

A systematic sequential sampling plan was developed in order to determine whether the percentages of defective tubes in the different heat exchangers are at an unacceptably high level. Further assistance was given in establishing confidence limits on these percentages for various heat exchangers on the basis of the sample results.

Irradiation Processing Department

A study is being made to determine the operating characteristics of the MERCY program in testing for nonconstant biases. The power of the test is being estimated by a sampling procedure because of the difficulty of obtaining an analytical solution.

In the routine quarterly audit using MERCY to assess the nature and extent of measurement variation in nondestructive bond integrity measurements, transformations are used to stabilize the variance. A review of these transformations indicates that some change is necessary for the total and maximum successive bad discs variables. The new transformation is now being tested on the audit data.

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Preirradiation dimensional measurements have been taken on fuel elements canned by the AlSi and hot-die-sizing processes preliminary to a production test to compare their reactor performance characteristics. A study is being made to determine how the populations of these measurements for the two processes compare.

Counting times and counting numbers were determined in order that contaminated and noncontaminated materials could be discriminated between with certain specified degrees of assurance.

A study is being made to determine whether there are significant differences between the stud pull results for three cleaning processes on fuel elements canned by the hot-die-sizing process.

Assistance was given in designing a test to compare different deoxidation methods in the preparation of nickel-plated cores for the hot-die-sizing process.

A comparison is being made between the pull strength of rails made with the transverse grain and with the longitudinal grain. A test is also being analyzed which compares soap and 3-in-1 oil for cleaning in the ultrasonic welding of rails.

An analysis is being made to see whether the corrosion rates of welded plates differ from those of unwelded plates.

Continued assistance is being given in fitting R values according to their circumferential tube locations by a harmonic-type regression relationship. The purpose of this is to determine how the temperature imbalance in the downstream portion of the tube is affected by the eccentricity of the tube.

An analysis is being made, using UE-1 internal and external count values to compare the normal "F" canning process with an experimental process, determined to be better in some respects by previous tests.

Data are being analyzed in order to determine the reproducibility between different observers of a metallographic technique which estimates the percentage of UAL₂ around the circumferential bond obtained by a transverse cut of a fuel element.

Chemical Processing Department

Current status of minor constituent (70-58) content control required extending part sampling to three sampling positions (flange, boomerang, and sprue) to evaluate possible segregation within part. For plutonium content control of delta parts, part-by-part acceptance will be used until sufficient data

is available to permit use of random sampling. As a protection against nonrandom measurements or impurity segregation, a control limit was adopted for the difference between the observed "by difference" impurity estimates on the flange and sprue sampling positions.

To provide a more meaningful description of actual inspection practices, a sample of current production is being followed through the entire residence path in the inspection facility. This basic information is needed to permit an effective method of recording and processing inspection data, to optimize inspection procedures, and to estimate the reliability of the inspection system.

In response to a request, an over-all review of the statistical quality control of sampling and volume measurements by CPD is to be made.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HANFORD LABORATORIES

2000 Program

Work continued on the analysis of data from a block of 48 experimental runs on the pulse column. A gamma-type mathematical function containing three parameters and used to express organic volume fraction as a function of linear position in the column was fitted to each of these experimental runs. A regression analysis is in progress to correlate column pressure drop on the column driving variables and three additional variables determined from the organic volume fraction profile for each run. Specifically, these variables are the position of the organic interface at the base of the column, the maximum organic volume fraction between the points in the column where the pressure drop is measured and the total organic resident in the column between these two points.

The mathematical development concerning the boundary value approach to the solution of the nonlinear system of second order differential equations expressing mass transfer in the column was completed. This approach considers the column as a large number of small cells in each of which the system of differential equations is assumed to be linear. A smooth transition is demanded across adjacent compartment boundaries by forcing concentrations and concentration derivatives with respect to column position to be identical on the common boundary. Data at the top and bottom of the column and the extractant and feed stream ports provide absolute boundary conditions and give a unique solution to the system. The mathematical development reduces this system to a single set of four simultaneous linear equations expressing the solution in the top compartment of the column. Solutions to all other compartments can be expressed in terms of the solution in this top compartment. A FORTRAN language program has been written and is currently being debugged to perform the reduction to a single compartment and solve the

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system in that compartment. Test data are being compiled to check out this program.

A solution to the partial differential equation governing the electrostatic field in a controlled-potential coulometer has been obtained. An examination of this solution should yield valuable information on the efficiency of an ion separation process.

3000 Program

The first experimental magnetic tape made on the 7090 for the purpose of numerically controlling the prototype Sheffield rotary contour gauge has been completed. This tape is to be tested on the mock-up model being assembled at the Schenectady laboratories of Equipment Development Services.

Documentation of the EDPM program which produces the magnetic tape input for the experimental, numerically controlled δ - ω lathe is nearing completion. Priority scheduling has prevented any recent experimentation on the lathe itself.

4000 Program

Further work was done on the regression analysis of plutonium homogeneous spherical critical mass experimental data to determine the relative merits of several reactor physics code systems which theoretically calculate eigenvalues and critical masses. The regression analysis technique is being extended to a set of Yankee critical mass experimental data and a set of plutonium aluminum critical mass experimental data.

Preliminary results from the analysis of the spherical experiments indicate that four-group macroscopic cross sections should probably be used in place of two-group cross sections. Current efforts are directed toward preparing the four-group cross section data and entering it into the regression analyses.

An EDPM program was written and put into service which aids in computing the heat transfer characteristics of the radiant heat spray calciner.

Because of the lack of appropriately sized sieves, particle sizes corresponding to those called for by the often successful regular cylindrical packing model have not been available for a current fuel element model. Since difficulty has been experienced in obtaining sufficiently high densities with the available sieve sizes, particle sizes and mass ratios have been based on a recently developed (and incomplete) cylindrical-spiral packing model.

A new mode of driving the vibrational compactification equipment has been suggested which should possess several desirable properties lacking on the present equipment. These properties were suggested by a mathematical analysis.

Aid and assistance was given in formulating and solving a set of differential equations describing an experiment conducted toward the development of reactor coolant systems.

5000 Program

Work progressed satisfactorily on the program to index crystals. Plans were developed to store parts of the program permanently on tape.

A new input subroutine was written for program GEM and is now being debugged. This input routine will read "unsummed spectra" on cards and do the necessary summing required by the GEM program. The routine will also accept cards of the previous format and print a listing of card input in each case. Primary purpose of the new input routine is to make the GEM program available as a routine research tool to laboratory radiochemists without the complexities of going through the IRA system.

Several specifications of improvements and modifications to the IRA data processing system were made. A subroutine was written which instructs the Benson-Lehner plotter to plot multichannel spectrometer data as background corrected counting rate spectra. The routine is now being debugged. Work continued, using Monte Carlo techniques, to improve the GEM program calculation of estimates necessary for IRA system evaluations. Particular effort is directed toward the optimum method of calculating uranium fission product contamination to program estimates, which involves the question of standards estimation and the optimum grouping of channels when analyzing sample data.

6000 Program

First results from the Monte Carlo program for one-dimensional diffusion were delivered to the customer.

Data was analyzed from Ce^{144} - Pr^{144} inhalation experiments performed on dogs. The purpose of the analysis was to express the whole-body retention of Ce^{144} - Pr^{144} after inhalation of oxide particles as a function of time and to compare the statistical models for different dogs.

Statistical analysis of data from experimental studies being conducted by Biology on Tribolium was continued. The effects of X-irradiation, temperature, time, species (*Tribolium confusum* and *Tribolium castaneum*), and group of species (separate or mixed) on the number of adult progeny by mating pairs of Tribolium are being investigated.

Other

The statistical analysis of data from a well survey study was completed. The purpose of the analysis was to calculate the precision with which the

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location and elevation of project wells can be determined. The information is needed to assess statistically the several sources of variation that may enter the in-place measurement of the permeability distribution for the project.

GENERAL

Assistance is being provided for evaluation of bids received on the process control computer for Hanford Laboratories.

R. Y. Dean

Acting for
Manager
Applied Mathematics

RYDean:dgl

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REACTOR DEVELOPMENT - O4 PROGRAMPLUTONIUM RECYCLE PROGRAMCurium Production Study

Calculation of curium-244 production and its influence on fuel cost at various prices is essentially completed.

Results have shown that the most economical fuel schemes would be plutonium enrichment of natural uranium or thorium. Most other schemes considered would be economical only if the curium-244 price were set at from \$750 to \$1000 per gram. As the percent of plutonium-242 in a plutonium batch increases, its worth as a curium-244 producer also increases. Plutonium with a composition of 95% Pu-239, 5% Pu-240 will yield 0.3 mill fuel cost reduction at 30,000 exposure whereas plutonium (76% Pu-239, 18% Pu-240, 5% Pu-241, 1% Pu-242) would show approximately 0.5 mill reduction. In each case a \$400 per gram curium-244 credit was allowed.

It is obvious, from this example, that the higher plutonium isotopes have added value in a thermal reactor because of their ability to produce curium-244. A preliminary evaluation of this added value when curium-244 is set at \$350 per gram follows:

TABLE IISOTOPE VALUES WITH CURIUM CREDIT

Isotope	Pu-239	Pu-240	Pu-241	Pu-242	Am-243
Cm-244 Producer Value, \$/gram	4.50	10.00	15.00	60.00	200.00
Thermal Reactor Value, \$/gram	<u>10.00</u>	<u>0.00</u>	<u>10.00</u>	<u>0.00</u>	<u>0.00</u>
Total	\$14.50	\$10.00	\$25.00	\$60.00	\$200.00

This preliminary value is for a fixed curium-244 value of \$350 per gram, which is not necessarily the worth to users. The values are determined to yield the same fuel cost for a slightly enriched uranium cycle and a cycle using plutonium to produce curium-244.

The purity of curium-244 in the curium-242-curium-244 mixture must be as high as possible (at least 98%) to achieve a constant power output over a long period of time. Hence, it appears desirable to recycle both the plutonium and the americium formed in each irradiation. After two recycles of this material, each achieving 15,000 megawatt days per ton exposure, the resulting purity was above 98%. The purity is mainly a function of the plutonium-242 cross section, the flux, and plutonium-241 half life. The plutonium-241 half life is fixed, but the flux level can be altered by increasing the specific power and to a lesser extent, the flux level can be altered by changing the neutron spectrum hardness or the neutron temperature. (The influence of neutron spectrum and temperature on flux level is due to epithermal fission of plutonium-239 and plutonium-241.) For a given initial reactivity, the flux increases in a plutonium-uranium-238 fuel as the neutron temperature increases due to the reductions of the uranium-238 cross section and increases in the plutonium-239 and plutonium-241 cross sections which allow criticality at a lower enrichment level. This same effect is much less pronounced in the plutonium-thorium system. Increase in spectrum hardness also increases purity as the plutonium-242 cross sections increase; but, in some cases a moderator temperature change from 100 C to 600 C will be more effective than a change in spectral index from 0.05 to 0.25.

Zone Spectrum

The MELEAGER code is able to simulate a reactor with separate fuel zones, each with a different neutron spectrum. By moving fuel into zones where the spectrum is favorable, so that the reactivity is greater than one, but not so much that a large amount of neutrons must be absorbed in control, a more efficient utilization of the fuel is realized than with straight batch loads.

Some calculations of burnup and fuel cost have been done in a D₂O moderated reactor, although the zone spectrum principle can be applied to any fuel and moderator combination. The fuel costs were calculated for six fuel schemes in the D₂O reactor and are summarized in Table II.

TABLE II

ZONE SPECTRUM MINIMUM FUEL COSTS

Fuel	U-235- U-238	Pu-U-238	U-233- U-238	U-235- Th-232	Pu-Th-232	U-233-Th
Fuel Cost mill/kwh _e	1.101	0.953	0.977	1.607	1.131	0.845
Fuel Exposure Mwd/ton	15,158	17,536	18,659	23,080	16,494	25,624

Fuel Homogenization and Neutron Spectra

Several trial calculations were made using the ALTHAEA diffusion code to illustrate the influence of fuel and coolant configuration on neutron flux. Table III lists the configuration used in six typical calculations.

TABLE III
CONFIGURATION

Radius in cm	Region				
	1	2	3	4	5
	1.3	1.8	2.2	2.54	12.0
Case 1	PuO ₂	PuO ₂	PuO ₂	PuO ₂	Carbon
Case 2	PuO ₂ -H ₂ O	PuO ₂ -H ₂ O	PuO ₂ -H ₂ O	PuO ₂ -H ₂ O	Carbon
Case 3	PuO ₂	PuO ₂	H ₂ O	H ₂ O	Carbon
Case 4	PuO ₂	H ₂ O	PuO ₂	H ₂ O	Carbon
Case 5	H ₂ O	PuO ₂	PuO ₂	H ₂ O	Carbon
Case 6	H ₂ O	PuO ₂	H ₂ O	PuO ₂	Carbon

For simplicity, pure plutonium-239 was used and the six cases each contained the same total amount. In Case 2, the PuO₂ was assumed to be very finely divided and mixed with water. Case 1 is presented as the extreme with no coolant (or gas coolant) and assumes the PuO₂ is mixed with negligible cross section diluent such as MgO. The remaining four cases represent various fuel element situations, solid rod, rod and annulus, internal-external cooled, and inverted cluster. The computed spectral index and Westcott flux are presented in Table IV (normalized to flux of 1.0 in Region 5).

TABLE IV
SPECTRAL INDEX AND FLUX DISTRIBUTION

Case	Region									
	1		2		3		4		5	
	r*	Flux	r*	Flux	r*	Flux	r*	Flux	r*	Flux
1	0.62	0.28	0.52	0.34	0.43	0.41	0.35	0.49	0.136	1.0
2	0.58	0.35	0.47	0.42	0.36	0.55	0.25	0.76	0.110	1.0
3	0.54	0.16	0.33	0.26	0.14	0.71	0.09	1.06	0.52	1.0
4	0.57	0.20	0.34	0.45	0.29	0.39	0.17	0.85	0.081	1.0
5	0.25	0.63	0.42	0.29	0.34	0.35	0.19	0.82	0.086	1.0
6	0.25	0.86	0.43	0.39	0.34	0.65	0.31	0.52	0.129	1.0

* r = Westcott spectral index.

It should be noted that the spectral index - flux product, which is a measure of the epithermal flux, is largest and most constant in Cases 1 and 2. In fact, the fast flux is slightly higher in Case 2 where water is mixed with the fuel. The inverted cluster, Case 6, also produced large values for the epithermal flux.

Fuel Conservation Study

The objectives of the fuel conservation study were clarified and are the following:

1. To be able to compute for any reactor simulation a variety of indices, including those proposed by other investigators, and the bracketing values of these.
2. To formulate these indices for a plutonium recycle system.

It should be noted that often the proponents of various different reactor concepts have proposed fuel conservation indices that favor their particular design. Therefore, because this program will enable a comparison with the different indices and with more rigorous formulations in a wide variety of situations, considerable insight into the merits of each will be gained. Comparison with the most practical index of reactor performance, the total fuel cost will also be made.

A sample computation of the conservation indices for plutonium recycle in two water moderated reactor types is shown in Table V. These data show both the effect of reactor type -- by comparing the simulated pressurized water reactor with the simulated boiling water reactor -- and the effect of plutonium recycle -- by comparing Step 1 with Steps 2, 3 and 6 in each case. Note that the burnup data for this computation are taken from the plutonium value study to be reported in HW-72217 - 2 and that the reactor specifications and the recycle criteria are as stated therein.

More rigorous conservation indices are currently being formulated to reflect the fact that the uranium discharged from a reactor has a composition that is different from that of natural uranium.

Concentrations greater than natural will receive a credit that is proportional to the amount of U-235 that would have been lost in tails had this uranium been generated from natural uranium in the diffusion cascade. Concentrations less than natural but greater than tails will be penalized accordingly. In all of the present formulations that give a credit for the uranium in the discharged fuel (for example, megawatt days per fissile gram lost in Table V), the uranium composition does not influence the calculation unless it is below the tails composition in which case the U-235 contained is considered to be lost.

Multicomponent Diffusion Cascades

A calculation designed to roughly estimate the cost of "enriching" prospective structural materials, that is, molybdenum, in their low absorption cross section isotopes (Mo-98 and Mo-100) is in progress. It is assumed that the gaseous diffusion process is used and the cascade is operated without stripping the tails stream (that is, as a simple cascade) to avoid the problems associated with mixing two streams having different isotopic abundances. Because extremely "enriched" compositions will not be encountered and since the tailings will have an appreciable value, it is likely that, in any event, a simple cascade would be used in practice.

If it is assumed that the isotopes will effuse in inverse proportion to the square root of their molecular weights, the feed cost and tails credit portion of the enrichment cost can be determined by mass balance relations. The separative duty portion of the cost will be approximated by comparison with the cost of uranium enrichment.

TABLE V
CONSERVATION INDICES FOR PLUTONIUM RECYCLE
IN TWO DIFFERENT REACTOR SIMULATIONS

Graded Irradiation									
July 1, 1962 Uranium Price Schedule(1)									
Step No.	Cumulative Exposure Mwd/t	Fissile Enrichment, %			Conservation Indices, Mwd per:				
		Total	Uranium	Final	Gram U-235				Fissile Gram Destroyed
					in Nat. U				
					Gram Lost				
Simulated Pressurized Water Reactor(3)									
1	4,300	2.48	2.48	0.97	0.53	0.77	1.06	1.74	
2	48,300	2.45	2.05	1.02	0.53	0.85	1.13	1.77	
3	72,600	2.49	2.05	1.05	0.54	0.89	1.16	1.79	
6	148,500	2.58	2.11	1.05	0.54	0.93	1.18	1.79	
Simulated Boiling Water Reactor(3)									
1	22,300	1.59	1.59	0.73	0.24	1.18	1.54	2.81	
2	45,900	1.66	1.22	0.77	0.23	1.42	1.68	2.83	
3	69,600	1.71	1.21	0.80	0.25	1.53	1.75	2.83	
6	140,000	1.75	1.24	0.83	0.27	1.62	1.90	2.79	

(1) Optimum tails composition = 0.2531%.

(2) Determined by minimum total fuel cost (see HW-72217).

(3) As described in HW-72217.

OTHER ACTIVITIESThe Hanford Isotopes Production Plant

HW-77770, "The Hanford Isotope Production Plant Engineering Study," a blue cover document, was completed and released to the Atomic Energy Commission, Division of Isotopes Development. This study which was requested by the Commission last fall reported on the preliminary design and the economics of a large scale integrated facility for the production of four encapsulated fission products, strontium-90, cesium-137, cerium-144, and promethium-147. This new radiochemical processing plant of unconventional circular design was estimated to cost \$14,300,000. The annual production basis, costs including depreciation, and full credit to waste management for the cost of the feed materials and current production cost for these encapsulated products are as follows:

Isotope	Annual Production in Megacuries		Cost				\$/kwh _e * HIP
	Present	HIP	\$/curie		\$/gram HIP	\$/watt _{th} HIP	
			Present	HIP			
Sr-90	5	10	1.50	0.127	18.00	19.00	(10 yr)** 5.60
Cs-137	3.5	10	1.50	0.102	8.90	21.10	(10 yr)** 5.80
Ce-144	3.5	100	0.50	0.007	23.10	0.90	(1 yr)** 5.60
Pm-147	0.5	30	1.00	0.033	30.60	92.60	(3 yr)** 143

* Assumes 5% conversion efficiency thermal to electrical. For comparison the cost of electricity from flashlight batteries is \$100 per kwh_e.

** Duration of use of isotope in thermoelectrical device. For comparison the electricity cost from Pu-238 (at a low figure of \$500 per gram) over a ten year period would be \$228 per kwh_e.

The low costs for isotope production are obviously due to the large production of the four products. The cost for strontium-90 from the plant exclusive of the credit to waste management for the feed is less than one-half the cost currently incurred to ship the bulk product from Hanford to Oak Ridge.

Nuclear Health and Safety

AEC approval was obtained for the use of higher enrichments of $\text{PuO}_2\text{-UO}_2$ fuel elements in the PRTR. The use of 1% plutonium is authorized providing the reactivity does not exceed that of a Mark I plutonium-aluminum fuel element containing 330 grams of fissile plutonium. AEC approved the use of long-cooled irradiated fuel in the Plutonium Recycle Critical Facility.

Evaluation of a drop test of a model of the HAPO I-B cask was completed and documented (HW-77962, "Drop Test of an Unbuffered 1/4 Size Model HAPO I-B Cask"). The results indicate that the cask would not rupture or release its contents if dropped during routine operations without the protective buffer. This information will be transmitted to the Technological Hazards Council in response to a recent recommendation.

Containment Test Proposal

In response to an invitation of the Division of Reactor Development a preliminary research proposal was transmitted to AEC-RL00 (HW-78354) for a Containment Test Program. The Commission's primary interest is in testing components of containment systems for water-cooled power reactors. Their main objective is to improve reactor siting criteria.

Standards Committee

To permit more active conduct of the nuclear engineering standards work at national levels, the following have been completed and distributed.

1. A review of past, present and proposed nuclear engineering standards activities.
2. An up-to-date list of HAPO representatives on such committees.
3. A revision of "Guides for HAPO Employees," first issued in June 1959.

Acting Manager
Programming

M. Lewis:jm

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF JULY 1963

A. ORGANIZATION AND PERSONNEL

Paul L. Sheldon joined the Internal Dosimetry Operation. Janice B. Persons was reactivated into Environmental Studies and Evaluation for a temporary assignment. Nancy F. Kirkland transferred from Environmental Studies and Evaluation to the Irradiation Processing Department. Stanley D. Waggoner transferred from Internal Dosimetry to the Chemical Laboratory.

B. ACTIVITIES

Occupational Exposure Experience

Three new plutonium deposition cases, each estimated to be less than one percent of the maximum permissible body burden (MPBB plutonium with bone as reference is $0.04 \mu\text{c}$), were confirmed by special bioassay analysis during the month. Two of the incidents occurred in the Weapons Manufacturing Building (234-5) and one in the Critical Mass Laboratory (209-E). The total number of individuals who have received internal plutonium deposition at Hanford is 321. With the reactivation of one employee, the number of deposition cases currently employed is 233.

There were eight incidents involving 15 employees that required special bioassay sampling for plutonium analysis this month. The following is a brief description of the more significant incidents.

A Chemical Process Operator at the Weapons Manufacturing Building (234-5) received a plutonium contaminated injury at 11:10 p.m. on July 3, 1963. The injury occurred when a sliver from a wooden handle of a tamping hammer pierced the hood and surgeons glove as he was preparing plutonium metal turnings as feed material for a press in Hood HA-27-DB (briquetting hood). Initial examination with the plutonium wound counter showed $0.134 \mu\text{c}$ plutonium at the site of the injury. After three excisions, approximately $1 \times 10^{-4} \mu\text{c}$ plutonium remained.

Three CPD employees were exposed to high airborne contamination at the Weapons Manufacturing Building (234-5) on July 17, 1963, while performing maintenance work in Room 9A. After completing glove changes in the lower level of the 9A hood, the employees surveyed the immediate area and removed their respiratory protection. A Radiation Monitor started a routine survey of the upper level of 9A and found contamination up to 10,000 d/m. They immediately evacuated the room and called

for a personal survey. The two operators had smearable nasal contamination ranging from 450 d/m to 3,000 d/m. A survey of the room after the incident revealed contamination > 40,000 d/m in an area adjacent to the area where the monitor was surveying. The contamination apparently came from a faulty 36" hood port. The two operators experiencing the nasal contamination were working below this hood port.

An HL employee at the Critical Mass Lab (209-E) received a plutonium nitrate contaminated injury at 3:30 p.m. on July 24, 1963. The employee was removing strip coat from the floor of the mixing hood when apparently an obstruction penetrated the glove causing the injury. The incident was not reported until 8:30 a.m. on July 25, 1963. Examination at the wound counter showed 1.4×10^{-3} μc plutonium at the wound site. After two tissue excisions, the contamination was $< 1 \times 10^{-4}$ μc (detection limit of plutonium wound counter 1×10^{-4} μc). The total number of plutonium contaminated injuries for the year 1963 now stands at 13 with 8 requiring excision. In 1962 there was a total of 15 contaminated injuries with 8 requiring excision.

Also during the month, five employees were involved in incidents that required special evaluation for possible intake of radioactive materials other than plutonium. Only one positive intake was confirmed. This incident occurred during the startup of the 105-KW reactor at 10:30 a.m. on July 12, 1963, when a rod on which a seal was recently replaced failed to function properly. While the employee was attempting to free the rod, gas blew by the seal. The employee left immediately and called for monitoring assistance. A survey revealed 600 c/m on the skin of his neck and cheek and 20,000 c/m on his nose. After nasal irrigation, it was noted that the exhaled air from the employee contained radioactive contamination. Examination at the whole body counter indicated the presence of 0.068 μc Zr-Nb⁹⁵ (= 0.4% MPBB - MPBB Zr-Nb⁹⁵ for total body is 20 μc), and a trace of Tc^{99M}.

Several other incidents of significance which occurred during the month are summarized below.

A CPD employee at Redox (202-S) was inspecting a faulty sample port in the south sample gallery at 3:30 a.m. on July 16, 1963. Upon completion of the work, a band of contamination approximately 6" wide was found around his waist. The contamination measured from 600 mrad to 1,000 mrad with field instruments. Evaluation of the exposure from this contamination assuming a maximum of 3 minutes exposure indicated the employee received a dose of approximately 0.2 rad to a localized skin area around his waist.

A routine survey of the 105-KE gate house at 6:45 a.m. on July 11, 1963, showed 4,000 c/m beta-gamma contamination on the floor and 1,000

c/m on the desk. One of the guards in the gate house had 750-1,000 c/m of contamination on his shoes. The bus carrying the graveyard shift guards from the 100-K Area was surveyed at the bus lot at 8:00 a.m. No contamination was found in the bus or on the guards riding on the bus. RPO was notified at 1:35 p.m. that contamination was found in the 100-KW Area. One of the retention tanks was down for several days to prepare for the repair of the water chute. It appeared the contamination from the basin had become airborne and was spread throughout the area when the retention tank dried out. Contamination up to 10,000 c/m was found in the 1717 garage and contamination ranging from 500-1,000 c/m was found on several employees' shoes. A survey along the road from the 100-K gate house towards Route 1 showed 7 to 10 particles/100 square feet in the 100-K parking lot, diminishing to 3 particles/100 square feet at a distance of 0.1 mile south of the 100-K gate house. Particle concentrations of approximately 1 particle/100 square feet were found along Route 1 for a distance of 1.3 miles west of the central fire station. No contamination was found in roadway surveys which continued north to the 100-N Area, west from 1.3 miles west of the central fire station to approximately 2.5 miles west of the central fire station on Route 1, and south from the central fire station on Route 4N to the 100-B Area cutoff. A survey of the lawn at the central fire station revealed approximately 1.5 particles/100 square feet. The majority of the particles read less than 4,000 c/m with a maximum reading of 35,000 c/m. Roads within the 100-K Area, the 100-K parking lot, and for approximately 0.2 mile south of the 100-K gate house were washed down by the fire department in an effort to reduce the contamination. Sand bags were placed around the drain in the tank and water was added in an effort to prevent further spread of the contamination. Some 250 to 300 employees were surveyed before they left the area.

On July 18, 1963, a specialist from 100-K Area reported to Environmental Monitoring the possibility of contaminated scrap being sent to the excess yard in the 1100 Area as a result of the above incident. Two loads of scrap metal were cleared for transporting to the excess yard on Wednesday, July 10, 1963, the day before the incident occurred. These two loads were transferred to the excess yard on Thursday without being resurveyed. A survey performed by 300 Area Radiation Monitoring personnel on July 18, 1963, revealed contamination up to 10,000 c/m in the general area where the two loads were dumped. It appears that at least 6 loads of scrap will have to be removed from the excess yard before all of the contamination is removed. The contaminated scrap is being sent to 300-N for burial.

A contamination spread occurred in Room 12C following the accidental breakage of an irradiated arsenic ampoule. The incident occurred while a chemist was attempting to wrap the ampoule in a lead sheet preparatory to making a gamma scan. The contents of the ampoule

erupted, blowing solution out of the hood and, consequently, contaminating the employee's personal and protective clothing in addition to the laboratory floor area. No skin contamination was encountered. Nasal smears were negative and the air activity level was less than 1.0×10^{-10} $\mu\text{c As/cc}$. The maximum concentration noted was 1.2 rads per hour on the floor and on the employee's shoes.

Subcritical experiments using PuO_2 -polystyrene cubes on the Remote Split Table Machine were continued. A worker dropped a cube onto the hood floor, severely damaging one corner. The immediate floor area was contaminated to $> 10^6$ d/m; the gloves used while bagging the damaged cube were contaminated to 80,000 d/m; and low-level contamination was found on the Remote Split Table Machine frame. No other contamination spread was found. The contamination was quickly removed or fixed for future decontamination, and work resumed.

Numerous authorized entries were made into A cell of the PRTR during reactor operation at power levels ranging from 1-70 megawatts for inspection and necessary system maintenance. Dose rates ranged from 200 mrem/hour including 100 mr/hour to 7.5 mrem/hour including 4.5 r/hour. Personnel exposure did not exceed 200 mrem for any entry.

A review of the whole body dose from internally deposited tritium oxide for PRTR employees was conducted. Through 28 weeks of calendar year 1963, 18 individuals received a dose from assimilated tritium oxide exceeding 0.5 rem, based on urine analyses. The maximum tritium dose received by an individual for this period was 1.1 rem. The total whole body dose for the employee, a PRTR technician, from assimilated tritium and external penetrating radiation was 2.4 rem. For the 18 employees referred to above, the whole body dose from tritium averages 40 percent of the total whole body dose as compared to an average of 30 percent for all PRTR Operation and Maintenance employees. The total tritium dose for all HAPO employees for the first 28 weeks of CY 1963 was 27 rem as compared with 38 rem for all of CY 1962.

Environmental Experience

Concentrations of radioactive fallout materials in the air of the Pacific Northwest averaged 5 pc β/m^3 during the five-week period ending July 26, 1963, compared to the previous month's average of 7 pc β/m^3 .

Five aerial monitoring flights were made this month. Two flights were in connection with a special iodine test, two were to check Columbia River slough conditions, and one followed standard pattern 1-E to make background measurements and provide training.

The following 263 biological, produce, and food samples were obtained for radiochemical analysis:

Milk	67 samples	209 gallons
Pasture grass	58 samples	116 pounds
Beef thyroids	27 sets	
Oysters	2 samples	4 pounds
Ground round	3 samples	6 pounds
Fresh vegetables & fruit	13 samples	26 pounds
Dried oysters	10 samples	
Fish	83 samples	

During the period July 26 through July 31, 1963, approximately 3.5 millicuries of mixed fission products were discharged from the Radiochemistry Building (325) ventilation stack. Although this small emission is not significant from an environmental contamination standpoint, it represents a measurable increase over the normal rates of less than one millicurie per week. The cause of the emission is not known. All filter samples collected by RMO personnel from ducts and rooms within the building were at normal activity levels throughout the period in question. The continued low-level emission may mean a small leak has developed in one of the ventilation filters.

Studies and Improvements

Tentative plans were formulated whereby essentially all environmental sampling and monitoring data would be handled by electronic data processing. The environmental surveillance program was divided into several major groups, each of which contains related sampling and monitoring data for output reports. These are: river monitoring, farm produce and food-stuffs, waste disposal, and other miscellaneous monitoring programs.

Evaluation of the effects of a KE reactor fuel element failure on the Columbia River showed remarkably close agreement between estimated amounts of fission products released and the amounts of fission products found in the river at the 300 Area. This experience is in contrast to predictions that preferential dissolution of fission products would occur in the reactor. Also, predictions that significant losses might occur in the reactor effluent water complex or in the river bottom did not prove to be valid. This conclusion was based on interpretation of data obtained at the Automatic Columbia River Monitor where both a continuous measurement and a continuous sample were taken. In the near future continuous samplers will be installed in many of the locations where "grab" samples are taken presently, thereby providing more accurate measurement of incremental exposures due to any activity surges.

A cooperative study with other Hanford Laboratories sections on the dispersion of I^{131} from a ground release was begun this month. Approximately 0.1 curie of I^{131} was released on July 25 on the Atmospheric Physics Operation grid. RPO participation in this test included the prerelease predictions of the environmental contamination which would result, the collection of special samples from established atmospheric monitoring facilities (614 Buildings) augmented by additional temporary stations, the collection of I^{131} on charcoal paper strips and exposed surfaces of carbon tetrachloride, the collection of samples of native vegetation both on and off the project, measurement of any dose rate changes with a sensitive 40-liter ionization chamber, and the uptake of I^{131} in the thyroids of two men positioned downwind from the release point.

The N Reactor Department was advised that they could discharge Vertan 690 directly to the Columbia River as planned following decontamination of the secondary cooling system. This recommendation resulted from evaluation of all available information contributed by other Hanford Laboratories sections, from conversations with Industrial Hygiene, and from discussion of the proposed release with representatives of the Washington State Pollution Control Commission.

A listing of all of the routine electronic data processing output for Radiation Protection Operation was made and a folder containing examples of all of the outputs (reports and cards) was assembled. RPO receives a total of 64 reports from Electronic Data Processing Operation. There are 39 reports dealing with external radiation, 13 dealing with internal radiation, 5 dealing with environmental monitoring, and 7 dealing with instruments. This includes reports received weekly, bi-weekly, monthly, quarterly, and yearly.

The design of the mechanical system for the mechanized densitometer including assembly and part layouts is now 80% complete. A mechanical film tray feed hopper was designed, fabricated, and placed into test operation. With this feed system, the mechanized densitometer will have a capacity for 200 pieces of film or about a 15-minute run without operator attention.

The construction of the payroll number parity punch was completed and the instrument is now ready for routine use by External Dosimetry. With this equipment, additional coding will be provided on the lead tape identification component of the Hanford personnel film badge dosimeters. The new mechanized densitometer will be able to audit itself to assure correct reading of each payroll number through the utilization of the parity dots.

A mechanized pocket dosimeter reader was placed into routine use during the past month. The operator dials the payroll number that is typed on the dosimeter label and then places the pocket dosimeter in the reader.

The dose is visually displayed on a digital voltmeter and, along with the payroll number, is punched into electronic data processing cards. This reader represents another in a series of mechanized processing systems that are designed to minimize tedious or costly operations and opportunity for human error in the Hanford dosimetry systems.

A new K-source radiator was assembled for the K-fluorescent X-ray. A mixture of equal parts of samarium and gadolinium were used for the source. The K-fluorescent X-ray beam energy from this radiator was measured with a Nuclear Data 256 channel analyzer to be about 43 Kev. This energy peak agrees with other published data for these radiators. Study is continuing to determine the best filters to further improve the monoenergetic 58, 78, and 100 Kev radiators. It now appears that a lead filter may provide more monoenergetic radiation than that obtained with the currently used tin filters. The tin filters now in use are producing a small energy peak at about 26 Kev from their own K-fluorescent reactions.

Development and testing of the beam pulser for the negative ion Van de Graaff accelerator was completed. The modifications in the Van de Graaff accelerator to provide this mode of operation was suggested by Radiological Physics. Pulse lengths from 15 milliseconds to 22 seconds were produced with a reproducibility of $\pm 5\%$ in most instances. Personnel beta-gamma film badge dosimeters were exposed to pulse doses of 21 r and 1.3 r at dose rates between 2×10^3 r/hour and 5×10^5 r/hour (10 cm from target). The doses were accumulated of time periods ranging from 15 milliseconds to 22 seconds. All film results were within $\pm 10\%$ of the known dose indicating no dose rate or time dependence in the regions studied.

The gamma radiation energy dependence of LiPO_3 (Lithium Metaphosphate) glass at 16.1, 23.7, 52, 62, 82, 100 and 120 Kev was compared with the response to radium-gamma radiation. The relative energy sensitivity of LiPO_3 glass and AgPO_3 glass is given in Table I. The superior energy independence of the LiPO_3 glass was clearly demonstrated.

TABLE I

Glass Dosimeter Energy Dependence

Energy (Kev)	Response Normalized to Ra- λ Radiation	
	<u>LiPO_3 Glass</u>	<u>AgPO_3 Glass</u>
16.1	1.2	7.2
23.7	3.7	9.2
52	9.1	18.8
62	7.3	18.2
82	5.5	14.0
100	3.7	10.0
120	3.1	7.3
Ra- λ	1.0	1.0

The calibration of a neutron spectrometer using a Li^6 detector was carefully checked and the linearity and energy calibration are known to better than $\pm 5\%$. The calibration is accurate on both the 256 and the 512 channel analyzers.

The first attempts to achieve coincidence operation of the He^3 detector neutron spectrometer were not successful due to a noisy electronics system. The proton recoil peak from the $\text{He}^3(\text{n,p})\text{T}$ reaction showed up nicely but the triton peak seemed to be lost in the electronic noise. A bad tube that could possibly have caused excess noise was found. In a further test with high energy neutrons, the proton and triton peaks appeared along with a sum peak (the previous work was done with thermal neutrons only). The He^3 detector is now working properly and has a considerably higher counting efficiency than the Li^6 detector, about a factor of 20 times. The improved counting efficiency is necessary to provide a capability to measure Hanford work location neutron spectra.

Further measurements of the fast neutron dose in the PCTR test core were made at the request of the Biology Laboratory. The measurements show that the gamma-to-neutron dose ratio was decreased by making the exposures inside a lead shielded box. In this particular case, an exposure at a reactor power level of 100 watts for six minutes yielded a fast neutron dose of 185 rads when measured with a silicon diode neutron dosimeter. A measurement in the same environment with a sulfur pellet corrected for the spectrum as measured with the Li^6 detector neutron spectrometer yielded a dose of 191 rads.

The program "Spec" for analysis of the input spectrum data was successfully written and used to determine the least squares fit of the various neutron spectrums measured with the Li^6 and He^3 neutron spectrometers. Through the use of this program, it was indicated that the Calibrations Unit's PuF_4 neutron source has a maximum neutron emission rate at an energy of approximately 0.8 Mev. The energy of this maximum emission rate corresponds to that reported by the U. S. Naval Radiological Defense Laboratory. Neutron spectrum measurements utilizing the Calibrations Unit's PuBe source were not in good agreement with other published spectral data. The Hanford source showed a maximum emission rate at about 1.2 Mev while published values indicate this peak occurs at about 3 Mev. There is no immediate explanation to define the difference observed between these measurements.

A one curie Co^{60} source used by the Calibrations Unit for special dose rate calibrations was inadvertently ruptured in the 3745A Building on July 29. Contrary to expectations when the source container was broken, the source material was not a solid rod of material but was a silvery white metallic powder. The employee in the building at the time immediately reported the occurrence, shut down the air conditioning and sealed

the building. No spread of contamination to the surrounding areas or even within their radiation room occurred (a total of about 10 square inches of concrete floor was contaminated). Air samples taken in the room immediately after the spill did not show abnormal airborne contamination. The powdery Co^{60} material was collected initially by the use of strong permanent magnets. Final cleaning was by the use of masking tape and chipping the top surface from the concrete floor. No personnel contamination or abnormal exposure occurred during the event or clean up. A major point of interest and concern was the fact that the Co^{60} material within the capsule was entirely powdered and presented a significant contamination hazard at the time of the rupture. It appears that many Co^{60} users may have built up an unfounded confidence in the use of Co^{60} radiation sources, believing the radioactive material to be of a solid integral metal and incapable of being spread easily if a source rupture should occur. Originally, this Co^{60} was an integral mass of cobalt. It appears, however, that cobalt sources may crystalize and powder from internal radiation damage to present serious contamination hazards if source ruptures should occur. One should handle all Co^{60} sources with the same care as that afforded to radium sources. One should not be lulled into a sense of security by anticipating that the Co^{60} source he is using consists of a metallic block of cobalt and that dropping or rupturing the source will not result in a serious and costly gross spread of contamination.

A total of two Emergency Monitoring Kits were serviced during the month. A total of 105 instruments received an audit calibration as part of the Calibrations Unit quality control program.

Six sections of the Manual of Emergency Plans and Procedures were updated to reflect current call lists and phone numbers.

Research Studies

Effect of Reactor Effluent on Quality of Columbia River Water (02)

A study of the effects of reactor effluent on river water quality with emphasis on temperature is in progress. Temperature and gamma dose radiation measurements continued between Hanford and Pasco. This phase of the study is designed to correlate shoreline and mean river temperatures at fixed monitoring points, plus search for locations at which a single-point measurement will reliably give a mean river temperature. Equipment for dye studies of time of travel and effluent dispersal patterns was nearly complete, but delivery of portable meteorological instruments for heat budget studies lagged.

Shoreline-to-river cross section mean temperature ratios increased with higher air temperatures and decreasing river flows. An initial temperature traverse of the shoreline of McNary pool between Richland and the 300 Area

was made with the new thermistor instrument. The localized effect of back-water areas was quite marked.

Mechanisms of Environmental Exposure (02)

The routine collection of new data continued in four general areas:

a. The contract with the Washington State Department of Game for collection of information of fish consumption was renewed for FY 1964. No additional returns were received during July, however.

b. Approximately 70 dietary questionnaires completed by persons who took whole body counts were received from Internal Dosimetry. Time was not available to begin an evaluation of results obtained from this survey.

c. Cooperative studies with Internal Dosimetry and the Biology Laboratory on the transfer of I^{131} from cow's milk to human thyroids continued with the counting of thyroids of persons participating in the last scheduled test of this series. This test began in May 1963 and involved the ingestion of milk from cows which received stable iodine in addition to tracer amounts of I^{131} . By the end of July, the I^{131} content of the human thyroids had essentially returned to background.

d. Ingestion of Columbia River fish by one individual in order to determine the associated body burden of Zn^{65} was discontinued in April. Whole body counting was continued, however, in order to measure the rate of loss of the Zn^{65} from the body.

C. RELATIONS

Two suggestions were submitted by personnel of the Radiation Protection Operation during the month. No suggestions were adopted and none were rejected. Three suggestions are pending evaluation.

Safety meetings were held throughout the Section during the month. Films on "Outboard Outings" and "Radiation in Perspective" were shown. A bulletin entitled "Das Safen und Sounden Calendar" was prepared by D. McConnon and distributed to all Radiation Monitoring personnel.

Radiation protection orientation lectures were presented to personnel of CPD Maintenance, Metallurgy Development, Electrical Utility, Metal Fabrication Development and Fuel Design, Reactor Metals Research, Physical Metallurgy, Radiometallurgy Laboratories, Reactor Lattice Physics, Power, Plutonium Metallurgy, and PRTR. A two-hour talk on timekeeping responsibilities for entry to a high-level radiation zone to effect rescue was presented to 8 members of the 300 Area Emergency Rescue Squad. A thirty-minute

talk on radiation protection practices was presented to approximately 40 technical graduates. This same talk will soon be presented monthly for all persons being assigned to work in the 300 Area. Also, a one-hour talk on 300 Area Emergency Plans was presented to 30 members of the Radiological Analysis Operation and several of the Biophysics Building (329) Fire Wardens.

A new monthly combined Safety and Radiation Protection orientation for new-hires or transfers was given to approximately 110 technical graduates.

An orientation lecture on the design and performance of the current Hanford personnel beta-gamma film badge dosimeters was presented to the Maintenance and Equipment Engineering group at the PRTR.

D. SIGNIFICANT REPORTS

HW-76525-6 - "Radiological Status of the Hanford Environs for June 1963"
by R. F. Foster.

HW-77533 - - "Evaluation of Radiological Conditions in the Vicinity of
Hanford, January-March 1963" by R. F. Foster and Staff.

HW-78481 - - "Monthly Report - July 1963, Radiation Monitoring Operation"
by A. J. Stevens.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDSExternal Exposure Above Permissible Limits

	<u>July</u>	<u>1963</u>
Whole Body Penetrating	0	1
Whole Body Skin	0	0
Extremity	0	0

Hanford Pocket Dosimeters

Dosimeters Processed	6,138	45,215
----------------------	-------	--------

Hanford Beta-Gamma Film Badge Dosimeters

Film Processed	9,746	66,464
Results - 100-300 mrad	135	1,154
- 300-500 mrad	12	113
- Over 500 mrad	1	24
Lost Results	18	156
Average Dose Per Film Packet - mrad (ow)	6.5	6.8
- mr (s)	31.8	31.8

Hanford Neutron Film Badge DosimeterSlow Neutron

Film Processed	1,598	11,746
Results - 50-100 mrem	3	13
- 100-300 mrem	0	2
- Over 300 mrem	0	0
Lost Results	6	82

Fast Neutron

Film Read	536	3,158
Results - 50-100 mrem	23	232
- 100-300 mrem	0	388
- Over 300 mrem	0	4
Lost Results	7	62

Hand Checks

Checks Taken - Alpha	40,636	260,473
- Beta-Gamma	57,019	411,322

Skin Contamination

Plutonium	25	168
Fission Products	36	275
Uranium	0	4
Tritium	0	0

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Whole Body Counter

<u>Subject</u>	<u>Number of Examinations</u>			
	<u>747-A WBC</u>	<u>1963</u>	<u>Mobile WBC</u>	<u>1963</u>
GE Employees				
Regular	58	443		30
Incident Cases	13	110		
Terminations	16	71		
New Hires	93	357		
Special Studies	84	392		
Non-Employees				
Children	7	12		
Visitors	8	29		
Environmental Studies	<u>4</u>	<u>13</u>		
	283	1,427		

Bioassay

	<u>Current Reporting Limit</u>		<u>Results Above Reporting Limit</u>		<u>Samples Assayed</u>	
			<u>July</u>	<u>1963</u>	<u>July</u>	<u>1963</u>
Plutonium	2.2×10^{-8}	µc/sample	101	668	303	4,252
Fission Products	3.1×10^{-5}	µc/sample	6	68	217	3,840
Strontium	3.1×10^{-5}	µc/sample	0	45	0	45
Tritium	5.0	µc/l	93	960	151	1,639
Uranium	0.14	µgm/l			159	1,128
Special Studies			0	0	12	311

Calibrations

	<u>Number of Units Calibrated</u>	
	<u>July</u>	<u>1963</u>
Portable Instruments		
CP Meter	1,056	7,410
Juno	246	1,848
GM	562	3,941
Other	154	1,304
Audits	<u>105</u>	<u>741</u>
	2,123	15,244
Personnel Meters		
Badge Film	1,140	5,900
Pencils	90	810
Other	<u>372</u>	<u>2,198</u>
	1,602	8,908

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	<u>Number of Units Calibrated</u>	
	<u>July</u>	<u>1963</u>
Miscellaneous Special Services	300	11,497
Total Number of Calibrations	4,025	35,649

Manager
RADIATION PROTECTION

AR Keene:CMU:10

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FINANCE AND ADMINISTRATIONACCOUNTINGCost Accounting

Financial Plan #1 for FY 1964 was received from RL00-AEC during the month, but does not become effective until the FY 1964 Congressional Appropriation Act is passed. During the interim period, Hanford is authorized to operate under the "Continuing Resolution" passed by Congress on July 1, 1963. Following is a summary of the research and development funds contained in the Financial Plan as compared to the FY 1964 Revised Budget submission:

(Dollars in thousands)			
<u>Program</u>	<u>Financial Plan #1</u>	<u>Budget Request</u>	<u>Increase (Decrease)</u>
02 (HAPO)	\$10 280	\$10 280	\$ --
03 (HAPO)	1 650	1 650	--
04 (HL)	13 485	14 025	(540)
05 (HL)	1 456	1 765	(309)
06 (HL)	3 604	3 780	(176)
08 (HL)	100-a)	500	(400)
Total	<u>\$30 575</u>	<u>\$32 000</u>	<u>\$(1 425)</u>

- (a- Subsequent to the issuance of Financial Plan #1, the Washington AEC Director of the Division of Isotopes Development authorized Hanford Laboratories, by letter, to operate at a level of \$240,000 which will permit initiation of the promethium fuel study.

Pending passage of the FY 1964 Congressional Appropriation Act, HAPO has been authorized \$1,130,000 for new equipment obligations and \$105,000 for general plant work orders for the months of July and August. Of these amounts, Hanford Laboratories has been allocated \$565,000 and \$40,000, respectively. The capital equipment allocation is detailed by program as follows:

02	\$100 000
03	40 000
04	175 000
05	100 000
06	125 000
08	25 000
Total	<u>\$565 000</u>

The Commission has indicated the total FY 1964 authorization may not be granted by September 1, 1963. Therefore, a request for equipment obligational authority for the month of September 1963 was forwarded to Contract Accounting in response to a request from RL00-AEC. Additional program amounts requested by Hanford Laboratories for the month of September are as follows:

02	\$200 000
03	20 000
04	300 000
05	100 000
06	100 000
08	<u>25 000</u>
Total	<u>\$745 000</u>

Activities for which special accounting codes were established are described below. Included therein is an authorization for \$400,000 to undertake research and development studies for the Space Nuclear Propulsion Office (SNPO) in four program areas.

- .3N SNPO - Basic Cermet Studies - Internal HL budget of \$100,000.
- .3P SNPO - Cermet Fuel Fabrication - Internal HL budget of \$100,000.
- .3R SNPO - Cermet Joining Studies - Internal HL budget of \$100,000.
- .3T SNPO - Honeycomb Fuel Development - Internal HL budget of \$100,000.
- .8C Packaging and shipping of contaminated materials to Turco Products, Inc. - Bill actual cost of packaging and shipping. Arrangement in connection with the testing of decontaminating agents on contract SA-294.
- .8D Packaging and shipping of contaminated materials to Wyandotte Chemicals Corporation - Bill actual cost of packaging and shipping. Arrangement in connection with the testing of decontaminating agents on contract SA-293.
- .8E Packaging and shipping of contaminated materials to Du Bois Chemical Company - Bill actual cost of packaging and shipping. Arrangements in connection with the testing of decontaminating agents on contract SA-292.

The following program code changes became effective July 1, 1963:

Codes Established

- .17 Specific Fuel Cycle Analysis (O4 Program)
- .21 Rupture Loop Operation (O4 Program)

Codes Canceled

- .34 DR-1 Loop Operation (O4 Program)
- .57 Experimental Gas Cooled Reactor (O4 Program)
- .99 Euratom (O4 Program)

The summary booklet of Hanford Laboratories' Budget for FY 1965 and Revision of Budget for FY 1964 was published during the month. The booklet presents an abstract of each research and development program proposal as well as summary personnel and financial data.

General Accounting

Following is the status of letters requesting AEC concurrence in proposed actions:

AT-295	Professional Research and Teaching Leave - K. R. Merckx	Approved 7-10-63*
AT-298	Participation in Standardizing Activities - National Fire Protection Association - R. H. Scott	Approved 7-10-63
AT-300	Participation in Standardizing Activities - ASTM Committee B-3 - Corrosion of Non- Ferrous Metals - R. L. Dillon	Approved 7-9-63
AT-304	Participation in Standardizing Activities - ASTM Committee C-5 - Manufactured Carbon and Graphite Products - E. M. Woodruff	In hands of AEC

*Approved with certain stipulations regarding period of obligated service and possible transfer liabilities.

The following revised OPGs were issued during July:

<u>OPG No.</u>	<u>Title</u>
1.3	Consolidated Reports to AEC
7.8 (pp. 19, 19a, 19b, 20, 41 & 42)	Control of Documents Classified Secret and Confidential

<u>OPG No.</u>	<u>Title</u>
2.3.6	Relations and Occupational Health Operation Manager Position Guide
1.5	Issuance of Official Correspondence and Documents
1.9	Records Management Program
22.3.1 (pp. 3 & 9)	Approval Authorizations
7.5	Reporting Violations of Federal or AEC Security Regulations
1.7	Production Forecasts and Schedules
2.1	Organization Charts and Directories

Total investment in Plant and Equipment In Service for Hanford Laboratories at June 30, 1963 was \$75.2 million, compared with \$69.6 million at June 30, 1962, and \$31.2 million at September 30, 1956. The net increase of \$5.6 million during FY 1963 includes additions from transfers from Work in Progress (\$2.8 million), transfers from the AEC (\$2.9 million), transfer of equipment from FPD (\$.5 million), and other miscellaneous items valued at \$.6 million. Included in the transfers from Work In Progress and AEC are projects High Level Utility Cell, 327 Building; Biology Laboratory Improvement, 108-F Building; Shielded Analytical Laboratory, 325 Building; Fuel Element Rupture Test Facility, 309 Building; and other projects of lesser magnitude. The major offsetting reductions during FY 1963 included retirements (excess and PDR) of \$.5 million and transfers to other HAPO components of \$.2 million.

Following is a summary of Plant and Equipment In Service at June 30, 1963, by major category:

	(In thousands)
Building and Structures	\$ 30 062
Equipment	45 078
Utilities	22
Improvements to Land	50
	<u>\$ 75 212</u>

Total Hanford Laboratories investment in plant and equipment by Area is tabulated below:

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	(In thousands)
100-B	\$ 103.4
100-D	2 132.4
100-F	3 850.7
100-H	18.3
100-K	923.0
100-N	.6
200-E	1 472.1
200-W	5 277.4
300	58 006.2 -1)
600	1 022.5
700	451.0
1100	17.7
White Bluffs	616.3
Off-Site	1 320.0
Total	<u>\$ 75 211.6</u>

(1- Includes such major facilities, each worth more than \$1 million,
as:

	(In millions)
306 Building	\$5.3
308 Building	5.0
309 Building	15.3
314 Building	1.4
321 Building	2.4
325 Building	9.5
326 Building	4.2
327 Building	3.4
328 Building	1.6
329 Building	2.2

Hanford Laboratories' investment in materials at July 1, 1963 was:

	(In thousands)
SS Material	\$23 130
Reactor and Other	
Special Materials	929
Spare Parts	<u>328 -1)</u>
Total	<u>\$24 387</u>

(1- Includes a reserve of \$81,267.

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The value of nuclear material consumed in research by Hanford Laboratories during FY 1963 was \$3.6 million comprised as follows:

02 Program	\$1 008
03 Program	1 056
04 Program	<u>1 576</u>
Total	<u>\$3 640</u>

The quarterly inventory of Other Special Materials in the custody of 109 holders on June 30, 1963 disclosed a net shortage of 6 grams of platinum, due to weighing variances, and a net increase of 123 grams of gallium. The major portion of the gallium variance resulted from the unreported receipt of 122 grams (\$131.76) from CPD's Essential Material Inventory.

RLOO-AEC was requested to arrange for the transfer to Hanford Laboratories of UCLRL equipment valued at \$69,839 acquired to support University of California Lawrence Radiation Laboratory research and development programs at Hanford.

During the month, Project CAH-867 - Fuel Element Rupture Test Facility (PRTR), valued at \$1,498,020, was unitized and a report issued.

The physical inventory of Spare Parts by C&AO indicated a total statistical deficit of \$4,584 for Hanford Laboratories' holdings.

The heavy water inventory at the end of July 1963 showed a loss of 544 pounds valued at \$7,546 for the PRTR and a loss of 21 pounds valued at \$288 for the PRCF. Heavy water scrap generated during the month amounted to 1,199 pounds resulting in a \$2,621 charge to operating cost. Heavy water accumulated at July 31, for return to SROO amounted to 10,867 pounds valued at \$135,648. Fifty-five drums, 27,445 pounds, of heavy water were received July 29, 1963; the charge will be booked in August.

The following is a summary of Heavy Water and Zirconium activity for FY 1963:

	<u>Heavy Water</u>	<u>Zirconium</u>
Beginning Balance (7-1-62)	\$ 303 148	\$ 222 267
Receipts	858 434	110 962
Transfers from Spare Parts		143 731
Inventory Adjustment	(5)	
Shipments to SROO	(487 782)	
Charges to Cost	(273 196)	(228 952)
Sponge Cr. from Pittsburgh Naval Office		(461)
Transfers to other HAPC Components		(81)
Ending Balance (6-30-63)	<u>\$ 400 599</u>	<u>\$ 247 466</u>

A review of the depreciation rate for capitalized heavy water (PRTR) was made during the month. The rate .0120 appears to be adequate, at least for the present. The net value of capitalized heavy water is \$13.44 a pound at June 30, 1963, compared to \$13.88 average inventory price.

Laboratory Storage Pool activity for the month of July 1963, is summarized below:

	<u>Quantity</u>	<u>Value</u>
Beginning Balance	1 480	\$811 520
Items Received	286	90 021
Items Reclaimed by Custodians	(17)	(12 987)
Equipment Transfers	(18)	(8 547)
Items Disposed of by PIR	(34)	(868)
Items Disposed of by Excess	<u>(14)</u>	<u>(62 324)</u>
Ending Balance	<u>1 683</u>	<u>\$816 815 -1)</u>

(1- Includes 136 items valued at \$113,584 on loan at July 31, 1963.

During the month, 23 items valued at \$9,924 were loaned and/or transferred in lieu of purchases.

Total investment of equipment and material in custody of the Laboratory Storage Pool at July 31, 1963, is \$1.6 million including Reactor and Other Special Materials (\$299,212), SS Material (\$175,500) and other materials (\$354,866).

Assistance is being provided in the cleanup of the 325 Building basement. Material of \$221,502 and equipment valued at \$58,141 were transferred to the Laboratory Storage Pool during July.

The following contracts were processed during the month:

CA-396	M. E. Ensminger
CA-401	John E. Dorn
CA-404	George W. Watt
DDR-162	Pressure Products Industries, Inc.
DDR-167	General Dynamics Corp.
DDR-170	Wah Chang Corp.
SA-259 - Suppl. 1	Schwarzkopf Microanalytical Laboratories
SA-269 - Suppl. 1	Geophysical Service, Inc.
SA-292	Du Bois Chemicals Corp.
SA-293	Wyandotte Chemicals Corp.
SA-294	Turco Products, Inc.

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Hanford Laboratories' allocation for Assistance to Hanford for FY 1964 is \$30,500. The following authorizations have been issued:

ATH-HL-1-64	Advanced Technology Laboratories (Standards Certification and Instrument Design)	\$1 500
ATH-HL-2-64	Service Shop Department (Instrument and Meter Repair)	500
ATH-HL-3-64	Research Operations Department (Electron Microscopy Consultation)	1 500
ATH-HL-4-64	Electrical and Engineering Information Laboratory (Consultation Services - Mr. T. T. Slot)	5 000
ATH-HL-5-64	Marketing Section - APED (Analytical Services)	2 000

Personnel Accounting

An inventory of college graduates engaged in research and development activities was made for the Los Alamos Annual Survey of Professional Scientific Salaries. Data submitted showing the number of employees with college degrees (B.S., M.S., or Ph.D.) distributed by discipline (first degree) and by assignment category (supervisory or non-supervisory) are given below:

	<u>1963</u>	<u>1962</u>	<u>1961</u>	<u>1960</u>	<u>1959</u>	<u>1958</u>	<u>1957</u>
B.S. - M.S. Non-Supervisory	425	392	386	410	401	261	264
Ph.D. Non-Supervisory	88	79	66	58	59	69	61
B.S. - M.S. Supervisory	37	34	30	30	34	26	37
Ph.D. Supervisory	38	36	37	37	46	36	36
	<u>588</u>	<u>541</u>	<u>519</u>	<u>535</u>	<u>540</u>	<u>392</u>	<u>398</u>

The following Laboratories' employees died during the month:

7-17-63	R. W. Erickson	Summer Technician	Biology Laboratory
7-25-63	B. E. Page	Mgr., PRTR Maintenance	Test Reactor & Aux.
7-30-63	C. B. Sams	Spec., Scheduling & Methods	Test Reactor & Aux.

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Patent Awards were made to the following employees:

Jerry J. Cadwell	HWIR 1550	Quick Disconnects for Tubing
Lane A. Bray	" 1588	Use of Sugar to Neutralize Nitric Acid Waste Liquors
Earl C. Martin	" 1588	Use of Sugar to Neutralize Nitric Acid Waste Liquors

Personnel statistics follow:

<u>Number of Hanford Laboratories Employees</u> <u>Changes During Month</u>	<u>Total</u>	<u>Exempt</u>	<u>Nonexempt</u>
Employees on payroll at beginning of month	1 756	764	992
Additions and transfers in	76	51	25
Removals and transfers out	27	18	9
Employees on payroll at end of month	<u>1 805</u>	<u>797</u>	<u>1 008</u>

<u>Overtime Payments During Month</u>	<u>July</u>	<u>June</u>
Exempt	\$ 11 562	\$ 6 193
Nonexempt	38 376	67 139
Total	<u>\$ 49 938</u>	<u>\$ 73 332</u>

<u>Gross Payroll Paid During Month</u>		
Exempt	\$ 777 557	\$ 697 172
Nonexempt	545 042	729 740
Total	<u>\$1 322 599</u>	<u>\$1 426 912</u>

<u>Participation in Employee Benefit</u> <u>Plans at Month End</u>	<u>July</u>		<u>June</u>	
	<u>Number</u>	<u>Percent</u>	<u>Number</u>	<u>Percent</u>
Pension	1 557	98.0	1 520	99.3
Insurance Plan - Personal	426		420	
- Dependent	1 323	99.9	1 277	99.8
U. S. Savings Bonds				
Stock Bonus Plan	158	41.7	158	41.5
Savings Plan	69	3.8	66	3.7
Savings and Security Plan	1 225	85.9	1 165	84.7
Good Neighbor Fund	1 228	71.4	1 249	71.1

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<u>Insurance Claims</u>	<u>July</u>		<u>June</u>	
<u>Employee Benefits</u>	<u>Number</u>	<u>Amount</u>	<u>Number</u>	<u>Amount</u>
Life Insurance	0	\$ -0-	1	\$15 370
Weekly Sickness and Accident	9	594	8	631
Comprehensive Medical	37	2 769	41	2 876
<u>Dependent Benefits</u>				
Comprehensive Medical	82	6 983	90	7 539
Total	<u>128</u>	<u>\$10 346</u>	<u>140</u>	<u>\$26 416</u>

TECHNICAL ADMINISTRATIONEmployee Relations

Thirty nonexempt employment requisitions were filled; 19 remain to be filled.

Suggestion plan activity included 50 submissions, 11 adoptions and 18 rejections.

Information and PresentationsVisitors Center activity:

July attendance	3 310
Average attendance per day open	107
Cumulative attendance since 6-13-62	51 090
Conducted groups	30 (totaling 21 people)

Plant tour activity:

	<u>Number</u>	<u>Total People</u>
General public relations tours	2	18
Special tours	7	57

Documented information flow during the month was comprised of 1,874 titles (14,928 copies) received at Hanford and 96 titles (9,557 copies) sent off-site.

Professional Placement

Advanced Degree - One Ph.D. applicant visited HAPO for an employment interview. Two offers were extended; one acceptance and three rejections were received. Nine offers are currently open.

BS/MS (Direct Placement) - Nine offers were extended. Five acceptances and six rejections were received. One offer is currently open.

BS/MS (Program) - No offers were extended. One acceptance and three rejections were received. There are no current open offers.

Technical Graduate Program - Seven Technical Graduates were placed on permanent assignment. Twenty-three new members were added to the roll and one member terminated. Current program members total 75.

FACILITIES ENGINEERING

Projects

At month end Facilities Engineering Operation was responsible for 10 active projects having total authorized funds in the amount of \$6,735,000. The total estimated cost of these projects is \$10,640,000. Expenditures through June 30, 1963 were \$836,000.

The following summarizes project activity in July:

Number of authorized projects at month end -----	10
Number of new projects authorized -----	0
Number of projects completed -----	0
New projects submitted to the AEC -----	0
Projects awaiting AEC approval -----	1
CAH-106, FRTR Storage Basin and Experimental Facilities Modifications	
Project proposals nearing completion -----	2
FRTR Decontamination and D ₂ O Cleanup Heat Transfer Apparatus for Model Studies	

The current status of projects authorized or awaiting approval is:

CAH-916 - Fuels Recycle Pilot Plant - Directive number AEC-187 Mod. 9, dated July 26, 1963, increased the authorized funds to \$5,660,000 plus \$100,000 worth of transferred capital property. Construction is 2 percent complete and on schedule. Approximately 160 cubic yards of concrete have been placed in the substructure of the wet storage basin, waste tank vaults and metallurgical cell. Excavation, forming and piping work is continuing.

CAH-922 - Burst Test Facility for Irradiated Zirconium Tubes - Construction is 34 percent complete compared to a scheduled 36 percent. Backfill has been placed inside and outside the building walls. The concrete sump, heat exchanger pit and walls for the building steps have been poured.

CAH-958 - Plutonium Fuels Testing and Evaluation Laboratories - 308 Building - Construction is 35 percent complete compared to a scheduled 37 percent. Plastering work is essentially complete and electrical work is progressing. Wall erection has stopped because of material shortages resulting from a railroad routing error.

CAH-962 - Low Level Radiochemistry Laboratory - Detailed design by the Architect-Engineer is 65 percent complete and on schedule. A review of part of the design was made during July.

CAH-977 - Facilities for Radioactive Inhalation Studies - Detailed design by the Architect-Engineer is 18 percent complete and on schedule.

CAH-982 - Addition to Radionuclide Facilities - 141-C Building - The design has been contracted between the AEC and Carson and Kesterson, Architect-Engineer. No schedule has been established.

CGH-992 - Additional Fuel Loading Equipment - 308 Building - Construction is 80 percent complete as compared to a scheduled 90 percent. J. A. Jones Construction Company achieved a planned partial completion by July 31, 1963. Three glove boxes have been made available to General Electric Company plant forces for final startup.

CAH-995 - 309 Building Air Conditioning Modifications - Construction is 50 percent complete compared to a scheduled 80 percent. The 60 ton chiller was delivered July 31, 1963. Some shipping damage was noted but has not been appraised. Completion of this project by the directive completion date is questionable.

CGH-999 - Plutonium Recycle Critical Facility Conversion to Light Water - Detailed design is 100 percent complete. An interim authorization has been given to J. A. Jones for start of fabrication and construction. Some field work has been started. Fabrication of control rods and other components has also started.

CAH-100 - High Temperature Lattice Test Reactor - Vitro Engineering is proceeding with preliminary (Title I) design and has prepared a schedule showing Title I complete by September 30, 1963.

CAH-106 - Plutonium Recycle Test Reactor Storage Basin and Experimental Facilities Modifications - This project was reviewed and approval deferred by the Commission.

Pressure Systems

Engineering review and surveillance of pressure systems continued. Inspection of the PRTR main loop heat exchanger was conducted with the Third Party Inspector and the inspector witnessed the pressure test of the C-1 Loop. Specific items of work during the month included: (1) review of a

pressure vessel design for the tube burst facility, (2) survey and recommendations on 325 Building gas piping, (3) placement of orders for 300 Area propane system equipment to achieve code conformance except safety valves and (4) analysis of safety valve requirements and commercially available equipment for 300 Area propane system.

Engineering Services

Engineering work on active projects included (1) completion of design change to CAH-916, Fuels Recycle Pilot Plant, to provide a basement under the cold canyon, (2) review of A-E drawings on CAH-962, Low Level Radiochemistry Building and (3) completion of design for CGH-999, PRCF Conversion to Light Water.

Plant Engineering

Plant engineering work included: (1) design scope on 314 Building addition for high temperature gas loop, (2) preliminary engineering on electrical requirements for 314 Building HTLTR mock-up and testing, (3) estimating sketches for lighting improvements to 328 Building offices, (4) preliminary engineering on electrical and lighting modifications to 3705 Building, (5) preliminary engineering on requirements for revisions to aerial distribution system in 100 Area Biology Facilities, (6) preparation of purchase specification for replacement of electrical distribution panels for 325 and 328 Buildings, (7) engineering of modifications to 306 Building process sewer, (8) sketches and estimates for modifications to balance rooms in 1706-KER laboratories, (9) preparation of purchase specifications for 325 Building emergency generator, (10) study of filter methods in the sub-micron range, (11) design scope on 326 Building basement modifications, (12) preliminary design studies for a fuel element loader indexing mechanism, (13) preliminary sketches of a PRTR viewing center, (14) study of electrical components in vibration compactor equipment, and (15) field testing, modification and purchase of components for PRTR alarm and paging system.

Facilities Operation

Landlord costs for the month of June were \$166,223 which is 104 percent of the forecast for the month. Total costs for the fiscal year were \$2,057,983, which is 102 percent of the predicted.

The monthly audible criticality alarm checks were completed July 30, 1963. All building systems alarmed properly; however, there are small items for repair or correction being located as a result of the monthly checks. Corrections are being made the day following the tests.

The space survey of laboratories and offices was completed and a summary report issued.

The transformer on the normal circuit at 306 Building was overhauled by Electrical Distribution during a scheduled Saturday outage.

The #3 filtered water pump at 315 Building experienced bearing trouble which delayed completion of installation. It is still awaiting replacement bearings.

Corrective action is being taken to free the swing of the backflow dampers in the 308 Building.

The following tabulation summarizes waste disposal operations:

<u>Item</u>	<u>May</u>	<u>June</u>
Concrete waste barrels disposed to 300-N burial ground	20	8
Loadluggers of dry waste disposed to 300-N burial ground from the 325 Bldg.	4	3
Loadluggers of dry waste disposed to 300-N burial ground from other 300 Area sites	25	19
Crib waste volume (gallons)	300 000	270 000

One of the 307 retention basins exceeded the activity level of 5×10^{-6} μC β/ml .

Four thousand five hundred gallons of liquid waste from 321 Building were disposed in the 216 BC trench. To date 11,500 gallons have been emptied into this 270,000 gallon capacity trench.

The number of components using the vertical burial tubes has increased; consequently, Waste Disposal has assumed the responsibility for the installation and maintenance of the tubes at the burial ground.

The AEC has requested that 300-N be closed as soon as possible. Work is under way to excavate a new trench and vertical burial tubes at Wye burial ground.

Twenty-seven (325-A) casks were decontaminated using the wash tank in room 530-A. About 22 man-hours were saved. The remote handling phase of the work is being introduced, but the material handling equipment needs improvement.

Drafting

The equivalent of 160 drawings were produced during the month for an average of 26 man-hours per drawing. Major jobs in progress are: (1) PRTR as-builts, (2) inhalation studies hood as-builts, (3) salt cycle experimental laboratory for "C" cell, 325 Building, (4) scope drawings, fast supercritical power reactor, (5) chemical decontamination system for D₂O, (6) high temperature gas loop, (7) HTLTR mock-up core and (8) miscellaneous drawings for Technical Shops purchase of equipment parts.

Drafting work was assigned to Vitro Engineering Company estimated at 364 man-hours; 128 man-hours of work was completed by Vitro. Jobs assigned included (1) PRTR Gas Loop revisions, (2) shielded containers, (3) PRTR as-builts, (4) electrolytic machine, (5) PRTR Test Loop, (6) burst test vessel and (7) fuel element assembly jig.

Construction Supervision

Activity during the month on construction work (J. A. Jones Company) being performed for Hanford Laboratories components is given below:

	<u>Unexpended Balance</u>	<u>Waste Calcination (Job 7005)</u>
Orders outstanding beginning of month	\$128 386	\$23 644
Issued during the month (inc. suppl. and adj.)	148 735	74 932
J. A. Jones expenditures during month (includes C.O. costs)	91 619	26 561
Balance at month's end	185 502	72 015
Orders closed during month	217 660	

In addition, work on two maintenance work orders issued to plant forces and having a face value of \$1,435 was supervised.

Major active nonproject jobs in progress, by building, are: (1) 108-F, installation of furniture and hoods in room 405, (2) 108-F, installation of pre-cooling system for rooms 201-202, (3) 141-C, installation of X-ray equipment, (4) 141-C, fabrication and installation of feeder stalls, (5) 231-Z, installation of tensile test glove box, (6) 241-CR, fabrication and installation of steel framing, (7) 646, repair of dams at Rattlesnake Springs, (8) 306, installation of service to arc melt furnace, (9) 308, replacement of steam and condensate lines, (10) 309, installation of shielding, (11) 309, fabrication and installation of filters in fan pit, (12) 309, electrical and mechanical modifications of gas loop, (13) 314, installation of PRTR tube removal mock-up equipment, (14) 314, construction

of gas bottle facility, (15) 325, modification of 20-ton shielding door, (16) 325, installation of heating and ventilation system on mezzanine, (17) 325, modification of rooms 400 and 406, (18) 325, construction of roof enclosure, (19) 327, fabrication and installation of machine exhaust system, (20) 327, installation of floor drains in canyon, (21) 327, fabrication and installation of stairway and doors, (22) 329, modification of laboratory 11-B, (23) 3702, exterior renovation, (24) 300, Wye Burial Ground - construction of burial trench and a 40 tube burial area, (25) 747-A, construction of concrete pad for mobile whole body counter, (26) waste calcination demonstration prototype.

Ten requisitions totaling \$13,435 were issued during the month.

Total value of equipment being procured is \$85,000. M & E lists were issued for Projects CGH-922 and CGH-992.

Manager
Finance and Administration

W Sale:JVM:whm

O4 PROGRAM - REACTOR DEVELOPMENTPLUTONIUM RECYCLE PROGRAMPlutonium Recycle Test ReactorOperation

Reactor output for July was 929 MWD, for an experimental time efficiency of 51% and a plant efficiency of 41.6%. There were nine operating periods during the month, of which five were terminated manually and four were terminated by scrams. A summary of the fuel irradiation program as of July 31, 1963, follows:

	<u>Al-Pu</u>		<u>UO₂</u>		<u>PuO₂-UO₂</u>		<u>Other</u>		<u>Program Totals</u>	
	<u>No.</u>	<u>MWD</u>	<u>No.</u>	<u>MWD</u>	<u>No.</u>	<u>MWD</u>	<u>No.</u>	<u>MWD</u>	<u>No.</u>	<u>MWD</u>
In-Core	15	1106.9	10	1445.0	60	3865.9			85	6396.8
Maximum		114.3		192.7		103.3				
Average		73.8		144.5		64.1				
In-Basin	28	1850.7	56	4047.1	13	296.3	1	7.3	98	622.4
Chem. Proces.	32	2309.3	1	78.0	—	—	—	—	33	2387.3
Program Totals	75	5266.9	67	5570.1	73	4162.2	1	7.3	216	15006.5

Note: (MWD/Element) X 20 = MWD/TU for UO₂ and PuO₂-UO₂.

The first three 1 w/o PuO₂ in UO₂ elements, charged late in the month, began leaking fission gases immediately when reactor operation was begun, and were discharged.

D₂O and indicated helium losses for July were 554 pounds and 144,681 scf, respectively.

Equipment Experience

A total of 136 reactor outage hours were charged to repair work. Of this amount, 68 hours were for E-7 valve replacement, 40 were used for helium system repairs and nine hours were for changeout of breakers in MCC-1E.

Preventive maintenance required 284 hours or 5% of the total maintenance effort.

Improvement Work Status (significant items)Work Completed

Recorder Ammeter Installation for Primary Pumps
E-7 Replacement and Air Supply Adjustment
Refrigerative Cooling of Stack Pit
Water Softener Controls
Revision of PRTR Exhaust Fan to Emergency Power
Explosion Proof Receptacle Outlet

Work Partially Completed

Inline Gas Sampling
Process Tubes Level Indicator
Independent Criticality Alarm System of the PRTR Control Room
Deionized Water System
Inlet Gas Seal Replacement
Backup Emergency Pump Low Speed Controllers
Helium Compressor Unloading Orifice Modification
125-Volt Battery Disconnect Contactor
Control Wiring for 13.8 KV Motor Operated Switches
Filtered Water Pump #3
Shim Rod Shroud to Top Cap Modification
Improved RTD Connector Sealant and Bracing
Provide Separate Emergency Backup Water for FEEF
Gamma Scanner Installation

Design Work Completed

Instrument Power Transfer System
Stiffener on Air Baffle

Design Work Partially Completed

Additional Fuel Storage and Examination Facility (CAH-106)
Vibration Snubbers for Earthquake Protection
Control Room and Instrument Calibration Shop Air Conditioning (CAH-995)
Decontamination Building and D₂O Cleanup Facility
DT-1 Storage Tank Sight Gauge
Flux Wire Scanning System
Supplemental Emergency Water Addition

Process Engineering and Reactor Physics

PRTR Test No. 57 (Increased Phosphate Concentration in HX-1) was partially completed. The free hydroxide in the steam generator was not eliminated as anticipated. Supplement 2 to the test was prepared to continue the investigation.

PRTR Test No. 68 (Primary System Flow and Pressure Drop Data) was completed.

The interior of the shroud tube in process channel 1847 was examined and no evidence of non-uniform or excessive corrosion product was seen.

A study was initiated to investigate the problems of testing the PRTR containment vessel at elevated pressures.

A study of the functions of the steam generator shutdown cooling valve was completed. It was found that the valve could be eliminated.

Two stagnant water tests were performed during the month and four fuel elements, each confirmed by the observance of fission gas bubbles in the storage basin, were removed.

An analysis was made on the fuel element rupture detection and location system to determine analytically the signal to background ratio that could be obtained if the system is placed into operation. A S/B ratio of 3 was found for steady state fission gas release; however, this calculation was made using somewhat questionable gamma yield information on the shorter-lived fission gases.

Procedures

Revised Operating Procedures issued	3
Revised Operating Standards issued	12
Temporary Deviations to Operating Standards issued	13
Revised Process Specifications accepted for use	2
Equipment Standards issued	1
Equipment Manuals issued	2
Maintenance Procedures issued	1

Drawing As-Built Status:

	<u>July</u>	<u>Total</u>
Approved for As-Built	5	989
In Drafting		60
In approval	9	20
Deleted or voided		81
		<u>1 150</u>
Scheduled for review		352
		<u>1 502</u>

Manhours

Personnel Training:

Qualification subjects	182
Specifications, Standards Procedures	68
Emergency Procedures	24
Maintenance Procedures	<u>97</u>
	<u>371</u>

Status of Qualified Personnel at Month-End:

Qualified Reactor Engineers	9
Qualified Lead Technicians	6
Qualified Technicians	17
Provisionally Qualified Technicians	2

Experimental Reactor Services

The status of the various test elements on July 31, 1963, is shown below. Those test elements which had reached their assigned goal exposure or had been permanently discharged for other reasons prior to July 1, 1963, have been deleted from this table.

Test No.	Location	F.E. Number	Description	Date Initial Charge	Date Discharged	Approx. Accumulated MWD
14	1956	5097	Moxtyl-Swaged	4/2/62	--	58.7 repad
14	1352	5098	Moxtyl-Vipac	5/8/62	--	103.3 repad
14	1758	5099	Moxtyl-Vipac	5/8/62	--	82.7 repad
37	1649	1097	UO ₂ Physics	5/12/62	--	99.9
37	1552	1098	UO ₂ Physics	5/12/62	--	96.4
37	1548	1099	UO ₂ Physics	5/12/62	--	98.8
37	1651	1100	UO ₂ Physics	5/12/62	--	87.4
48	Basin	5150	Moxtyl ($\frac{1}{2} \times \frac{1}{2}$ " pads)	8/1/62	6/29/63	61.9
54	1948	5116	Moxtyl (clip-on pads)	5/8/62	--	83.8 (68.3 w/clip)
54	1554	5118	Moxtyl (clip-on pads)	5/8/62	--	95.3 (80.3 w/clip)
61	1150	5179	Moxtyl-Physics	5/28/63	7/11/63	19.3
61	1443	5185	Moxtyl-Physics	5/28/63	--	29.0
61	1756	5186	Moxtyl-Physics	5/28/63	--	29.7
61	1847	5187	Moxtyl-Physics	5/28/63	--	33.8

Two additional unautoclaved LX Pu-Al elements were charged under Supplement 2 to PRTR Test 47, and are performing satisfactorily.

The final report was issued for PRTR Test No. 40 (Graphite Oxidation Tests).

Process Tube #1847 was removed this month as part of the regular tube analysis. It was cut into sections, two of which were shipped to Radiometallurgy Laboratory for testing.

Nine fuel elements were examined. Two were dismantled and one rod from each was shipped to Radiometallurgy Laboratory for analysis.

Plutonium Recycle Critical FacilityOperation

The Pu-Al substitution experiment was completed without incident. A reactivity measurement series for Pu fuel samples was started.

Preparations for irradiated fuel handling included modifications and refinements to equipment as well as operational testing and confidence building exercises for the thimble coolant system, the fuel transfer lock, remote operation of the cell crane, and underwater handling functions.

Improvement Work StatusWork Completed

Filter Modification to Permit Irradiated Fuel Handling
Changes to Facilitate Irradiated Fuel Handling

Design Work Completed

PRCF Light Water Conversion

Procedures

Two Process Specifications and one special condition were accepted during the month.

During the month, a TPO physicist became qualified as PRCF Supervisor.

Fuel Element Rupture Test FacilityWork Status

Design testing was completed during the month. The last test was a measurement of system response to a complete power failure in which the system flow rate decreased from approximately 200 gpm to 20 gpm; during this transient the system pressure decreased from 1800 psig to 1675 psig, which is acceptable.

Operation

A program was developed by the Experiment Scheduling Council for the initial six months operation of the Rupture Loop Facility.

GAS-COOLED POWER REACTOR PROGRAMPressurized Gas-Cooled Loop Facility (Project CAH-822)Work Status

The blower testing was completed successfully, with over 200 hours of operation and numerous starts. Parallel operation and successive starting of two blowers were achieved without incidence.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 21,327 hours. This includes 14,317 hours performed in Technical Shops, 3,078 hours assigned to J. A. Jones Company, 3,616 hours assigned to off-site vendors, and 316 hours to other project shops. Total shop backlog is 22,093 hours, of which 90% is required in the current month with the remainder distributed over a three-month period. Overtime worked during the month totaled 1,599 hours.

Distribution of time was as follows:

	<u>Man Hours</u>	<u>% of Total</u>
N Reactor Department	2 072	9.75
Irradiation Processing Department	2 903	13.61
Chemical Processing Department	248	1.14
Hanford Laboratories	16 097	75.47
Hanford Utilities and Purchasing	7	.03

LABORATORY MAINTENANCE OPERATION

Total productive time was 15,500 hours of 18,500 hours potentially available. Of the total productive time, 86% was expended in support of Hanford Laboratories components with the remaining 14% used in providing service for other HAPO organizations. Manpower utilization for July was as follows:

A. Shop Work	2 400 hours
B. Maintenance	<u>5 800</u> hours
1. Preventive Maintenance	1 700 hours
2. Emergency or Unscheduled Maintenance	1 600 hours
3. Normal Scheduled Maintenance	2 500 hours
4. Overtime (Included in above figures)	1 000 hours
C. R&D Assistance	<u>7 300</u> hours

WD Richmond

Manager
Test Reactor and Auxiliaries

WD Richmond:bk

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.

INVENTORTITLE OF INVENTION OR DISCOVERY

C. E. Fitch, Jr.
R. B. Jacob
G. L. Waldkoetter

Critical Angle Ultrasonic Tests

C. E. Fitch, Jr.

The Control of the Transmitting and
Receiving Properties of Ultrasonic
Transducers

R. F. Warnick

Device for Controlling Reactivity in a
Reactor
(Patent Award - July 24, 1963)

Manager, Hanford Laboratories

END

**DATE
FILMED**

3 / 29 / 93