[redacted]

716568

HAN-85454

HW-77709

19A

DECLASSIFIED

HANFORD LABORATORIES MONTHLY ACTIVITIES REPORT

MAY, 1963

REPOSITORY PNL

COLLECTION I-131 Atmosphere Studies

BOX No. N/A

FOLDER N/A

JUNE 14, 1963

HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

GENERAL ELECTRIC

DECLASSIFIED

[redacted]

A.E.C.-G.E.-RICHLAND, WASH.

1103403

DECLASSIFIED

[redacted]

HW-77709

This document consists of 226 pages. Copy No. [redacted] of [redacted] copies. Series [redacted]

HAN-85454

HANFORD LABORATORIES MONTHLY ACTIVITIES REPORT MAY 1963

Classification Cancelled (Change to DECLASSIFIED [redacted]

By Authority of CG-PR2
Robert Stein 8-21-92
By J H Wells 9-11-92
DG Kreso 9-14-92

Compiled by
Section Managers

June 14, 1963

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

[redacted]

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.

Route To:	P. R. No.	Location	Route Date	Signature and Date
Koo				
JE Travis				

DECLASSIFIED [redacted]

1103404

DISTRIBUTION

Copy Number	
1	W. E. Johnson
2	H. M. Parker
3	F. W. Albaugh
4	C. A. Bennett
5	F. E. Crever, GEAPD
6	I. H. Dearnley
7	W. E. Foust
8	P. F. Gast
9	A. R. Keene
10	H. A. Kornberg
11	R. S. Paul
12	W. H. Reas
13	W. D. Richmond
14	W. Sale
15	H. P. Shaw - W. S. Frank
16	W. K. Woods
17 - 20	Atomic Energy Commission, Richland Operations Office Attn: J. E. Travis
21 - 23	F. P. Baranowski, Director, Division of Production Washington 25, D. C.
24	Savannah River Operations Office, Aiken, South Carolina
25	300 Files
26	Record Center

TABLE OF CONTENTS

TABLE I - HANFORD LABORATORIES FORCE REPORT

DATE: May 31, 1963

	At Beginning of Month		At Close of Month		
	Exempt	Salaried	Exempt	Salaried	Total
Chemical Laboratory	133	125	137	127	264
Reactor & Fuels Laboratory	187	177	186	178	364
Physics & Instruments Laboratory	103	71	104	77	181
Biology Laboratory	40	60	40	62	102
Applied Mathematics Operation	17	4	17	5	22
Radiation Protection Operation	44	90	43	90	133
Finance & Administration Oper.	106	113	98	112	210
Programming Operation	11	3	12	3	15
General	3	4	3	4	7
Test Reactor & Auxiliaries Oper.	57	304	58	306	364
TOTAL	701	951	698	964	1662

BUDGET AND COST SUMMARY

May operating costs totaled $2,689,000, an increase of $136,000 over the previous month; fiscal year-to-date costs are $26,398,000, or 86% of the $30,536,000 control budget. Hanford Laboratories' research and development costs for May compared with last month and the control budget are shown below:

(Dollars in thousands)	COST Current Month	Previous Month	To Date	Budget	% Spent
HL Programs					
02 Program	$ 112	$ 87	$ 858	$ 1 069	80
03 Program	22	27	140	175	80
04 Program	1 166	1 012	10 934	12 683	86
05 Program	112	128	1 176	1 353	87
06 Program	267	254	2 833	3 154	90
08 Program	16	21	73	97	75
	$1 695	$1 529	$16 014	$18 531	86
NRD Sponsored	202	186	1 103	1 270	87
IPD Sponsored	66	70	832	897	93
CPD Sponsored	107	96	1 275	1 421	90
FPD Sponsored	--	--	493	493	100
Total	$2 070	$1 881	$19 717	$22 612	87%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

No detrimental effects from phosphorus impurity in uranium have been observed in clad-to-core bonds or braze-to-core bonds in N fuel. Phosphorus lowers the room-temperature ductility and ultimate strength of uranium, but tests at temperatures of 100 C and higher show no differences due to phosphorus. A study of the effect of sulfur impurity in uranium has been started.

Outer components for N fuel continued to appear satisfactory in performance and behavior as shown by the postirradiation examination of three N outer fuel element components.

[redacted]

The high temperature swelling performance of uranium fuel elements has tended to verify that fuel swelling is ultimately limited by coolant pressure.

Fluted, single tube fuel elements (N-Reactor size) operating under N-Reactor conditions continue to perform satisfactorily. A companion element failed during irradiation in a cold water loop.

Clad thinning, with intentional internal striations, has been experimentally produced during irradiation of experimental capsules.

Uranium samples containing dispersions of UAl_2, U_6Fe, uranium carbide, and uranium phosphide are being studied as alternate core materials.

Only slight differences have been observed between Zircaloy-2 and Zircaloy-2 containing up to 1 wt% uranium in corrosion tests in 360 C water after 150 hours.

Damage through wear on N-Reactor process tubes caused by fuel element supports after six simulated charging operations, with intervening autoclaving between each charging operation, has scratched the process tube to a maximum depth of 2.2 mils. This rate of wear is five times that of tests made on a tube without intermediate autoclaving.

A theoretical study has been completed on the effect of process tube vibrations and their relation to fuel element support design.

Multiproduct fuel element development is proceeding through a study of fabrication methods, recrystallization temperatures, diffusion of lithium in aluminum, and out-of-reactor target element defect tests.

Metallographic examination of N-Reactor self-supports on a Zircaloy-2 clad heater rod exposed for 146 days at 315 C in lithiated water at pH 10, and at a heat flux of 280,000 Btu/ft^2 hr, revealed accelerated corrosion at each of the crevices (eight) between the self-supports and the jacket. Maximum penetrations were about 6 mils.

[redacted]

Significant fuel element self-support flattening was encountered in N-Reactor charging machine tests when fuel columns were charged with 200 to 500 pounds back-pressure, but when the process tube was loaded with fuel elements to provide prototypical back-pressure the self-support flattening was reduced from about 25 mils to about 5 mils.

Equations have been developed for application of laboratory boiling burnout data to the N-Reactor.

Based on test operation of K-1 Loop for about 30 days with the pH controlled with NH_4OH, ammoniated systems control crud better than lithiated systems at low temperatures, probably because of hydrazine formation which getters oxygen in the system.

The K-1 and K-2 Loops were decontaminated sufficiently to allow maintenance operations during tube replacement in 105 KE. The K-2 Loop (stainless steel) was treated with alkaline permanganate followed by sulfamic acid; the K-1 Loop (carbon steel) was decontaminated using a single-pass procedure with inhibited ammonium citrate solution.

An unautoclaved section of 30% cold worked N-Reactor pressure tubing exhibits a primary creep rate about five to six times that of autoclaved tubing, and a much higher secondary creep rate. This strengthening effect of autoclave treatment was previously noted on annealed Zircaloy-2 strip material tested under uniaxial creep loading conditions.

Results of the N-Reactor atmosphere computer program have been correlated by means of an empirical expression which relates maximum graphite burnout rate, maximum graphite temperature, initial water vapor partial pressure, and initial hydrogen partial pressure.

The water-gas shift reaction ($CO + H_2O \rightarrow CO_2 + H_2$) was observed to be significant above 500 C during experiments in which He, CO, H_2O gas mixtures were allowed to diffuse through hot graphite thimbles. In previous experiments in which the gas mixtures flowed past a graphite bar little reaction occurred. It would appear that the water-gas shift reaction could produce significant amounts of hydrogen in the N-Reactor tube block, but probably not in the gas channels.

The in-reactor creep rate for 20% cold worked Zircaloy-2 after 1700 hours at 20,000 psi stress and 350 C was determined to be 1.8×10^{-6} in./in./hr. The rate during reactor outage was essentially the same. An error in the out-of-reactor rate measurement has been corrected and a rate of 0.5×10^{-6} in./in./hr at 2000 hours has been determined.

In-reactor creep capsules have been shown capable of operation at 900 C in a laboratory mockup. Capsules with this capability will be used to test stainless steels.

Zircaloy-2 bend-test specimens with 23% cold work have a ductile-to-brittle transition temperature of about -75 C (-167 F). Examinations of fracture surfaces by electron microscopy reveal similar crystallographic features for both transverse and longitudinal specimens.

In-reactor weight gain data have been obtained for Zircaloy-2 samples after 125 days in 540 F (282 C) water and a fast flux of 7.6×10^{13} nv. The average weight gain of 121 mg/dm^2 compares with 9 mg/dm^2 for reference coupons exposed in the out-of-reactor loop at similar conditions.

A proposal was prepared for experimental evaluation of N-Reactor shielding during plant startup using the top shield plug facility of the N-Reactor. The MAC shielding code has been modified slightly to eliminate an error in the gamma calculation, and the BARNS code has been debugged to calculate unit weighted shield material cross sections.

The gamma-induced oxidation of graphite by low concentrations of water in flowing helium was found to be independent of flow rate. In studies of the inhibition of oxidation of graphite by halo-methanes, the rate of oxidation could be related to surface areas as measured by the BET method.

[redacted]

DECLASSIFIED

The flash thermal diffusivity technique was successfully used to determine thermal conductivity of porous carbon having a value only about 1/30 of that of CSF graphite. The use of a lead sulfide detector to monitor the rear surface temperature rise has not been wholly successful.

In a half-reactor test the corrosion rates of reactor materials appear to be about the same in low alum (7.5 ppm) as in high alum (15 ppm) water, except for type 1100 aluminum (tube material). This aluminum corroded slightly more in the higher alum concentration.

The General Electric Technological Hazards Council accepted the final response to the audit of PRTR operation, approved the discharge method for the PRTR Fuel Element Rupture Test Facility, and approved modification of the PRCF for light-water-moderated critical tests.

The AEC approved a proposed increased of limits on PRTR maximum heat transfer flux and tube power.

Revised PRTR aqueous effluent activity limits have been developed which will become effective when the new water treatment plant at Richland begins operation.

By means of a new recommended procedure, recovery of the PRTR primary coolant system from boiling convection cooling could be accomplished without further loss or degradation of D_2O and with minimal thermal shock to the system.

Detailed design of the second-generation shim rod assembly for the PRTR is 90% complete, and fabrication of the two identical prototype units is 55% complete.

[redacted]

1103412

The entire PRTR fuel loading was examined in the basin, and ten selected elements were examined in the Fuel Element Examination Facility. In general, the fuel elements were in good condition. Defects included: scratches on exterior surfaces of many of the elements, displacement of some circumferential band-type bundle wraps, and loosening of rod wire wraps on some of the Al-Pu spikes. The clip-on wear pads on the UO_2-PuO_2 elements are performing very satisfactorily.

Individual fuel rods from a vibrationally compacted UO_2-PuO_2, PRTR Mark-I element that failed in the PRTR were examined in detail. Leaks were detected in the heat-affected zone of the top end-cap welds of three rods. Likely causes of this localized corrosion are either contamination of the Zircaloy during welding or the presence of some impurity in the fuel material which caused severe attack from the inside.

Four vibrationally compacted capsules containing UO_2–2.5 wt% PuO_2 fuel of the type under consideration for use in the EBWR were charged into the ETR on May 13.

The second UO_2-PuO_2 fuel cluster containing recycled PRTR plutonium was completed; the first cluster was charged into the PRTR.

Gamma scan measurements of Pu concentration correlated with calibrated UO_2-PuO_2 rods in the range 0.1 to 10% Pu.

A uniformity of Pu distribution of ± 0.1 wt% was obtained in "bottle loaded" vibrationally compacted UO_2 – 0.48% PuO_2 fuel rods for PRTR.

The rejuvenation fuel element (GEH-4-81) is ready for return to the MTR. Rejuvenation was accomplished by adding enriched UO_2 to a fuel rod from the fuel assembly which had been irradiated in January in the MTR.

DECLASSIFIED

[redacted]

1103413

Equipment has been developed and fabricated for rapidly positioning and tightening a cluster fuel element wire spacing member remotely, under approximately 10 ft of water.

Four elevated-temperature burst tests of PRTR pressure tubes (at temperatures from 200 F (93 C) to 400 F (204 C)) indicate that the burst strength decreases linearly with temperature over this temperature range.

The ratio of thermal-plus-epithermal to fast neutrons was found constant over 70% of the fuel length of the PRTR core. This ratio determined by nickel and cobalt wire monitors was 12.3 ± 0.2.

A Zircaloy-clad uranium honeycomb element has been successfully fabricated by coextrusion. Three coextruded Zircaloy-clad thorium-uranium fuel elements have been irradiated to 1100 Mwd/ton with no appreciable swelling occurring. Irradiation will continue in the ETR loop.

Autoclave tests show that thorium contamination of Zircaloy-5 Be braze has a definite detrimental effect on the corrosion resistance of the braze. Allowable levels of thorium contamination of brazing alloy are being established.

In preparation for fabricating EBWR fuel, uranium dioxide was prepared and densified successfully by pneumatic impaction, and preparation of PuO_2 from 60 kg of Pu metal was completed.

The pneumatic impaction technique was used to densify 1100 grams of PuO_2 at one time. Density achieved was 10.89 g/cc (95.1% TD).

PuN pellets heated in flowing hydrogen for 12 hours at 1700 C slumped and flowed together, possibly due to the formation of free molten plutonium.

Several plutonium-bearing borides were prepared by reacting PuO_2, carbon, and boron at 1600 C in vacuum.

The density of pneumatically impacted PuN–50 vol% W was 16.9 g/cm^3 (100% TD). However, the PuN was not evenly distributed in the tungsten.

Plutonium was selectively lost from molten UC–20 wt% PuC.

[redacted]

Thin wafers of Pu-15 wt% Zr imbedded in UO_2 pellets oxidized but did not diffuse into UO_2 after heating for 12 hours at 1700 C.

X-ray diffraction data revealed no crystallographic changes in MgO-13.5 wt% PuO_2 after irradiation to 0.5×10^{20} fissions/cc at a maximum temperature of 1700 C.

Analyses of the first two samples of pneumatically compacted PuO_2 revealed densities of 89 and 95% TD with oxygen-to-plutonium ratios of 1.94 and 1.91.

A UO_2-50 wt% W cermet was examined by reflection electron microscopy during controlled heating to 1900 C.

Six insulated cermet capsules (two Mo-UO_2, two Mo-UN, one W-UO_2, and one W-UN) were charged into the ETR and are to be irradiated with fuel surface temperatures in excess of 2000 C.

A tungsten-clad UO_2 capsule designed to operate with a completely molten core was charged into the ETR on May 13.

Chemical analysis of UOS powder prepared from a fused chloride melt revealed an uranium content of 84.7 wt% (vs theoretical of 83.2 wt%).

The absorption coefficient for single crystal UO_2 was measured at wavelengths between 0.6-15 microns. There is a large optical window in the range 3-13 microns, with a minimum at 11 microns.

The relative transmission of a laser beam (wavelength 0.69 μ) was significantly greater in single crystal UO_2 than in polycrystalline UO_2. Transmission at this wavelength decreased with increasing temperature to 1000 C.

At intermediate temperatures (e.g., 1200 C), single crystal UO_2 exhibits enhanced heat transmission, relative to polycrystalline UO_2, due to radiant heat transmission.

A single crystal of UO_2 was irradiated for 310 days, to an estimated exposure of 5×10^{20} fissions/cc.

Autoradiographs of fuel elements defected during irradiation revealed significant relocation of fission products in the immediate area of the defect.

Laboratory data for air cooling of 19-rod fuel bundles in the Fuel Element Examination Facility were correlated using the Colburn equation for forced convection heat transfer with a modified velocity term.

Eighty-eight boiling burnout data points obtained with 19-rod bundle test sections were correlated fairly well by a new empirical method appearing in the May, 1963, issue of Nucleonics. Data obtained with long (76-inch) test sections was not published.

Employing fog cooling (steam-water mixtures) 42 experimental boiling points were determined with an electrically heated 19-rod bundle test section. The burnout heat flux at low flow rates was quite sensitive to the hydraulic stiffness (i.e., availability of excess inlet pressure) of the system.

X-ray examination of molybdenum foils annealed at 900 C for 2 hours after irradiation to 10^{19} nvt (E > 1 Mev) showed that lattice parameters and line widths have returned to preirradiation values. The foils containing 100–200 ppm carbon and 400–500 ppm carbon have apparently recrystallized to some extent, as ratios of peak intensities are different from the corresponding ratios in the unirradiated foils.

Defect structures have been observed in irradiated molybdenum foils by transmission electron microscopy. Deformation of the irradiated foils causes dislocations to move in straight line channels. The moving dislocations interact with the defect structures sweeping them from the channel.

To prepare foils for transmission electron microscopy, spark machining and an electrolytic jet polisher have been successfully employed to cut wafers from a single crystal of molybdenum.

A molybdenum single crystal annealed at 2000 C for 1 hour was used for testing the validity of lattice parameter measurements. Six measurements at different points on the crystal face yielded an average value of 3.14707 A, with a mean deviation of 0.00003 A, or one part in 10^5. A low-carbon (10–20 ppm) crystal irradiated to 10^{19} nvt (E > 1 Mev) showed the same point-to-point variation in lattice parameter as was observed on unirradiated crystals.

A capsule containing eight high-purity molybdenum electrical resistivity specimens has been prepared for irradiation in the SNOUT facility.

It has been established that the creep rates of plutonium during the alpha-to-beta and beta-to-alpha transformations under compressive loads of 1000 and 2000 psi are greater than would be predicted from the rates of either of the phases individually.

Two thorium specimens irradiated to burnups of 0.18 and 0.92 at.% at less than 300 C (572 F) and then annealed at 750 C (1382 F) for 100 hours exhibited essentially no change in the as-irradiated optical microstructure. A small decrease in density was observed with the 0.92 at.% burnup sample.

Thorium-uranium tensile specimens irradiated from 0.5 to 1.5 at.% burnup at less than 300 C (572 F) are being tensile tested at room temperature. The first two specimens tested had limited ductility.

Optical and electron metallography were completed on the irradiated uranium specimens recovered from two previously opened swelling capsules which operated at 575 C (1070 F) to 0.15 at.% burnup. The enriched specimens suffered less damage than did the natural ones. Grain boundaries were readily observed in the enriched specimens.

Impregnation was found to be the most important factor in a series of radiation experiments with samples of graphite prepared from controlled particle size filler material, with different binder materials, and using different impregnation treatments. Multiple impregnations with nongraphitizing binder resulted in an increased contraction rate.

DECLASSIFIED

Document HW-77066 has been issued describing a preliminary Plutonium Fuel Spacecraft Reactor concept.

Preliminary calculations are under way to set core parameters for a 1000 Mw nuclear rocket reactor, fueled with plutonium. An unclad tungsten-base cermet fuel of honeycomb configuration, if found to be structurally practical, would provide appreciable savings in the size and weight of the reactor.

2. Physics and Instruments

Preparation of a formal report on N-Reactor physics experiments is nearing completion. All experiments have been normalized to a standard corresponding to the PCTR mockup lattice.

Data were obtained on an overbored exponential pile mockup of C-Reactor. Flux traverses were made with and without six safety rods.

The PCTR is being used to measure k_∞ for the K-Reactor lattice with zirconium process tubes. The worth of a poison spline is also being measured. Data from these experiments are still being analyzed.

Cooperative tests are being performed with Instrument Development, IPD, to evaluate possible N-Reactor startup instrumentation systems. Pulse height distribution measurements have been made at various neutron counting rates using BF_3 proportional counters. Source range log count rate and period instrumentation tests have also been performed at HTR. The results suggest the need for continued instrument evaluation tests before N-Reactor startup.

N-Reactor kinetics and primary coolant loop simulations (for the power range) have determined the effects of variable reactivity ramps, inlet temperature, instrument trip settings and time constants, initial power level, power setback rates, and scram time functions on a number of system variables.

One thousand two hundred Zircaloy process tubes were tested, sorted, and made available for KW-Reactor retubing in time to meet a May 6 goal date. One thousand six hundred tubes were inspected to meet this goal, 40 of which have been returned to the vendor as defective.

Nine possible methods of assuring the integrity of some 2000 N fuel elements whose support clip welds are under suspicion have been communicated to N-Reactor groups. On the basis of these suggestions, cooperative efforts are being made to develop a practical method of inspecting the weld nuggets.

Critical mass experiments on plutonium nitrate solutions have produced results which will permit upward revision of certain criticality data. A concentrated solution (435 g Pu/liter) was found to have a critical volume about 40$ larger than previously predicted from data extrapolations and geometry conversions.

Critical Mass Laboratory experiments are being shifted from plutonium solutions to solid mixtures of PuO_2 in polystyrene. The switch in instrumentation to the second hood where the solid experiments will be performed was completed. Preliminary subcritical neutron multiplication measurements were started on May 24.

Against experimental data, GAMTEC, a computer code, has been tested on a graphite-moderated reactor core. The code will be used to predict nuclear safety parameters for heterogeneous systems of slightly enriched uranium in water.

Improved numerical methods are being developed to predict the analytic value for the buckling of a bare hemisphere of a solution of fissionable material. INTERSET, the code which calculates the multiplication of a system of subcritical assemblies, has been completed, and a request for it from an offsite user has already been received and filled.

Measurements of slow neutron inelastic scattering from H_2O at 95 C have been made for 11 different values of the neutron energy change in the

DECLASSIFIED

range 0.0375 to 0.25 ev. Computer codes have been obtained from AERE, Harwell, for use in calculating the scattering law from the experimental data.

Development of a rotating-crystal neutron spectrometer continued with tests on phase stability of two rotors operating at 12,000 rpm, and tests on the time jitter of neutron detectors.

Data obtained in April on the total cross sections for neutrons from 3 to 15 Mev for 14 samples were processed to obtain cross sections.

PCTR data on high exposure (20.6% Pu^{240}) PuAl clusters in 6-1/2, 8-3/8, and 10-1/2-inch graphite lattices are being compiled for reporting.

PRCF startup experiments continued with investigations of the worth of the safety rods, the time response of the safety system, and the neutron flux distribution in the core.

Approach-to-critical measurements on 0.5-inch-diameter PuAl rods (16.46% Pu^{240}) in light water indicate that about 1300 rods would be required for criticality when the rods are spaced on 0.66-inch centers in a triangular lattice. This is based on an extrapolation from a loading of 850 rods.

Interpretation of data obtained in ARMF measurements on PuAl samples irradiated in the MTR continued. Values of the fission and absorption cross sections of the irradiated samples were estimated from the ARMF data. It appears that further sensitivity measurements should be made on the ARMF.

Work is under way to use several experiments performed in the PCTR in studying the moderating properties of graphite. Computer program SPECTRE is being modified for use in this work. The study of the theoretical scattering laws for water is continuing.

Some exploratory work is under way to determine the feasibility of carrying out reactivity coefficient measurements of some H_x Pu samples in a fast spectrum critical facility. These experiments should help to resolve the considerable uncertainties which exist in the high energy cross section data of the higher Pu isotopes.

The survey of UO_2, PuO_2, and PuN as fuels for small, compact, fast reactors has now been extended to include 30 vol% tungsten fuel elements in addition to the 0 and 50 vol% concentrations previously considered.

An investigation of control rod effectiveness of various neutron absorbers in a series of Pu environments was initiated during the month. Preliminary results on a 1/v absorber (e.g., boron) indicates that effectiveness generally decreases with increased Pu^{240} content.

The revised cross sections for Pu^{241} and Pu^{242} reported in HW-75716, "Updated RBU Basic Library," give considerable reactivity increase (~6%) for one of the "Phoenix Study" reactors. This effect should assist in obtaining a favorable (flat) reactivity response with burnup on Phoenix fuel reactors.

Experimental data are now being obtained on the L_x Pu-Al physics test elements 5051, 5095, and 5092 which had received exposures in the PRTR of 13.1, 40.2, and 71.1 Mwd, respectively. These are the first data to be received on the physics elements.

A major error in the construction of tables describing the distribution of target velocity magnitudes in the Monte Carlo part of RBU has been detected and corrected. This error was the major cause for the long delay in completing the RBU code. Correction of the error resulted in Monte Carlo generated neutron spectra which are in excellent agreement with theory. Complete check-out of the code can now proceed.

Development work continues on the CALX burnup code, the Hanford Revised GAM, HRG, and the ALTHAEA-MELEAGER codes. A substantial improvement has been made in the usefulness of the latter since the thermal and epithermal reactions can now be conveniently "zeroed" at the beginning of the analysis to agree with codes that treat the energy spectrum in greater

DECLASSIFIED

detail than Westcott methods allow. Two thermal neutron scattering codes, LEAP and ADDELT, were imported from Harwell, England, and put in operation. These will aid in our analysis of scattering data and in the development of scattering kernels.

The project proposal for design funds for the High Temperature Lattice Test Reactor has been approved by the Richland Office of the AEC. Design criteria are undergoing final review and approval steps.

Fabrication of some experimental extended-lifetime in-core neutron flux monitors is nearly complete. Their plutonium isotopic compositions are chosen to seek optimum regeneration behavior in the KE water-cooled irradiation facility. The neutron spectral parameters for this facility were recently determined experimentally.

Three Boron-11 beta-current neutron detectors, which were fabricated offsite to HAPO specifications, have been received and are now being made ready for reactor tests.

Four-parameter separation has been successfully achieved on the prototype multiparameter eddy-current tester as applied to a three metal layer test sample. Separation of the probe spacing parameter ranged from about 3 to 12 mils; each of the metal thicknesses were separated in the range of 2 to 6 mils. Instrument application is more complex for this case than for the three-parameter case.

An experimental procedure for measuring ultrasonic transducer frequency, band width, radiated beam pattern and damping (Q) has been developed. Applied to transducers used in routine test applications of high volume, these tests are expected to do much to assure uniform transducer characteristics and thereby improve the uniformity of test results. Arrangements have been made for transducer vendors to apply the tests before shipping transducers to HAPO.

[redacted]

Favorable customer evaluation of two improved prototype miniature gamma dose meters has promoted the purchase of 25 more units (from outside sources) for continued testing. The principal improvements of these models were relocation of the sensor to one edge of the unit, and the use of a molded Teflon central conducting rod. Tests revealed meter sensitivity dropped only 20% as the gamma source was positioned in the least favorable geometrical position with respect to the dose meter.

The energy sensitivity to neutrons and photons of our large tissue-equivalent ionization chambers were measured.

A final report on the cosmic-ray neutron measurement was finished.

Results from the survey by urinalysis in Alaska continued to be received. Fort Yukon is the only village sampled where body burdens appear to be high compared to other Alaskan villages.

Radiological physics work continued on: P^{32} counting shadow shield calibration, assistance to RPO in preparation of their truck shadow shield, X-ray counting, and standardization of the precision long counter.

Appraisal of the environmental consequences of reactor accidents will be improved by development of an atmospheric dispersion prediction scheme which permits more realistic accounting for the period of release of the contaminants. By superposition of incremental releases, Hanford atmospheric diffusion data were extended to prediction of downwind exposure levels to periods of several hours. The method was independently verified to a distance of 8 miles from a release near ground level.

3. Chemistry

The coating of aluminum reactor components with special inks continues to show promise for affecting reductions of radioisotopes in reactor effluent water.

The release of polonium from irradiated, aluminum-clad bismuth held at 1300 C for 2 hours was found to be 16 to 23%, whereas 50 to 60% release was observed for unclad bismuth after 1/2 hour.

Preliminary evaluation of several out-of-service cribs and trenches revealed very little movement of radioisotopes due to soil column drainage during the period from 1959 to the present.

The removal of radio-ruthenium from neutralized Purex process condensate by selected activated charcoals and oxides is found to range from 66 to 97% in 24-hour equilibration periods.

Continued observations of a sealed, strontium loaded, zeolite cartridge indicate no evidence of radiolytic pressure buildup.

A precipitation process was devised and tested for the recovery of strontium from acid-dissolved, Purex waste tank sludge. Ninety-five to 97% recoveries were achieved in the precipitation step.

The Pm^{146} content of well aged promethium was found to be a factor of 3 lower than expected and casts doubt on presently accepted nuclear constants and/or decay schemes associated with Pm^{146} production, burnup or decay.

The Sr^{90} isotopic concentration of strontium in the sludge of the Purex 106-A waste tank was found to be 52%, as compared to 55% in the current Purex 1WW and only about 17% in tank 103-A, the only tank previously analyzed.

The extraction of neptunium and plutonium from synthetic Purex 1WW by trilauryl amine was found not to be impaired by the presence of 5 vol% tributyl phosphate or uranium concentrations of 0.027$\underline{M}$.

Laboratory tests show the CSREX solvent (D2EHPA-BAMBP-Soltrol) can be protected from degrading in 1$\underline{M}$ HNO_3 solutions at 50 C by the presence of sulfamic acid, hydrazine, or urea.

Treatment of synthetic Purex waste tank sludge with a solution of phosphoric and nitric acids maintained at a low pH was effective in reducing the hard sludge cake to a pumpable slurry.

[redacted]

Either clinoptilolite or Linde AW-400 (20 to 50 mesh) is satisfactory for a plant test evaluating the extraction of cesium from Purex formaldehyde treated waste. Duolite C-3 resin was the most suitable exchanger for removal of cesium from Redox supernatant wastes.

Scaling and reduced heat transfer coefficients are observed for titanium heat transfer surfaces during the concentration of Purex acid waste solutions. Fouling of stainless steel heat transfer surfaces under similar conditions does not occur.

Preliminary experiments, for testing the removal of chlorine from Salt Cycle process off-gases (oxygen-chlorine mixtures) by low-temperature liquifaction techniques, resulted in recoveries of 80 to 96%. Storage of the liquified chlorine at 27 C in a sealed pressure cylinder was readily achieved.

Laboratory studies of the parameters governing the preparation of UO_2-PuO_2 solid solutions by electro-codeposition have shown that a plutonium enrichment factor of 40 and a promethium decontamination factor of 6 can be achieved in the preparation of a deposit containing 23 wt% PuO_2.

Mixed UOS and β-US_2, prepared by precipitation from chloride melts, can be washed with dilute hydrochloric acid to yield a crystalline UOS product uncontaminated with β-US_2 or UO_2.

The radiant heat spray calcination of a tartrate (and sugar) bearing alkaline Purex waste resulted in the formation of tars in the spray nozzle and also caused formation of unreacted organics in the product calcine.

4. Biology

The effect of reactor effluent exposure on the swimming performance of young chinook salmon is being measured. Since larger fish swim better than smaller ones, it is expected that fish exposed to 3% effluent will perform better than those exposed to 6% effluent because of the greater growth inhibitory effect of the latter.

DECLASSIFIED

Vertan 690 (active ingredient, EDTA) will be used to clean out N-Reactor pipes. Since N-Reactor personnel are considering the disposal of the waste material to the Columbia River, we ran some toxicity tests on Vertan 690. From results, it was estimated that an upper limit avoiding acute toxicity would be about 20 ppm. If the material is released to the river toxicity may get uncomfortably near this figure. This information was verbally passed on to personnel in N-Reactor.

In the course of temperature tests on sensitivity of columnaris, a new colony form was isolated. This one is smooth, as opposed to the usual rough form, and promises to offer extended possibilities for laboratory work on the microorganism.

The reserve capacity of the hematopoietic system in swine is apparently unaffected by feeding them 125 μc Sr^{90}/day. This was determined by noting that no difference in blood constituents occurred between this group and control animals.

Five of 17 ewes given 3 mc I^{131} 5 years ago failed to conceive or experienced early death of the embryo. All control ewes conceived at first mating and had uncomplicated pregnancy.

After 45 days of feeding inert iodine (2 g/day) to a cow also being fed I^{131}, the I^{131} in the milk had diminished to 50% of its previous value and I^{131} in the thyroid had diminished about 5% of its previous value.

Observations of rats being fed Cu and Mo (in a radiation synergism experiment) revealed that high concentrations of Mo in the diet causes increase in liver-Cu concentrations.

One dog died 1200 days after a single inhalation of $Pu^{239}O_2$. The highest concentration of Pu was found in the pulmonary lymph nodes—several hundred times higher than in the rest of the lung. Another dog died 8 months after 2 mc $Ce^{144}O_2$ via inhalation had been deposited in it. Remaining at death was about 1 mc distributed in the lungs, lymph nodes, and liver.

[redacted]

1103426

Six beagle dogs were flown to the Nevada Test Site and exposed to Pu released from a one-point detonation of a nuclear weapon and later returned to this laboratory.

The distribution of inhaled PuO_2 in rats was determined 4 months after exposure. Nuclear and cell debris contained about three-quarters of the Pu. Most of the remaining was in the fibrous tissue.

Cholestyramine, a material which was assumed to be capable of absorbing bile salts, was tested for its effectiveness in improving survival of irradiated rats. The negative results obtained were shown to be due, in part, to the only partial effectiveness of cholestyramine in removing the bile salts from the intestinal lumen. Another approach was tried in which the rats were surgically modified in such a manner that the bile drains into the urinary bladder. This procedure is apparently effective in preventing the diarrhea which normally follows irradiation.

Exposure of Tribolium castaneum to 980 rads of fast neutrons reduced reproductive ability in both sexes. Females were less affected reproductively than males.

Twenty mc of Rb^{86} were introduced into the upper end of Rattlesnake Springs. Activity in the water later decreased with distance downstream. No pulse in activity was found at the dam. Substrate samples are showing that the isotope is being rapidly absorbed by surfaces.

5. Programming

The purity of Cm^{244} is increased by operating at high specific power, since less Pu^{241} is allowed to decay. As the specific power is raised it becomes less important to keep the plutonium which is being burned separate from new Pu^{239}, insofar as purity of Cm^{244} is concerned. In fact, calculations made this month indicate that higher purity is obtained by mixing plutonium with U^{238} and operating at a high power per gram of plutonium present, than is obtained by mixing plutonium with thorium and operating

 [redacted]

at a low power per gram of plutonium. The actual comparison was made on the basis of the same power per ton of fertile material, but much smaller plutonium concentrations are needed to enrich U^{238} than to enrich thorium, because of the lower U^{238} cross-section.

Calculations of the production of fission product ruthenium, rhodium, palladium, krypton, xenon, and cesium as a function of fuel exposure were made during May, augmenting the results on four other elements reported during the previous month.

TECHNICAL AND OTHER SERVICES

Eleven plutonium incidents occurred during the month which required special bioassay sampling of the personnel involved to determine if internal deposition had occurred. In one of these cases excision was used to remove plutonium from a wound. Pending bioassay results for the persons involved in these incidents, the total number of plutonium deposition cases that have occurred at Hanford is 317 of which 229 are currently employed.

Several GE employees were inadvertently exposed to a 27-curie Ir^{192} source at N-Reactor on May 16, 1963. The source, licensed by the AEC for use by the Pittsburg Testing Laboratory, remained in an unshielded and unattended location in Cell 4 of the 109 Building for several hours. During the time the source was unattended, at least 14 GE employees were in the immediate vicinity. Evaluation of the film badge dosimeters worn by the GE employees showed a maximum whole body dose of 0.2 rem of gamma radiation. The exposures received by the construction personnel were investigated by the AEC Compliance Division.

General increases were noted in all air filter sample results during the week ending May 24, 1963. The average value at the offsite Pacific Northwest locations was 13 pc β/m^3, one of the highest weekly averages noted since the fall of 1961. This increase probably represents the expected spring influx of older fallout materials from past tests. Two of the air filters containing the highest concentrations of beta emitters were scanned in an attempt to estimate the age of the fission products collected. The results indicated an age of approximately 1 year.

A false critical radiation alarm occurred at the 309 Building on May 31, 1963, at 3:30 a.m. as the result of X-ray testing in the 11-foot level of C cell. The reactor engineer had just announced an X-ray shot over the loud speaker when the criticality false alarm sounded. Personnel evacuated the containment vessel. They did not evacuate the 309 Building since they assumed that the alarm was caused by the X-ray work. The alarm was bypassed for the remainder of the X-ray work. The reactor was not operating at the time of the alarm.

Continuing assistance is being given to the Fuels Production section of IPD in determining those canning conditions which will lead to a given quality of fuel element. In particular, the experiments have been designed and the resulting data analyzed to increase understanding of the effects of time, temperature, and pressure on important fuel element characteristics and to increase knowledge of the quality of incoming materials and process capabilities. Emphasis is placed on the correlation of these manufacturing conditions with observed postirradiation effects.

Analyses on the mathematical theory, use, error propagation, and EDPM program for both the optical and electronic process tube traverse mechanisms have been completed and recommendations for their improvement and use discussed with the customer.

The estimation of U^{235} content of UO_3 shipments will be made from duplicate measurements on each car shipped using the 234-5 laboratories thermionic emission mass spectrograph. At the same time a UO_3 standard of known isotopic content will be analyzed to permit subsequent estimation of absolute reliability.

[redacted]

The EDPM program which simulates the N-Reactor stack gas composition was modified in such a manner so as to be applicable over a much wider range of input parameters. This revision makes it possible to incorporate immediately any new or revised reaction coefficients as fast as they are obtained from laboratory experiments.

A series of conferences has led to a firm plan to convert a Sheffield rotary contour gage to numerical control. Appropriate mathematical relationships have been derived and a preliminary EDPM program is being written to prepare magnetic tape input for the system.

Routine calculations were performed to determine optimum particle size and proportionate mix factors for several cylindrical Vipac fuel elements. Considerable effort has also been devoted to the study of particle spiraling, a phenomenon suspected of playing a fundamental role in the vibrational compaction of annular-shaped fuel elements.

In connection with the indexing of X-ray defraction powder patterns, a new master program called INDEX was written, compiled, debugged, and successfully used. INDEX makes a subroutine out of the cubic program. Work is under way to add the hexagonal-tetragonal program to INDEX as a subroutine. Eventually an orthorhombic subroutine will be added so that INDEX will handle any of the four types of crystal.

SUPPORTING FUNCTIONS

PRTR output for May was 1010 Mwd, for an experimental time efficiency of 58.2% and a plant efficiency of 50.3%. There were 13 operating periods during the month, five of which were terminated manually, seven were terminated by scrams (two were manual) and one operating period extended through month-end. The reactor operated at about 60 Mw until May 17, when the heat transfer flux limit was increased, which permitted raising the power level to 70 Mw.

Fuel exposure history at May 31 was:

Maximum UO_2 exposure/element	3406 Mwd/ton_U
Average UO_2 exposure/element	2412 Mwd/ton_U
Maximum Pu-Al exposure/element	93.4 Mwd
Average Pu-Al exposure/element	73.2 Mwd
Maximum Moxtyl exposure/element	71.0 Mwd (~1420 Mwd/ton_U)
Average Moxtyl exposure/element	41.4 Mwd (~828 Mwd/ton_U)

A total of 156 reactor outage hours were charged to repair work. Of this amount, 76 hours were used to complete the steam export line containment bellows replacement, 24 hours for heavy water leak repairs and 21 hours for moderator pump repairs and cleaning of the strainer on the moderator pump discharge line.

The PRCF was operated routinely for the entire month. Safety rod worth experiments and control rod calibration and safety system time response consumed the majority of operating time. Report HW-77607, "PRCF Light Water Dilution" was issued.

The installation of the Rupture Loop test section of the Fuel Element Rupture Test Facility (Project CAH-867) in B Cell is approximately 90% complete.

Gas-bearing blowers, Units 2 and 3, for the Gas Cooled Loop Facility (Project CAH-822) were shipped from England. Arrival here is scheduled for mid-June.

Total productive time for the month in Technical Shops was 28,170 hours. Distribution of time was as follows:

	Man-Hours	% of Total
N-Reactor Department	3 226	11.46
Irradiation Processing Department	6 986	24.79
Chemical Processing Department	353	1.25
Hanford Laboratories	17 605	62.50

[redacted]

Total productive time for Laboratory Maintenance was 21,600 hours, of 23,000 hours potentially available. Of the total productive time, 88.6% was expended in support of Hanford Laboratories components, with the remaining 11.4% used to provide service for other HAPO organizations.

Laboratory maintenance manpower utilization for May was as follows:

	Hours	Hours
A. Shop Work		4100
B. Maintenance		8300
1. Preventive Maintenance	1900	
2. Emergency or Unscheduled Maintenance	2200	
3. Normal Scheduled Maintenance	4200	
4. Overtime (included in above figures)	1100	
C. R&D Assistance		9200
		21,600 Hours

The heavy water loss during May was 1882 pounds valued at $26,120. Of this total, 1871 pounds ($25,971) resulted from PRTR operations and the balance (11 pounds, $150) from PRCF activities. Scrap generated during the month (2227 pounds) resulted in a charge of $3073 to operating cost.

At this time an underrun in 04 equipment expenditures is apparent; therefore, $200,000 will be transferred from the capital equipment budget to the research and development budget.

Activity in the area of visitor presentations included:

	Number of Visitors	
	In May	Since 6-13-62
Visitors Center	2215	44,062
Plant Tours	372	

DECLASSIFIED

[redacted]

1103432

[redacted] xxx DECLASSIFIED HW-77709

HAPO professional recruiting activity this month:

	Plant Visits	Offers Extended	Acceptances Received	Rejections Received	Open Offers at Month End
Ph. D.	7	15	4	8	12
BS/MS (Direct Placement)	-	12	10	8	9
BS/MS (Program)	-	3	15	53	34

Seven technical graduates were placed on permanent assignment. One addition to the program was made; its current strength is 34.

Construction of the Fuels Recycle Pilot Plant was started May 27 by the Halverson Construction Company. The contract completion date is June 15, 1965, and the amount of the contract is $4,448,000.

Authorized funds for ten active projects total $6,435,500. The total estimated cost of these projects is $10,680,000 of which $594,000 had been spent through April 30, 1963.

HM Parker

Manager, Hanford Laboratories

HM Parker:JEB:dph

SECRET

1103433

DECLASSIFIED

REACTOR AND FUELS LABORATORY MONTHLY REPORT

MAY 1963

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 02 PROGRAM

1. Metallic Fuel Development

Phosphorus Impurity in Uranium. Investigation was continued on the effects of phosphorus impurity on uranium structure, fabricability, properties, and fuel element quality.

Examination was completed of NIE fuel containing 135 and 170 ppm phosphorus in the as-extruded, beta heat-treated, and braze heat-affected conditions. Examination of NIE fuel with 300 ppm phosphorus is in process. No detrimental effects on clad-to-core bonds, or braze-to-core bonds were observed. Initial property data obtained from sections of this NIE fuel are shown in Table I. Phosphorus lowers the room temperature ductility and ultimate strength. At temperatures of 100 C and higher no differences due to phosphorus are noted. Examination was completed of NOE fuel containing 22 and 80 ppm phosphorus. No detrimental effects on clad-to-core bonds or braze-to-core bonds were observed.

Preparation of small dingots (about 5 pounds) with intended phosphorus levels of 1000, 1500, and 2000 ppm was requested from MCW. The average phosphorus contents from top and bottom samples were 549, 555, and 558 ppm, respectively, indicating a limit of phosphorus addition in the bomb reduction. This metal will be used for remelting and alloying. Attempts to prepare higher phosphorus contents by arc melting buttons of uranium with UP have resulted in structures composed of uranium, uranium-UP eutectic, and an additional compound that is apparently a uranium-oxygen-phosphorus phase. This phase evidently results from oxygen impurity in the UP and prevents formation of the complete eutectic.

Sulfur Impurity in Uranium. An alternate uranium scrap recovery process at NLO will use uranyl ammonium sulfate (UAS) to replace the uranyl ammonium phosphate (UAP) feed to the Winlo process. The effects of sulfur addition to uranium are being studied. An arc melt button was prepared using uranium and US to yield approximately 5000 ppm sulfur. Compound formation was observed that is presumed to be US. The structural features of this system will require additional compositions and heat treatments to define.

[redacted]

TABLE I

EFFECT OF PHOSPHORUS CONTENT ON TENSILE PROPERTIES OF URANIUM*

Coextrusion Number	Phosphorus Content, ppm	Test Temp., °C	0.2 Yield Strength 1000 psi	Ultimate Strength 1000 psi	Elongation-Percent in 1-inch gage
1151	Less than 50	Room temp.	56.1	132.2	17.5
		100	40.2	99.0	26.0
		200	43.4	72.3	27.0
1779	170	Room temp.	54.3	100.5	7.0
		100	44.0	97.8	23.0
		200	39.6	72.3	30.0
1861	296	Room temp.	46.3	108.1	10.0
		100			
		200			

*Samples prepared from NIE coextrusions in the beta heat-treated condition.

All standard Fe and Si addition.

[redacted]

Alternate Core Materials. N-outer tube stock with the core material containing 640 ppm Al and 400 ppm Fe was studied to determine the effects of heat treatment on the size and dispersion of UAl_2 and U_6Fe compounds. Particle density determined from electron micrographs of cathodically etched specimens and utilizing the Zeiss particle size analyzer to obtain particle size distributions in a computer program similar to that employed for analysis of fission gas porosity has yielded values of 1.5×10^{11} and 8.2×10^{11} particles/cc specimens beta treated and alpha aged and gamma treated and alpha aged, respectively. The etching procedures used for preparation of specimens for electron microscopy is continuing in attempts to arrive at a standard procedure yielding true structures in these dispersed phase alloys.

Additional materials being studied as dispersed phase systems include: (1) refinement of carbide size by control of freezing rate, (2) addition of phosphorus to a uranium - 2 w/o Zr composition, and (3) addition of aluminum to a uranium - 1 w/o Zr composition. These alloys have been prepared as arc melt buttons and have been given gamma phase solution treatments, quenched, and aged in the high alpha phase. The carbides in a 700 Al, 400 ppm Fe, approximately 500 ppm C alloy were reduced from 15-20 micron size to approximately 1-2 micron at a spacing of 10-15 micron by the rapid chilling of the arc melt button. A uniform dispersion of a compound believed to be UAl_2 was observed in the as-cast condition in a U + 1 w/o Zr + 200 ppm Fe - 350 ppm Al alloy. The addition of approximately 350 ppm phosphorus in a U - 1 w/o Zr alloy resulted in the formation of the U-UP eutectic at original dendrite boundaries with the UP particle size being one micron or less in the cast condition.

Closure and Joining. Corrosion tests have continued on Zircaloy-2 + beryllium and Zircaloy-2 + uranium coupons. After 150 hours in 360 C water, Zr-2 coupons containing up to 1 w/o uranium still appear glassy black, with only a slightly higher light adsorption measurement than Zircaloy-2 and a slightly higher weight gain. At this exposure the coupons above 30 w/o uranium have cracked and/or completely corroded into a sludge. Zircaloy + 5 Be coupons intentionally contaminated with oxygen and nitrogen have essentially decomposed after 550 hours in 360 C water. The variable beryllium content coupons show a linear weight gain and a slightly higher light adsorption than the previous readings at 300 hours of exposure.

A ceramic mold has been developed together with a suitable wax pattern for investment casting mechanical property specimens of brazing alloys. As these brazing alloys are very brittle, an investment casting technique will be necessary to provide the required shaped tensile and impact bars. The molds are made of equal parts of silica powder and pottery plaster, baked at 400 C for 12 hours prior to casting. They are quite fragile and break readily from the specimen bars. The wax is made of equal portions of ceresin and paraffin, cast into a brass pattern mold. The wax melts cleanly from the investment mold leaving a clean surface for casting.

Fuel Element Closure - Hot Head. Tooling for the 1200 KVA resistance welder at the Rocky Mountain Arsenal has been scoped and designed. Detailed drawings will be completed by June 1, and fabrication of the tooling will be complete by July 1.

Examination of N-Fuel Components. The post-irradiation examination of three N-outer fuel element components irradiated at near prototypic conditions in KER has been completed. The metallographic appearance of the cap-core bond of one closure did not appear typical or normal. This observed difference is believed to have arisen during fabrication and not as a consequence of irradiation. An adjacent control piece is being examined for comparison with the post-irradiation sample. In other respects, the N-outer fuel components continue to appear satisfactory in performances and behavior. Two inner components from the same irradiation test are now at Radiometallurgy for post-irradiation examinations.

High Temperature Swelling Performance of Uranium Fuel Elements. A charge of 8-12 inch long KSE-5 fuel elements, irradiated to an average exposure of 140 MWD/T at volume mean fuel temperatures up to 520 C, has been weighed in the KE basin for the determination of fuel swelling. The data extend our knowledge of swelling obtained by the previous irradiation testing of KSE-3 elements operated at volume mean fuel temperatures in the range 420-470 C. The KSE-5's operated with a peak power of 190 kw/ft and a heat flux greater than 1,000,000 Btu/hr-sq ft. Fuel swelling ranged from 0.78 v/o at the highest exposure, highest temperature element. The measured swelling values are approximately the same as those predicted from empirical expressions developed from the KSE-3 irradiations and verify that fuel swelling is ultimately limited by loop pressure (1600 psi coolant). Selected elements will be further evaluated at Radiometallurgy.

Fluted Fuel Element Irradiations. Two fluted, single-tube N-Reactor size fuel elements have completed three cycles of irradiation (600 MWD/T) in the ETR M-3 facility. These elements are being tested under conditions approximately equal to N-Reactor operating conditions. No appreciable volume change could be measured at the end of the third cycle. The elements are now undergoing their fourth cycle of irradiation.

DECLASSIFIED

The cold-water irradiation of three N-inner size fluted elements in the KE-Reactor was terminated at 2000 MWD/T exposure by the failure of one of the elements. These elements were being irradiated at a specific power of 75 kw/ft with a corresponding maximum metal temperature of 350 C and were to be given their final irradiation in the ETR-P-7 high temperature water loop. The nature of the failure is not known at this time.

MTR Irradiations of I&E Fuel Elements. Irradiation of four Hanford I&E fuel elements continued in the MTR GEH-4 loop facility to investigate the possible occurrence of large volume increases (grain boundary tearing) at intermediate irradiation temperatures. The calculated time average maximum metal fuel temperature during the irradiation is approximately 500 C, (not 400 C as previously reported). An axial temperature gradient from 300 to 500 C exists longitudinally along the fuel column which places portions of the fuel elements in the range of interest. Average exposure on the elements is now 1270 MWD/T.

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 were discharged at 1100 MWD/T. Initial examination of the samples from two of these capsules showed cladding strains of 1.7 and 2.6%, respectively. Each of the fuel rod samples with intentional internal striations in the cladding showed evidence of localized clad deformation associated with the initial defect. Based on the results from these fuel samples, the remaining 27 capsules were discharged at approximately 1800 MWD/T. Examination of the fuel samples from the first nine capsules is continuing.

N-Reactor Fuel Support Development. The program to develop a manufacturing procedure for N-inner supports was completed this month. In this project, a procedure for the fabrication of ductile Zircaloy-2 strip and a hot forming procedure for making the parts were developed. Using these processes, a total of 103,000 inner supports were made in the laboratory and supplied to NRD for use on N-fuels.

Fuel Element Support Fabrication. Metal Fabrication Development has undertaken fabrication of further test lots of N-outer fuel element supports to permit the N-fuel production facility to continue to process N-outer fuel elements. Supports are being fabricated at the rate of 3000 per day for the first commitment of 30,000 parts. Eleven thousand supports have been produced since Thursday, May 16.

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 cycle consists of passing steel support shoes 40 times over a single path on the process tube, then autoclaving the tube in 400 C, 1500 psi steam for 14 days to give the corrosion product buildup on the bare Zr-2 equivalent to that expected to form in-reactor between the charge and discharge of the fuel elements.

After the fifth autoclave cycle, as in the third and fourth cycles, a total of 50 passes with four different shoes were made with no signs of scratching the autoclave film. Steel from the shoes was deposited on the re-autoclaved film. It was necessary to intentionally initiate the scratching of the process tube in order to continue with the original purpose of the cyclic scratching-autoclave test. Six series of 40 passes, with autoclaving between each series, have now been completed on the same region of the process tube section. The scratch depth now varies from 0.0012 to 0.0022 inch with an average depth over the entire length of the test section being 0.0016 inch. This rate of wear extrapolates to 0.015 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, 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 test. In-reactor the cold-worked surface of the tube would not be annealed as in the autoclave, but there would be a corrosion film formed.

To determine how much of the damage is caused by removal of the corrosion film, two new series of tests are being started. In the first test, 50 passes are made on a process tube section which is then annealed at 400 C for 14 days in a vacuum. The cycle will be repeated at least six times and compared with the cyclic scratching-autoclave test. In the second test, 500 passes are made on a process tube section which is also annealed at 400 C for 14 days in vacuum. This test will be repeated four times until a total of 2000 passes are made on the one tube section. Results from the latter test can also be compared with the previous 2000-pass test to see the effect of 400 C recovery of the cold work on the tube surface.

Analysis of Fuel Element End Caps. The difference in thermal expansion between fuel and end cap materials in metallic fuel elements is a potential cause of fuel element failures. Thickening the cladding near the end cap or shaping the end cap are possible methods of making a transition which will reduce the discontinuity stresses encountered in the end cap region. Studies previously reported have been extended to include a specific design for an N-outer fuel element. The transition zone has a linear thickening of the cladding 300 mils long which makes a 17.5° angle with the outer cladding and 21.0° angle with the inner cladding. The bending and shear stresses within the transition have been reduced by over a factor of four. The report describing this analysis and analysis of various shaped transitions has been written and is being reviewed for publication. The method of analysis was also extended to apply to the particular geometries encountered in the hot-headed closure techniques. Case studies run on geometries representative of those obtained by this fabrication method showed bending stresses and shear stresses within the transition region were two to three times less than those encountered in closures without transitions.

Vibrational Studies. A theoretical study has been completed on the effect of process tube vibrations and their relation to fuel element support design. Two programs were developed: one is a two-component model for the vibration of a fuel assembly, and the second calculated the spring constants of various shaped supports. The spring constant program only applies to non-sized supports. The vibrational program can be used with either calculated or measured values for the spring support constants. For measured values for spring constants and damping obtained on N-fuel elements, the lowest resonant frequencies ranged from 46 to 62 cycles per second. A document describing these programs (HW-77515) has been published.

Fuel Element Data System. An IBM data file for fuel test irradiation data has been developed and the program which controls the fuel data file has been completely debugged. The file now contains data for 147 N-inners, 135 N-outers, 41 capsules, and 12 KSE-3 elements. Additional programs to analyze the fuel performance data in terms of fabrication and test variables are being developed.

Fabrication of Crud Probe Stock. A long life, constant power heat source will be fabricated from a Th - 1½ w/o U (fully enriched) - 1.0 w/o Zr alloy for the purpose of studying the variation in heat transfer with film buildup on a Zircaloy-2 surface during irradiation. Double vacuum arc melting of the 50-lb pilot ingot of thorium - 1½ w/o natural U - 1 w/o Zr has been completed. Extrusion of the primary ingot into an electrode for remelting has proved to be a satisfactory procedure for this material.

Coextrusion of 2S - Aluminum Clad Li-Al. To date, three coextrusions have been made in an attempt to provide material for diffusion couples to study the rate of Li diffusion into a 2S aluminum cladding and to examine the feasibility of fabricating target fuel by coextruding a clad Li-Al target material.

The first billet contained a 1½% Li - Al core which was in the as-cast condition, while the following two billets contained 1½% Li - Al cores which were cast and pre-extruded at 350 C at a reduction of 3.24/1. All coextrusion billets were extruded at a reduction of 15.4/1 at 350 C. The interface with the as-cast core was very rough and irregular. The interfaces found with pre-extruded cores were smoother but somewhat out of round. In all three coextrusions the cast structure was still in the core material after coextrusion. An additional billet is being prepared in which a 1% Li - Al core will be pre-extruded cold at a reduction of 3.62/1 and recrystallized before coextrusion in an effort to further improve the interface of the coextrusion.

To obtain a bond between core and clad, it was necessary to thoroughly mechanically clean both the clad surface and the core surface just prior to assembly and electron beam welding.

Target Element Development. Several intensive heat treatment tests on aluminum-lithium alloys have been carried out to determine minimum recrystallization times and temperatures. Two worked conditions were investigated at two different times and five different temperatures. Hardness measurements showed a break after annealing at 400 C for one hour and 450 C for 30 minutes. Metallographic results are not available at this time.

Lithium-Aluminum Target Elements. Sixteen Zircaloy-4 clad experimental lithium-aluminum target assemblies have been fabricated for irradiation testing in KER Loops 1 and 2. Each test charge will consist of eight target elements and each will contain both 1 and 2 w/o lithium alloys canned in both anodized and nonanodized aluminum cans.

A defect target element assembly is currently under test in an ex-reactor water loop under simulated N-Reactor flow conditions. The corrosion of Al is being followed by periodically analyzing loop water samples for aluminum. This test is intended to give an estimate of the effectiveness of monitoring the irradiation test loop

DECLASSIFIED

water for aluminum activity for detecting a target element rupture.

The transport of lithium in the alpha phase region of the aluminum-lithium system under isothermal conditions is being investigated. Diffusion couples have been prepared by cold-pressure welding 1.3 w/o Li-Al alloy to aluminum. Tests at 600 C indicate that sufficient Li is vaporized to grossly attack the quartz capsules in which testing is conducted. Tests at 300 C have shown no evidence of Li vaporization after 14 days at temperature. The Li gradient in the couples will be determined after annealing the test specimen at 300 C for 200 days.

Fueled-Web Honeycomb Element. A honeycomb structure in which the web is made of Zircaloy-2 clad uranium has been successfully fabricated by coextrusion. The cross section of the structure consists of a hexagonal array of 61 cells enclosed by Zircaloy-2 one inch in outside diameter. The approximately 1/16-inch diameter holes are surrounded by the 0.038-inch total thickness web. The web is made up of 0.007" Zr-2, 0.010" U, 0.004" Zr-2, 0.010" U, and 0.007" Zr-2. The web on both ends of a sample section 1.25 inches long has been successfully closed by induction brazing with Zr - 5 w/o Be alloy.

The composite coextrusion billet was made up of 61 coextruded hexagonal rods sealed in vacuum in Zircaloy-2 and canned in copper. The rods consisted of outer Zr-2 cladding, a layer of uranium, a ring of Zr-2, and a uranium core. The final coextrusion completely bonded all parts of the billet together. The inner of each cell was removed by first stopping off the uranium in the web, then etching in hot HCl.

Vanadium Wire Drawing. Reductions in the diameter of vanadium rod, containing 0.5% Co, were accomplished using the asphalt lubricant. This rod is eventually to be reduced to 10-mil wire. However, after a few reductions were made, serious defects in the vanadium rod began to appear. These defects consisted of deep gouges, most likely produced by some gripping mechanism used in the original rod forming operation. In some subsequent rod sizing operation the very ductile vanadium was caused to flow over these gouges and hide them. The wire drawing of vanadium plus 0.5% Co will continue when defect-free rod is obtained.

Rhenium Wire Drawing. An attempt was made to draw some 0.020-inch rhenium wire down to 0.010 inch. The wire broke on the first pass, a reduction of 0.001 inch. Macroscopic examination of the wire showed that its surface was extremely rough and contained many cracks. Further inquiry into the history of this wire determined that it was highly work hardened and that the maximum allowable amount of cold work (10% had undoubtedly been exceeded. An attempt is being made to improve the surface of the wire and anneal it at 1700 to 1800 C.

<u>PRTR Rupture Loop</u>. The thermal hydraulic characteristics of an in-reactor assembly which included basket, NPR fuel element, and cutter device were determined experimentally. A test of the cutter device in an out-of-reactor high temperature-high pressure loop revealed that the operation of the device is affected by extended exposure to high temperature water. The design will be changed to give improved high temperature behavior of the hydraulic cylinders.

<u>GEH-10-64, 65, 66 - Thorium Alloy</u>. Three Zircaloy-2 clad thorium - 2.35 w/o U-235 - 1.0 w/o Zr fuel elements complete their first cycle of irradiation in the ETR-P7 Loop. The fuel achieved a calculated maximum specific power of 214 kw/ft, a maximum heat flux of 1.07 x 10^6 Btu/hr-ft^2, and operated with a maximum core temperature of 603 C. Fuel element weight measurements made at the end of the cycle indicated that no appreciable swelling had occurred. Exposure on the test elements at the end of the first cycle of irradiation is 1100 MWD/T.

2. <u>Corrosion and Water Quality Studies</u>

<u>Water Contents of Oxide Films on Autoclaved and Anodized C-64 Aluminum</u>. C-64 aluminum alloy tube sections were anodized or autoclaved. The oxide films were isolated by dissolution of the metal substrate in CH_3OH-I_2 solution. The isolated oxides were then dehydrated at 1300 C (2372 F) for one hour.

Water content for oxides removed from samples autoclaved in 170 C (338 F) water for 66 hours, autoclaved in 250 C (482 F) water for 51 hours and as anodized in chromic acid were 23.5, 18.4 and 4.0%, respectively.

Theoretical water contents for mono- and trihydrates are 15 and 34%, respectively. Thus, it appears that the 170 C (338 F) autoclaved film is a mixture of mono- and trihydrate; the 250 C (482 F) autoclaved film is largely monohydrate, while the anodized film is largely anhydrous.

<u>Corrosion Rates in High Alum (18 ppm) and Low Alum (7.5 ppm) Water</u>. Preliminary results have been obtained from the half-plant alum test at F-Reactor (PT-549) with alum feeds of 7.5 ppm (average) versus

[redacted] [redacted]

18 ppm. Visual examinations of fuel element surfaces from initial discharges show the same frequency of ledge and groove corrosion for both sides of the reactor. Coupon samples corroded in each type of water for five weeks at exactly comparable temperatures show equal penetrations of X-8001 aluminum and A212 carbon steel. Samples of 1100 alloy, however, showed corrosion penetrations 1.6 times as great in 18 ppm alum-treated process water as in the 7.5 ppm treated water. Corrosion penetration of 6061 aluminum alloy was 1.3 times as great in 18 ppm alum water as in 7.5 ppm water. Film weights for all materials approximately follow corrosion penetrations in both types of water.

Corrosion of Monel Resistance Temperature Detectors. Pitting attack of the RTD's at 105-KE and 105-KW have caused some concern - the attack is localized near the crevice and may be due to galvanic attack in acid solution, resulting from incomplete rinsing after decontamination. A testing program has been started to evaluate the effects of residual decontaminant (sulfamic acid) on the RTD Money and Inconel sheaths under conditions comparable to those expected during reactor operation.

Samples of Monel-sheathed resistance temperature detectors showed no evidence of pitting attack after 27 hours at 70 to 80 C in 6 ounce/gallon solutions of either Turco 4306-C or Wyandotte-5061. Assembled samples (with galvanic contact to stainless steel) did show slight local etching in areas exposed to saturated vapors from the Turco 4306-C solution.

The pH of rinse water "leaked" from KW RTD wells after reactor decontamination was measured with litmus paper. The measured pH values after 10 minutes of flushing were 2.0, indicating that flushing had not completely removed the decontaminant chemical. Additional rinsing with conventional reactor cooling for 30 minutes did remove the decontaminant, as shown by pH values of 6.6.

Decontamination of K-1 and K-2. The K-1 and K-2 loops were decontaminated to reduce activity levels and facilitate tube replacement operations in 105-KE. The K-2 Loop (stainless steel) was decontaminated using a two-step recirculation procedure: alkaline permanganate solution at 105 C for 90 minutes followed by sulfamic acid solution at 70 C for 90 minutes. Both solutions used were proprietary compounds.

The K-1 Loop was decontaminated using a single-pass procedure with a 5% solution of inhibited ammonium citrate at 75 C (rear face temperature). The solution was used for approximately one hour at a flow rate of 5-10 gpm.

The decontaminations reduced the activities at front and rear face for both tubes to acceptably low levels. The activity level on the rear face was slightly higher because of external contamination. Prior to shutdown a rupture had occurred in a tube at a higher elevation (3565). Leakage from this tube contaminated the external surfaces of a number of the other tubes including K-1 (2160) and K-2 (2864).

Corrosion Under Heat Transfer Conditions in Lithiated Water pH 10. A test was completed to determine corrosion behavior associated with N-Reactor fuel self-supports in lithiated water at pH 10 under conditions similar to those expected in N-Reactor. A Zr-2 clad heater rod with four self-supports welded to it simulated the N-Reactor fuel element. After 146 days in water at 315 C and a heat flux of 280,000 Btu/ft^2-hr, the element was examined. Accelerated corrosion had occurred on both the support and the jacket at each of the eight crevices between the self-support and the jacket; the maximum penetration was 6 mils with an average depth of 1-2 mils. The corrosion seemed general rather than pitting type of attack and covered about half of the area under the supports. The crevices were filled with oxide over most of the region of attack. From the observations, it appeared that the formation of oxide tended to force the support and cladding apart.

Massive hydriding, such as might have been expected from such extensive corrosion, was not found. Hydride concentrations were no greater than 100 ppm and were no greater near the corrosion than elsewhere.

Evaluation of Secondary System Descaling Procedures. Some evaluation tests were completed on a new proprietary compound for cleaning N-Reactor secondary system prior to startup. This compound removed the mill-scale but was not satisfactory for removing the lacquer.

Fretting Tests on N-Reactor Fuel Elements. Additional fretting tests were completed on the N-Reactor fuel element with the high support spring constants. No fretting was noted after 23 days of exposure in TF-7 at 530 F, pH 10 lithiated water with external vibration. During the next operating period the vibration will be changed to a lower frequency (from 54 to 40 cycles/sec) and a greater deflection (from 1.8 to 3.5 mils).

Testing of Defected Target Slugs. During the past month one predefected target slug (Zr-2 clad, lithium-aluminum alloy core) was charged into TF-7 to determine rate of leaching or dissolution and to evaluate ease of detection of rupture.

DECLASSIFIED

Evaluation of NH_4OH to Control pH in N-Reactor Primary Coolant. The injection of NH_4OH into KER-1 Loop was initiated last month in the first of a series of in-reactor tests to determine the feasibility of using NH_4OH rather than LiOH to adjust the pH of the N-Reactor primary coolant. The operation continued at high temperature (260-280 C) during the month of April, except for one reactor shutdown. The water quality and crud deposition were about the same as with lithiated systems. On April 30, as a result of leaks, the loop temperature was reduced (60-90 C); the loop operated on feed and bleed without venting until May 8, at which time dissolved gas concentration increased to a point where it was considered desirable to vent the loop. This additional gas was caused primarily by leakage of air into the makeup solution tank. In addition, the concentration of ammonia in the makeup tank was not controlled. When the solution was prepared originally, it was not agitated and therefore was not homogeneous. As the tank was emptied, the feed solution became more and more concentrated in NH_4OH.

During operation at low temperature the water quality and crud release rates appeared to be better in the ammoniated system than in lithiated systems, primarily because of the formation of sufficient hydrazine to prevent accumulation of oxygen in the system. In lithiated systems, unless hydrazine is added, the oxygen formed from radiolytic decomposition of the water, oxidizes the protective magnetite to a less protective form (Fe_2O_3) which is released as crud into the system.

One of the main difficulties in using NH_4OH in a reactor primary system may be the radiolytic and/or thermal decomposition. Results from the last test indicate about 50% of the injected ammonia was decomposed and/or adsorbed on the piping walls. Samples of gas were taken to determine the concentration of the decomposition products, but analyses are not complete.

3. Gas-Atmosphere Studies

Graphite Burnout Monitoring. Small graphite burnout monitors were discharged from channel 3478 at D-Reactor. The test covered the period from January 3, 1963 to May 1, 1963. The profile showed two peaks at the usual positions of 80 and 168 inches into the graphite stack. The height of the first peak was 23% per 1000 operating days (%/KOD), and the second was about 1.5%/KOD.

Monitors were also discharged from channel 2577 at H-Reactor. The test was from August 2, 1962 to May 3, 1963. The results revealed high rates from 50 to 92 inches into the graphite stack. The rates in this area ranged between 14 and 25%/KOD; however, the monitor farthest upstream at 46 inches measured 66%/KOD. There were no visible indications, such as deep pits or abrasions, to suggest unusual handling of this sample. Furthermore, all the monitors were constructed from CSF graphite, which is a high-purity, reactor-grade material, and they were used in a previous test in channel 2577 where no large differences in rates appeared. The only distinctive appearance of the sample was a definite taper. In the laboratory this geometry results when a highly reactive gas impinges on one end of a piece of graphite. It seems likely, then, that this monitor gave a reliable weight loss and that a significant quantity of air entered the reactor atmosphere. Gas analyses and reactor-gas losses are being checked for this possibility.

<u>Graphite-Zirconium Compatibility in N-Reactor.</u> The results of the N-Reactor atmosphere computer program (cf. HW-77397 A) have been correlated by means of an empirical expression which relates maximum graphite burnout rate, R(%/KOD), maximum graphite temperature, T_{max}, initial water vapor partial pressure, p_{H_2O}(mm), and initial hydrogen partial pressure, p_{H_2} (mm).

$$R = (T_{max} - T_o)\, S\, p_{H_2O}$$

$$\text{where:}\quad T_o = \frac{\log (p_{H_2}) + 19.5}{0.0286} \qquad (0.076 \leq p_{H_2} \leq 7.6)$$

$$S = \frac{\log (p_{H_2}) + 6.2}{208} \qquad (0.076 \leq p_{H_2} \leq 0.76)$$

$$\text{and}\quad S = 0.030 \qquad (0.76 \leq p_{H_2} \leq 7.6)$$

The equation applies in the T_{max} range from 750 to 820 C and for volume flows to the reactor in the range 400 to 1000 cfm. As emphasized in last month's report the numerical values obtained from the program can only be considered as order of magnitude estimates until additional experimental results are obtained to more accurately fix the values of the various parameters that go into the computation.

<u>Kinetics of the Water Gas Shift Reaction.</u> It was reported in March 1963 that there was little reaction observed between H_2 and CO as a

DECLASSIFIED

He-CO-H_2 mixture passed over an eight-inch graphite bar, 1-7/8 inch in diameter, in a two-inch ID quartz tube ($H_2 + CO \longrightarrow H_2O + C$). The water gas shift reaction ($H_2O + CO \rightleftarrows CO_2 + H_2$) was investigated in the same apparatus by observing how much water is lost when CO is added to an H_2O-He mixture, and again little reaction was observed. The apparatus has been subsequently changed so that the gas instead of passing over the graphite flowed through a graphite thimble 13/16-inch in thickness. This procedure increases the graphite surface area in contact with the gas. Six runs have been completed to date, and significant reactions between CO and H_2O were observed. The reaction was slow at 400 C (752 F) but increased rapidly above 500 C (932 F). This sequence of experiments indicates the homogenous gas phase reaction is much slower than the surface catalyzed reaction which would imply the shift reaction would be an important source of water loss in graphite reactor tube block, but may not be as significant in the reactor gas channels, unless radiation accelerates the homogenous reaction. Additional runs are planned on the reverse water gas reaction.

4. <u>Process Tube Development</u>

<u>Stress Rupture Testing of Reactor Pressure Tubes</u>. A section of 30% cold worked N-Reactor pressure tubing has been subjected to stress rupture test without benefit of previous autoclave treatment. Its performance was greatly inferior to similar tubing tested after having been autoclaved. A similar tendency had been noted earlier in the performance of 30% cold worked KER tubing tested in the unautoclaved condition. For example, in 30% cold worked tubing at a hoop stress of 57,000 psi and a temperature of 300 C (575 F) the primary creep (%), minimum creep rate (%/hr) and rupture time (hrs), respectively, were 0.25, 0.0002 and 4926 for autocalved N-Reactor tubing; 1.2, 0.005 and 209 for unautoclaved N-Reactor tubing; and 0.8, 0.0014 and 1174 for unautoclaved KER tubing. Although no differences in microstructure have been noted, it appears that some sort of aging phenomenon accompanies the autoclave treatment. Additional specimens in both the autoclaved and unautoclaved condition will be tested to confirm these tentative results.

5. <u>Thermal Hydraulic Studies</u>

<u>Present Reactor Studies</u>. Response times were determined for two thermocouple assemblies under study for use as outlet water temperature monitors at H-Reactor. Each assembly consisted of a copper-constantan thermocouple inserted in a well, ½-inch long x 7/32-inch OD x 5/32-inch ID. The response time of each thermocouple assembly was determined with the assembly immersed suddenly into boiling water

and also when the assembly was installed in a flow system where the temperature of the flowing water was changed suddenly. In each case temperature indications were recorded on high speed recorders.

A time of four seconds to respond to 63% of a sudden change in temperature has been deemed desirable in order for the thermocouple to give adequate information following a reactor power surge or a flow decrease. The following results, reported as time to reach 63% of the difference between the initial and final temperatures as determined from a bare wire reference thermocouple, were obtained.

T/C #1	Time, sec (For boiling water case)	Time, sec (For flowing water case)
1	6.9	5.14
2	5.94	4.32

In the case of T/C #1, it was found that the thermocouple wires were twisted together above the intended junction and were forming a junction which would be somewhat insulated from the wall of the well in comparison to the case for the intended junction.

N-Reactor Studies. Efforts were made to try to predict the probability and intensity of boiling under the devices used to support the fuel pieces within the process tube in N-Reactor. The interest in this lies in the fact that water quality studies have indicated that if LiOH (the pH control media for N-Reactor primary coolant) concentrates under a self-support, as might occur with boiling, severe corrosion of the Zircaloy jacket can occur.

One method of calculation has been to modify correlations existing in the literature, which predict boiling on an unobstructed surface, to the case of N-fuel supports by using laboratory data of boiling under supports. This method has some uncertainties since the laboratory data were obtained for the present reactors and the limited data for unobstructed surfaces do not check the correlations. Another calculation being tried is to calculate the water temperature under the supports assuming a decreased flow rate at that point and adjusting the results to fit results obtained with visual studies performed for present reactor supports. An uncertainty in this case, however, is the effect of the different support shape used on N-Reactor fuel pieces as compared to those used in the present reactors. Until experimental data are obtained for the N-Reactor supports, it appears that the accuracy of the calculations will be limited.

In other studies for the N-Reactor, equations were derived to fit the boiling burnout data obtained for the outer and center flow annuli. These equations are:

For the outer flow annulus:

$$\frac{(Q/A)BO}{10^6} = 0.695 + 0.0123 \left(\frac{G}{10^6}\right)^{1.73} - 3.16 \left(\frac{G}{10^6}\right)\left(\frac{\Delta H}{1000}\right)$$

and for the center flow annulus:

$$\frac{(Q/A)BO}{10^6} = 1.640 - 0.946 \left(\frac{G}{10^6}\right)^{-0.5} - \left[7.0 \left(\frac{G}{10^6}\right)^{+0.5} - 5.0\right] \left(\frac{\Delta H}{1000}\right)$$

where (Q/A)BO = burnout heat flux, B/hr-sq ft
G = coolant mass velocity, lb/hr-sq ft
ΔH = local enthalpy minus saturation enthalpy, B/lb.

Nearly all of the experimental data at 1500 psig fall within ± 25% of the lines defined by these equations. The data for the equations range from 20 F subcooled to 30% outlet quality for a flow rate of 500,000 lb/hr-sq ft, and from 40 F subcooled to 5% outlet quality for a flow rate of 5,000,000 lb/hr-sq ft. These equations are not meant to be general correlations but are mainly for use in the application of these data to the N-Reactor.

6. Shielding Studies

MAC Code Development. MAC code has been run several times during the month for two purposes: (a) checkout of several changes to the gamma subroutines, and (b) for an analysis of water shielding for the FSPPR concept.

Several changes have been adopted in the MAC gamma subroutine. The code now produces consistent gamma ray calculations and eliminates the problem pointed out by U. Canali, EURATOM, Italy.

Progress is being made with BARNS Code. The unit weight deck has been checked out. This code is being modified to supply cross sections ready for MAC input.

Fast Neutron Spectrometer. A fast neutron spectrometer being procured for core measurements may be useful for experimental shielding studies. A He^3 spectrometer has been ordered by Nucleonic Instrumentation

Operation, and information is being sought concerning a neutron sensitive plastic scintallation pulse height (proton recoil) device. These spectrometer tubes will be coupled to an available multichannel analyzer to complete the system.

N-Reactor Shield Evaluation Proposal. A proposal was prepared for NRD - Research and Engineering suggesting an experimental evaluation of the N-Reactor shields. The proposed experiments consist of a series of neutron and gamma ray flux and energy measurements to be taken during the initial physics and startup tests of the N-Reactor.

The measurements are proposed to be taken in a shield plug facility which penetrates the N-Reactor top primary shield. The data taken from the shield plug (supplemented by analytical determinations) will allow:

(a) Evaluation of various N-Reactor shield surfaces (by application of the analytical methods developed to fit the experimental shield plug data),

(b) Evaluation of the maximum power level at which the reactor, from a shielding standpoint, can operate,

(c) Prediction of shield deterioration (loss of water and/or radiation damage, and

(d) Assessment of possible needs and methods for shield protection mechanisms and procedures.

7. Reactor Technology

Large Fast Reactor Study - Thorium Case. Exploratory core design calculations were completed for a 3000 Mw, 5000 liter PuO_2-ThO_2 fueled reactor. A 16-energy group diffusion theory model was used with a spherical geometry representation for core and blanket. An investigation of sodium reactivity coefficients was also carried out as a function of coolant volume fraction. The results of these initial calculations are given in the following table:

Core Composition	Blanket Composition	PuO_2 Enrichment*	k	k (void)
23 v/o Fe 47 v/o Fuel 30 v/o Na	15 v/o Fe 70 v/o ThO_2 15 v/o Na	14.25% 13.5%	1.002 0.965	1.021 0.984
23 v/o Fe 37 v/o Fuel 40 v/o Na	Same	16.5% 16.0%	1.052 1.031	1.075 1.054
23 v/o Fe 27 v/o Fuel 50 v/o Na	Same	19.0% 18.0%	1.086 1.048	1.107 1.069

*Pu Isotopic composition:

Pu-239	57%
Pu-240	29
Pu-241	10
Pu-242	4

As anticipated for this core size, a positive sodium density coefficient is obtained. In all cases an increase in reactivity of close to 2% $\Delta k/k$ is observed upon complete loss of sodium from the core. It thus appears that for L/D ratios near unity the sodium coefficient is relatively insensitive to the sodium volume fraction. Therefore, L/D ratios much different from unity or some other alternate method for reducing the sodium coefficient must be devised for this reactor design.

A potentially higher U-232 equilibrium concentration than is the case for thermal breeders is indicated in a study carried out by NDA on fast Th-233 systems(1). This calculation is derived from effective cross section ratios which are estimates based upon limited data. Because of the difference in fuel and the importance of U-232 to fuel handling, an independent estimate of this problem is being made.

Initial surveys of possible power recovery cycles and correspondence with the Large Steam Turbine Department indicate that it will be most expedient to make minor revisions in the steam cycle proposed by APED for the Sodium Modular Reactor to adapt it to the conditions for power recovery cases to be included in this study. The SMR steam cycle has a peak temperature of 900 F vs a peak of 950 F currently envisioned for this reactor with power recovery.

(1) A.J. Goldman, et al., "A Feasibility Study of Fast U-233 - Th Breeder Reactors," NDA 2134-3, October 1960.

8. Graphite Studies

N-Reactor Graphite Irradiations. Irradiations of N-Reactor graphite continue to proceed satisfactorily in the GETR. The H-6-2 and H-4-3 capsules successfully completed another cycle of irradiation on May 5.

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 entire PRTR fuel loading was examined in the basin and selected elements were examined in the Fuel Element Examination Facility (FEEF). The basin examination consisted of a complete visual inspection of the elements and ring gaging the diameter over the gussets. As a result of the basin examination, two elements were rejected for broken rod wire wraps and six elements were examined more thoroughly in the FEEF. One of these six was tentatively rejected because of a questionable mark on the outer surface of one rod. A total of 10 fuel examinations were performed in the FEEF. In general, the condition of the fuel elements appears good; however, the following significant observations were made:

(1) Many elements have accumulated scratches on their external surfaces. The number of scratches is not necessarily related to the age of the element.

(2) Circumferential band type bundle wraps have been snagged and displaced on many elements.

(3) Rod wire wraps have loosened on some of the Al-Pu spikes as a result of rod shortening, thus indicating that core-clad interaction has occurred. No fretting or wear caused by loose wires has been observed. Apparently the wires are tight during operation.

(4) Corrosion of spacing wire spot fusion welds has occurred on some plutonium-containing elements.

(5) The surfaces of some fuel rods are discolored. The degree of discoloration seems to increase with exposure. This discoloration is probably caused by the surface staining which results from inadequate rinsing after the etching treatment.

(6) The clip-on wear pads on the UO_2-PuO_2 elements are performing satisfactorily.

(7) Several elements had bent hanger pins which required replacing.

Post-Irradiation Examination of PRTR Fuel Elements. The individual fuel rods from a vibrationally compacted UO_2-PuO_2, PRTR Mark-I element that failed in the PRTR were examined in detail. Three of the rods in the element had failed. All three failures occurred in the heat affected zone of the top end cap weld (this area does not have UO_2-PuO_2 fuel adjacent to the cladding) and appear to have been caused by severe localized corrosion. Undetectable amounts of particulates were released to the primary coolant following the failure of this element in the PRTR. The defect mechanism proceeded rapidly since the element exposure was only 106 MWD/T and the three fuel rods apparently failed at approximately the same time. A post-irradiation examination is being conducted to determine the cause of failure. The two most likely possibilities appear to be contamination of the Zircaloy during welding or the presence of some impurity in the fuel material which caused severe corrosion from the inside. (All rods were loaded from the same batch of UO_2-PuO_2.) Analyses of the fuel material and corrosion products as well as metallographic examination of the defect area are in progress.

Irradiation Testing of Prototypic EBWR Fuel Rods. Four vibrationally compacted capsules (GEH-14-421, -422, -423, and -424) containing UO_2 (depleted) - 2.5 w/o PuO_2 fuel were inserted in the ETR for irradiation in Cycle 55 which started May 13, 1963. These short simulated EBWR fuel rods are being tested to determine the irradiation behavior (in particular, the release of fission gas) of UO_2-PuO_2 fuels of the types now under consideration for EBWR fuel elements. The plutonium-bearing EBWR fuel elements are to be fabricated by the Vipac (vibrational compaction) process. The exact composition has not been established, but the most recent information indicates that the fuel composition may be lowered from UO_2 (depleted) - 2.5 w/o PuO_2 to UO_2 (depleted) - 1.5 w/o PuO_2.

Recycled PRTR Fuel. The second UO_2-PuO_2 fuel cluster containing recycled PRTR plutonium was completed; the first cluster was charged into the PRTR. Five clusters with Zr-Co wire wraps for flux monitoring were completed; each of the clusters contained three rods with sintered UO_2-PuO_2 pellets and the remaining rods were swage or vibratory compacted.

Nondestructive Test for Plutonium Segregation. Gamma scan measurements of Pu concentration correlated with calibrated UO_2-PuO_2 rods in the range 0.1 to 10% Pu. A program was initiated to gamma scan all PRTR fuel rods before cluster assembly. Further tests are planned to determine gamma scan sensitivity to radial Pu distribution.

Powder Loading Studies. A uniformity of Pu distribution of ± 0.1 w/o was obtained in "bottle loaded," vibrationally compacted, UO_2 - 0.48 PuO_2 fuel rods of PRTR size. The bottled powder was blended for one hour before loading. Three PRTR fuel clusters will be made by these techniques to determine statistically the reliability of bottle loading.

Fuel Element Design Studies. A UO_2, PRTR Mark-I fuel element with ring-type, continuous contact surfaces on each end bracket has been exposed for 77 days in the TF-7 facility at PRTR coolant conditions. Only superficial wear has occurred on the ring contact surface of the bottom end bracket.

Fuel Element Rupture Studies. A program letter was issued proposing the use of the PRTR Rupture Loop for evaluating both operating and rupture behavior of PRTR and exploratory PRP fuel elements.

Hazards Analysis. The Facility Hazards Analysis for the Plutonium Fabrication Pilot Plant (308 Building), HW-74007-RD, PartI, was issued.

Irradiation of Uranium-Plutonium Oxide. Analytical samples from selected areas of irradiated $(U,Pu)O_2$ fuel capsules (GEH-14-86) were obtained by using an ultrasonic microsampling unit in conjunction with high resolution film plate autoradiographs. Specimens have been removed from 10 irradiated UO_2-PuO_2 capsules and from one irradiated Al-Pu-Ni alloy capsule.

Fast Reactor-Thermal Reactor Exchange Element. The designs of both the Fermi-PRTR proposed exchange element and the special PRTR process tube in which it will be irradiated were submitted for nuclear safety review. Both components were approved for present reactor operating levels.

Pneumatic Impaction of Plutonium-Containing Materials. Pneumatic impaction equipment and techniques being developed for fabrication of EBWR fuel material were used to compact a sample of metallic plutonium powder for Plutonium Metallurgy Development. The mechanical properties of the impacted specimen will be compared with those of similar specimens fabricated by conventional pressing and sintering techniques.

The pneumatic impaction technique was used to denisfy 1100 grams of PuO_2 at one time. Density achieved was 10.89 g/cc (95.1% TD).

Remote Fuel Element Closure. The welding turntable in the remote fabrication facility was installed; it operated satisfactorily.

Fuel Refurbishing. Equipment for quickly and reliably positioning, tightening, and securing a cluster fuel element wire spacing member remotely, under approximately 10 feet of water, was developed and fabricated.

Fuel Element Rejuvenation. The rejuvenation fuel element (GEH-4-81) is ready for return to the MTR. Rejuvenation was accomplished by adding enriched UO_2 to a fuel rod from the fuel assembly which had been irradiated in January in the MTR. The UO_2 was packed into a ZrO_2 tube in the center of the fuel rod. A new cap was then welded to the rod and the rod welded into a cruciform containing three other previously assembled fuel rods. The radiation level, measured approximately two inches from the rod end, was 500 R. The surface heat flux of the element during the second cycle will be approximately twice that of the original.

Magnetic Force Resistance Welding Studies. The welding chamber was decontaminated after welding beryllium. A second chamber was ordered. It is approximately three times as large and will provide improved access to the working parts and greater freedom of movement within the chamber.

Corrosion and Water Quality Studies

Corrosion of Various Stainless Steel and Nickel Base Alloys. Descaled corrosion rates have been obtained on various stainless steel and nickel base alloys exposed 100 days in 550 C (1022 F), 3000 psi deoxygenated water. Samples of the various alloys were descaled electrolytically with a molten caustic bath and weighed. The corrosion weight losses for each alloy were converted into mils penetration. The alloys may be grouped into three general orders of increasing corrosion. The most corrosion resistant group (< 0.1 mil) includes Incoloy, Inconel "X", AISI 446 stainless steel, Hastelloy "X", Nichrome VI, PDRL-102, and R-27. The next group (0.1 to 0.2 mil corrosion) includes two experimental Fe, Cr, Al alloys, and 17-4 pH stainless steel. The least resistant group of alloys (> 0.2 mil) in the test includes two heats of AISI 406 stainless steel, AISI 430 stainless steel, AISI 316L stainless steel, and AISI 304L stainless steel.

A comparison with corrosion results obtained previously in oxygenated steam at 550 C (1022 F), 3000 psi has shown very little difference between oxygenated (3-4 ppm) and deoxygenated (< 50 ppb) steam.

Except for one heat of AISI 406 stainless steel, the three general order of groups have remained the same. The one heat of AISI 406 stainless steel has shown an increased corrosion and has shifted from group two to group three.

A metallographic examination of the as-corroded alloys has revealed a relatively uniform corrosion attack. There was no significant sign of intergranular or internal oxidation.

Fretting Corrosion of PRTR Element. Additional exposure was obtained on a modified PRTR fuel element having a full 360° support to minimize fretting. After 40 days in TF-7 at 530 F with external vibrations (30 cps, 7-8 mils deflection), no fretting was detected on the supports or on the tube in the area of the supports. Approximately five mils metal was removed from a single-rod wire wrap at three locations where it contacted the pressure tube wall.

Measurement of Zirconium in PRTR Primary Coolant. Zirconium concentrations in the PRTR primary coolant are being measured routinely using emission spectrographic techniques. During the past month the concentrations have generally been in the normal range of 0-1 ppb with no indication of accelerated (fretting) corrosion. A review of previous data in terms of PRTR operation indicates that high zirconium concentrations were encountered on four separate occasions following pressure tube-nozzle regasketing operations.

Decay of Activity After Shutdown in PRTR. In the past routine radiation surveys of the activity at different locations of the PRTR primary system have been taken 8-12 hours after shutdown. To provide a basis for comparison with readings taken during decontamination, for example, when short-lived activities have decayed, surveys of the primary system were made 8, 32, and 104 hours after shutdown. The first data were obtained during the past month. The greatest rate of decay occurred between 8 and 32 hours after shutdown; factors varied from 2 to 20. After 32 hours, the decay was minor; in most cases, the readings after 104 hours were about the same as after 32 hours.

Reactor Components Development

Pressure Tube Monitoring. Work continued on the assembly of the Mark III monitoring equipment. A new borescope objective lens section has been delivered. Installation and checkout of the new objective lens section is pending completion of ID gaging of the PRTR process tube to be installed in the EDEL-I loop. Electrical wiring for instrumentation and operating controls is now approximately 50% complete.

PRTR Pressure Tubes. Four of five planned elevated temperature burst tests have been performed. These tests were done at temperatures of 200 F (93 C), 300 F (149 C), and 400 F (204 C). The purpose was to obtain data to show how burst test strength and elongation of irradiated PRTR tubes is influenced by temperature.

The burst test data obtained previous to these four tests indicate that a burst test temperature change of 100 F (37 C) would cause 10% or greater change in ultimate tensile strength. The results obtained from these four tests confirm the 10% per 100 F dependency for annealed material. Only one specimen of cold worked material was included in this group, and it indicated that cold worked material may not have a 10% per 100 F dependence.

Tube number 5683 was discharged from the PRTR on April 5, 1963. Six burst test specimens were cut from this tube by using remote cutting tools for underwater work at the PRTR basin. Two of these specimens will be used to examine areas on this tube adjacent to fuel element supports, two will be used to examine defects on the OD of this tube and two will be used to examine the effect of the PRTR environment on the burst and other metallurgical properties of the annealed and cold worked portions of the tube.

Second Generation Mechanical Shim Rod for PRTR. Detailed design of the entire shim rod assembly is estimated to be 90% complete. All drawings are now in check print status. Fabrication of components for both assemblies is now proceeding concurrently and is considered to be 55% complete. An order has been placed for three ball screw assemblies in which the screws will be fabricated of Zircaloy-2. Delivery of these rods is scheduled for August 1.

Shim Rod Environmental Test Facility. Operation of the facility with a new first-generation PRTR shim rod installed in it has continued during the month. The shim rod assembly has been operated briefly every two days. The tendency for the single rod drive to coast down is still present although to a lesser degree than when the rod was initially installed. Consideration is being given to devices to prevent this uncontrolled movement.

A control console has been assembled for use with the second generation shim rods. The selsyn position readout equipment was installed in the console. Operation of the selsyn position readout equipment has been accomplished successfully.

Considerable difficulty has been experienced with the plastic test section fogging on the inside. The hot water is apparently causing a change in the composition of the plastic which renders it translucent rather than transparent. The plastic test section will be replaced with one made of glass.

Fretting Corrosion Investigation. Four amplifiers for use with the oscillographic recorder have been received; three remain to be delivered. Design of the piping to be used in the test pit is complete. Procurement of material is estimated to be 90% complete. Remaining equipment is scheduled for delivery by June 15. Installation of the test pit piping has been started. Fabrication of an instrumented fuel element for use in the test pit mockup was started and is estimated 10% complete.

PRTR Rupture Loop Components. The three stainless steel Grayloc connectors, replacements furnished by Gray Tool Company because of a design error on the original connectors, were received. Testing is presently in progress on one of these to determine the leak rate during temperature cycling. About 29 cycles out of 50 have been completed. Leakage rates have fluctuated sporadically and the results have not been analyzed sufficiently to determine trends. A total of 93 hours of steady state conditions of 600 F and 2100 psig have been logged. The steady state leak rates have varied between 0.0147 lb and 0.0252 lb/day. The average over-all leak rate for cycles 1 through 10 was 0.0135 lb/day, although for cycles 3 through 8 the sampling line was clogged. The over-all leak rate for cycles 11 through 20 was 0.0177 lb/day. Leak rates during heatup varied between a minimum of 0 and a maximum of 0.529 lb/day.

PRTR Pressure Tube Seals. The design of a 3-station pressure tube seal test assembly for use in conjunction with EDEL-I is complete. Fabrication was started and is estimated 10% complete. All components are on hand except for a few stainless steel fittings which are scheduled for delivery June 14, 1963.

Shroud Tube Replacement Mockup. Construction of the shroud tube mockup pit is estimated to be 90% complete. Procurement of materials for and fabrication of the mockup crates which will be installed in the pit was started by Minor Construction. Design of tools for use in shroud tube removal and replacement was started.

Plutonium Recycle Program Hazards Analysis

General. The final response to the audit of PRTR operation and the discharge method for the PRTR Fuel Element Rupture Test Facility were reviewed by the General Electric Technological Hazards Council at the eleventh meeting. The final response to the audit was accepted and the discharge method for the rupture loop was accepted as assuring adequate safety provided a check valve is added to the pressure tube inlet hose fitting.

Notification was received of AEC approval of the proposed increase of limits on PRTR maximum heat transfer flux and tube power to 650,000 Btu/(hr)(ft^2) and 1800 kw, respectively.

A study has been completed to determine the appropriate revision of the PRTR aqueous effluent activity limits when the new water treatment plant at Richland begins operation. Since this new plant is only four miles downstream of PRTR instead of the present 16 miles to the nearest sanitary water intake, a change in the PRTR aqueous effluent activity limits will be necessary. It is planned to install new, more sensitive aqueous activity monitors at PRTR before the new Richland plant begins operation. The aqueous effluent activity alarm point will be reduced by a factor of 10, but the automatic aqueous containment trip point will remain unchanged.

Recovery From Convection Cooling. A study has been undertaken to determine how recovery of the PRTR primary coolant system from boiling convection cooling can be accomplished without (further) degradation of the D_2O. The following procedure would permit recovery without further loss or degradation of D_2O, without the necessity for sending personnel into the containment vessel, and with minimal thermal shock to the system. When normal electric power is restored, reduction of the steam generator pressure should be discontinued and a D_2O injection pump should be operated until the primary coolant system pressure starts rising rapidly, because of covering a substantial part of the condensing surface in the steam generator tubes, exposed at the onset of boiling convection by the expulsion of D_2O through the pressurizer relief valves. Injection should be stopped at a primary coolant system pressure of about 500 psig and boiling convection allowed to continue about an [redacted] under aqueous containment has been completed. The study will be published as document HW-77677.

Case studies presented show that with a fuel element rupture releasing the entire contents of a single fuel rod to the primary coolant, and a pre-existent leak of D_2O into the steam generator shell of 10 lb/hr, the aqueous effluent alarm limit of 5×10^{-4} μc/ml would be quickly reached, but the automatic containment trip level of 5 μc/ml would not be reached, even at the normal boiler blowdown rate. Under present operating instructions, less than 60 curies of mixed fission products would pass to the river over an 8-hour period. More than half of this would escape via assumed leakage of 50 lb/hr of boiler water through a shutdown cooling valve, a 6-inch gate valve subject to full boiler pressure of 410 psig.

If the incident were handled more promptly than assumed in the study, the initial peak emission during the first half hour would not be greatly affected, but the rate of escape of contamination to the river thereafter could be reduced from 1-2 curies/hr to about 0.5 curie/hr.

An incident causing a containment trip in 0.1 hr after the rupture was also analyzed. Both the D_2O leak and the rupture were 7.65 times as severe as in the cases above. In this case, several hundred curies of mixed fission products would escape to the river; 90% would escape through liquid leakage and only 10% through steam leakage.

Removal of the shutdown cooling valve (no longer used) and provision of a steam generator depressurization valve discharging inside the containment vessel are recommended.

Plutonium Recycle Critical Facility

Plans for modification of the PRCF for light-water moderation and the conduct of critical tests using PuO_2-UO_2 fuel rods fabricated for the EBWR were reviewed by the General Electric Technological Hazards Council at the eleventh meeting. The Council approved the modification for light water critical tests but requested further information on the uniformity of distribution of PuO_2 in the EBWR mixed oxide fuel elements.

Thermal Hydraulic Studies

Air Cooling of PRTR Fuel Elements. Experimentally determined heat transfer coefficients for a 19-rod bundle cooled by transverse air flow were compared with values calculated from an equation developed by Colburn[1] for cooling of staggered banks of tubes. The experimental data had been obtained with an electrically heated model in tests investigating performance of the PRTR Fuel Examination Facility cooling system. With air as the coolant, the Colburn equation reduces to the following:

$$\frac{h_m D}{K_f} = 0.3 \left(\frac{D\ G_{max}}{\mu\ f} \right)^{0.6}$$

where: h_m = average heat transfer coefficient of rod
D = rod diameter
G = mass velocity based on minimum free area
μf = coolant viscosity at film temperature
K_f = coolant thermal conductivity at film temperature.

The 19-rod bundle, with a central rod surrounded by rings of 6 and 12 rods, lacks the symmetry of staggered tube banks with equal numbers of tubes in each bank. Earlier studies had shown that use of the Colburn equation, with G_{max} as defined above, produced good agreement with the data only for rods located in the region of minimum flow area. For rods in locations of greater free area and, thus, lower air velocity, agreement was poor.

In the latest comparison, the average velocity at each rod location was used to determine G. Calculations based on this mass velocity showed good agreement between the equation and the experimental data; generally within 10%, with a maximum disagreement of 25%. The comparisons were made for the center rod and for rods at various locations in the two rings. Air velocities ranged from about 15 to 90 ft/sec.

2. Plutonium Ceramic Fuels Research

Plutonium Nitride. Pelletized PuN was heated in flowing hydrogen for 12 hours at 1700 C in a molybdenum boat. The PuN pellets slumped severely and flowed together during heating. This action may have been caused by formation of free molten Pu at high temperatures.

Plutonium Borides. Several plutonium borides have been prepared by reacting mixtures of PuO_2, carbon, and boron at 1600 C in vacuum.

(1) Colburn, A.P., "Trans. Am. Inst. Chem. Engrs.," 29, pp. 174-210, 1933.

Plutonium Cermets. The density of a pneumatically impacted, PuN - 50 w/o W specimen was found to be 16.9 g/cm^3 (100% TD). However, metallographic examination showed a nonuniform distribution of the PuN in the tungsten matrix. Modification of the powder blending procedure should correct this.

Instability of Molten (U,Pu) C. Buttons of (U,Pu) C, held molten for increasing lengths of time, showed a preferential loss of Pu. Two 20-gram buttons were prepared by arc-melting 20 w/o PuC and 80 w/o UC. In one sample the UC was stoichiometric (4.8 w/o C) and in the other it was hyperstoichiometric (5.2 w/o C). The starting buttons were remelted six times for homogenization and then sampled. They were subsequently held molten for additional time periods (1, 2, 4, and 8 minutes) and sampled at the end of each period. Both samples showed a regular decrease in Pu/U ratio to 0.1 after eight minutes.

Plutonium Alloy-UO_2 Compatibility Studies. UO_2 pellets with a thin wafer of Pu-15 w/o Zr imbedded in each pellet were heated in flowing helium for 12 hours at 1700 C. Chemical, metallographic, and x-ray diffraction analyses showed no diffusion of the alloy into the UO_2. The alloy oxidized to form PuO_2 and an unidentified compound.

Irradiation Performance of MgO-PuO_2. X-ray diffraction data show no crystallographic changes in a MgO-13.52 w/o PuO_2 specimen irradiated to an exposure of 0.5 x 10^{20} fissions/cc. The calculated maximum core temperature was 1700 C.

Characterization of Pneumatically Impacted PuO_2. Plutonium dioxide powder is being densified into compacts by pneumatic impaction. The first compact of plutonium dioxide powder was 89% TD, with an O/Pu ratio of 1.94. The second compact was 95.1% TD, with an O/Pu ratio of 1.91. X-ray diffraction analyses are in progress.

3. Ceramic (Uranium) Fuel Research

Electron Microscopy of UO_2-W Cermet. Thermal reactions of a UO_2-50 w/o W cermet at temperatures to 1900 C were recorded by cine' micrography during reflection electron microscopy. Extensive thermal etching and recrystallization of the tungsten occurred above 1200 C. The UO_2 retreated below the tungsten surface with increasing temperature. The specimen was heated with an auxiliary electron gun. The 1900 C temperature was achieved with an input of less than 20 watts.

Pneumatic Impaction of Cermet Shapes. A technique has been developed for fabricating complex cermet shapes by pneumatically impacting the powder into the cavities of a mild steel block. The steel block containing the cermet powder was inserted in a can, heated, and impacted. The steel was removed mechanically and chemically, leaving a compacted solid of the desired shape. The structural quality of thin-wall cermets fabricated by this technique has been poor thus far.

Preparation and Properties of UOS. Chemical analysis of UOS prepared by precipitation from a fused chloride melt at 550 C revealed 84.7 ± 1.5 w/o uranium (vs 83.2 w/o theoretical). Analyses for oxygen and sulfur are in progress.

Preparation and Properties of UP. Uranium phosphide, produced by the solid-solid reaction of uranium shot and red phosphorous, contained only 2.8 w/o phosphorous (vs theoretical 11.5 w/o). In reporting the melting point of UP last month, it was assumed that the observed melting points were those having two-phase mixture of UP and UO_2. This conclusion now seems questionable because of the low phosphorous content.

Because of the low product yield, a solid-gas reaction of uranium shot and phosphine is now planned. A similar reaction with plutonium is also planned to produce PuP.

Optical Absorption in UO_2 Single Crystals. The absorption coefficient (α) for a single crystal UO_2 at room temperature has been measured at wavelengths from 0.6 to 15 microns. Values of α vary from 100 cm^{-1} at 2 microns to 1 cm^{-1} at 11 microns. In the infrared region between 3 and 13 microns there is a large optical window with a minimum absorption at 11 microns. These observations support predictions that internal radiation contributes significantly to heat transfer in single crystal UO_2.

Transmission of Laser Beam Through UO_2. The relative transmission of a laser beam (wavelength .69μ) was significantly greater in single crystal UO_2 than in polycrystalline UO_2. Transmission at this wavelength in the single crystal significantly decreased as the specimen temperature was increased from room temperature to 1000 C.

Thermal Conductivity of UO_2. The thermal conductivity of single crystal UO_2 relative to polycrystalline UO_2 was measured to 1800 C using a radial heat flow technique. The presence of a radiation contribution, suggested by earlier studies, was confirmed in this experiment through the use of tungsten coatings on hollow, cylindrical specimens. The purpose of the opaque surface layer of tungsten is to intercept all radiation impinging upon it. This permits precise computation of the total flux through the specimen from a measurement of the specimen surface temperature. Earlier experiments yielded anomalous results because radiation passed directly through the specimens without being detected.

Conductivity values obtained to date coincide at intermediate temperatures (800 to 1100 C) with the measurements made by BMI. At 1200 C, however, a maximum of 0.065 watts $cm^{-1}C^{-1}$ occurs, followed by an exponential decrease to 0.020 watts $cm^{-1}C^{-1}$ at 1800 C. This lower value is 10-15% greater than the conductivity of pressed and sintered UO_2 at 1800 C. The enhanced conductivity of single crystals at intermediate temperatures results from internal transmission of infrared radiation. The drop-off at higher temperatures is probably related to an increase in absorption coefficient.

Irradiation of Single Crystal UO_2. A single crystal of arc-melted UO_2 has been irradiated for 310 days in the ETR to an estimated exposure of 5×10^{20} fissions/cc. Post-irradiation property measurements were initiated. Data will be compared to those for an identical single crystal previously examined after irradiation to 1.44×10^{20} fissions/cc.

Fission Product Distribution. Autoradiographs of fuel elements defected during irradiation show significant relocation of fission products in the immediate area of the defect. The depletion band, generally associated with the high density portion of the columnar grains, is enlarged near the defect.

Materials and Information Exchange. A request was received from ORNL to provide 11 UO_2 pellets (1-inch length by 0.430-inch diameter) and single crystals for possible use as gamma absorptiometric standards in fuel fabrication studies. An irradiated UO_2 specimen irradiated in the Argonne TREAT reactor by APED was received for ceramographic examination.

Irradiation of Molten UO_2. A tungsten-clad UO_2 capsule (GEH-14-420) designed to operate with a completely molten core was charged into the ETR on May 13, 1963. The tungsten capsule is centered in an evacuated stainless steel capsule identical in design to the capsules previously used for irradiation testing of cermets.

A 0.015-inch diametral gap was left between the fuel pellets and the tungsten to allow for the 25% volume increase experienced by the UO_2 during the transition from room temperature to 2800 C (molten).

Examination of Electrical Resistivity Element. Preliminary post-irradiation examination of the electrical resistivity element (GEH-4-82) indicates that electrical contact was lost in the area of the mechanical joint between the tungsten center coil and the nickel lead. This joint was made within a porous alumina spacer which apparently retained some oxygen during outgassing. The observed recrystallization of the alumina at the joint area probably allowed entrapped oxygen to escape and come in contact with the hot tungsten coil. The oxidation of the tungsten and the subsequent volatilization of WO_3 apparently severed the connection, thus allowing the central coil to drop to the bottom of the molten zone of the UO_2. Nonporous alumina spacers will be used to eliminate this type of failure.

Irradiation of Cermets. Six vacuum insulated cermet capsules were charged into the ETR. These capsules are to operate with cladding temperatures of approximately 2000 C. The ultra-high operating temperature is attained by suspending the refractory metal clad cermet within a stainless steel outer capsule and evacuating the intervening space before welding the final closures. Radiant heat transfer is the primary cooling mechanism for the inner fuel capsule.

This group of irradiation tests (GEH-14-414 through 419) include two Mo-UO_2 and two Mo-UN cermets clad in Mo, one W-UO_2 and one W-UN cermet clad in tungsten. One Mo-UO_2 and one Mo-UN cermet will be irradiated for two ETR cycles. The other capsules will be irradiated for one ETR cycle.

4. Basic Swelling Program

Irradiation Program. Six controlled-temperature general swelling capsules were constructed and two were charged into a reactor. The other four will be charged as soon as space is available. The two capsules that were charged each contain eight uranium specimens with different compositions and heat treatments. These capsules were modified slightly to include four pairs of specimens instead of three, and five thermocouples instead of four. The internal space housing the specimens was also redesigned to accommodate different shapes and numbers of specimens.

Post-Irradiation Examination. Optical and electron metallography was completed on the irradiated uranium specimens recovered from two previously opened capsules which operated at 575 C (1067 F) to 0.15 a/o B.U. of the specimens. One capsule contained natural uranium (0.72 a/o U-235) specimens (two as-extruded and one beta quenched), whereas the second contained uranium enriched to 1.44 a/o U-235. The metallography agreed qualitatively with the density values in that the natural uranium specimens suffered more damage than did the enriched ones. Severe tearing, psuedo second phase, and porosity were observed in the normal specimens similar to the poorer of the as-extruded enriched samples. The beta-quenched specimen (large grained) was in extremely poor condition. As there was an inconsistency between the densities and metallography of one of the enriched as-extruded specimens, both types of evaluation were repeated on different halves of the split tubular specimens; namely, the halves that had been used for density were processed for metallography and vice versa. The inconsistencies were resolved and arose from the fact that one of the half specimens on which metallography was originally performed had suffered more damage than the other half. The observations on the specimens of these two capsules verify that the original grain size has not changed, some tearing is present, and considerable porosity is present, especially at grain boundaries. The uranium lattice appears to be virtually free of cold work of the type that can be detected metallographically.

Two additional, irradiated, controlled-temperature, general swelling capsules were opened in Radiometallurgy and the specimens (two as-extruded and one beta-quenched in each capsule) recovered. One capsule had been irradiated at 625 C (1157 F) to 0.1 a/o B.U.; the second had been irradiated at 525 C to 0.03 a/o B.U. Optical metallography was completed on the 625 C (1157 F) specimens and replicas are currently being processed for electron metallography. Density measurements are also in progress.

The optical metallography of the specimens from the 625 C capsule correlates well with observations made on other specimens irradiated at 625 C but to higher burnups, 0.15 and 0.27 a/o. The as-extruded specimen irradiated at 625 C in the present capsule indicates the following: (1) grain boundaries are readily seen, (2) the grain size is identical with the preirradiation grain size, (3) large pores have segregated to the grain boundaries, (4) no lattice distortion, tearing, or psuedo-second-phase is observed. The second as-extruded specimen which operated at about 580 C (1076 F) was similar to the 625 C specimen except that crystallographically aligned pores, tears, or psuedo-second-phase were also present. The large grained specimen, beta-quenched prior to irradiation, which operated at about 585 C

(1085 F) exhibited considerable warpage, porosity, tearing, and psuedo-second-phase. Both of the as-extruded specimens showed macro longitudinal striations on the surface which were parallel to the extrusion direction. It would appear that "growth" is still a factor to be considered even at 625 C. One of the more remarkable observations is the fact that the grain size did not change during the irradiation. Non-irradiated control specimens exhibit exaggerated grain growth after vacuum annealing at 640 C (1184 F) for only two hours. This indicates that irradiation immobilizes grain boundaries in this material. Additional annealing studies will be conducted in simulated capsules.

Restrained Irradiations. The influence of restraint on the swelling of uranium is being investigated. Zircaloy-2 clad rods of unalloyed uranium and uranium - 2 w/o zirconium are being irradiated in NaK-filled capsules. Thirty-six capsules containing a total of 94 fuel rod samples have recently been irradiated. Nine of these capsules were irradiated to an exposure of 1100 MWD/ton, and the remainder to 1800 MWD/ton. Initial examination of low exposure unalloyed samples from capsule 3031 revealed diameter increases of 1.6% at volume mean fuel temperatures of 560 C (1040 F). Diameter increases of 2.7% were observed on U - 2 w/o Zr samples from the 3A24 capsule at volume mean fuel temperatures of 400 C (752 F) and 1100 MWD/ton. Measurements of density and burnup from the 27 samples in the low exposure capsules is planned for the next month.

Thorium. Two thorium specimens (B-2 and S-3) irradiated to 0.18 and 0.92 a/o burnup, respectively, and annealed at 750 C (1382 F) for 100 hours have been processed for hardness, density, and optical metallography. Replicas were prepared and their examination is currently in progress. The annealing did not significantly alter the optical microstructure. The densities after annealing of the 0.18 (B-2) and 0.92 a/o (S-3) burnup specimens are, respectively, 11.65 and 11.45 g/cc which compares with 11.67 g/cc for non-irradiated thorium. Thus, B-2 has not increased in volume at all, and S-3 has increased but about 2%. It should be emphasized that S-3 showed a volume increase of about 1½% in the as-irradiated condition, and this was attributed primarily to the existence of cracks in the specimen. Annealing at 750 C (1382 F), therefore, has not caused much swelling. Appreciable recovery in hardness, however, has occurred. Specimen B-2 (0.18 a/o B.U.) decreased in hardness from R_B 78 to R_B 50 and specimen S-3 (0.92 a/o B.U.) decreased from R_B 69 to R_B 48. When the electron metallography is completed, the specimens will be reannealed at a higher temperature.

Thorium-Uranium. The four thorium-uranium tensile specimens (1.0, 4.0 and 5.4 w/o U) recovered from noninstrumented NaK-filled capsules have been tensile tested at room temperature. The first sample was tested at a crosshead speed of 0.02-inch per minute and was observed to fail outside the gage length in an extremely brittle fashion at about 0.02% offset. The other three samples were tested at a crosshead speed of 0.005-inch per minute and while the fractures were all brittle in nature, the specimens exhibited reductions in area of 5 to 12%. The 0.1% offset yield strengths were 75 to 79 Ksi, and the ultimate tensile strengths were 81 to 85 Ksi. Hardness tests made on the buttonhead of the specimens prior to tensile testing ranged from 74 to 92 R_B. The hardness will be measured again on a polished cross section of the broken ends of the tensile specimens after these ends have been prepared for metallography. Annealing studies will be conducted in order to evaluate the swelling characteristics of these alloys with post-irradiation annealing.

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. Delivery dates for A-286, AISI 304, AISI 348, AM 355, and Zircaloy-2 are all prior to the third week of June 1963. The heat analysis for AISI 348 indicated that the tantalum content was above specifications. The heat was rejected on this basis, thus causing a two-week delay in delivery. Ingots of AISI 304 have been bloomed and will be hot-rolled into bar and sheet in the near future. Extrusion of Zircaloy-2 bar stock has also been completed. Rolling of Zircaloy-2 plate will be done at the Westinghouse Blairsville plant from an ingot sent to that site during the week of May 10, 1963. Quantities of Inconel 600 and Inconel X-750 bar, sheet, and plate have been hot-rolled during the past month by the Huntington Alloy Products Corporation. This material will be shipped after final inspection has been completed.

Tensile specimens of several nickel base alloys have been fabricated. These alloys include R-27, R-235, Inconel 718, Hastelloy N, Inconel 625, Hastelloy X-280, and TD Nickel. Specimens from each alloy will be irradiated in 280 C (536 F) water, 80 C (176 F) water, and 525 C (977 F) gas.

In-Reactor Measurements of Mechanical Properties. The test in progress is the first of a series to determine the stress dependency of Zircaloy-2 creep during irradiation. The present test is being conducted on 20% cold worked Zircaloy-2 at a temperature of 350 C

(662 F) and a stress of 20,000 psi. The test has been in progress for 2800 hours. The creep rate at 1700 hours was determined to be 1.8×10^{-6}/hr. This rate has remained unchanged during reactor operation for the length of the test. After correcting for a thermocouple error, the ex-reactor test exhibited a rate of 5×10^{-7}/hr at 2000 hrs. Creep rates during all reactor outages have been equal to or slightly lower than reactor operating creep rates. Opposite observations have been made for all previously reported tests at 250 C (482 F), 310 C (590 F), and 350 C (662 F) where the applied stress was 30,000 psi. In these tests, reactor outage creep rates exceeded reactor operating creep rates. It should be pointed out that the scatter in creep data is usually large. Additional testing will be required to determine if the data from the present test is truly representative of the material at 350 C (662 F) and 20,000 psi stress.

Two additional creep capsules were charged in the latter part of May 1963. The creep tests scheduled for these capsules are 35,000 psi stress at 310 C (590 F) and 25,000 psi stress at 350 C (662 F). These two tests will allow a more complete evaluation of the stress dependency of in-reactor rates to be made.

In addition, a heater wire life-time test capsule was charged. The heater life-time test capsule houses eight heaters, two each of Karma, Kanthal, Tophet A, and 406 stainless steel heater wires. The heaters are wound non-inductively, identical to those incorporated in the present creep capsules. One heater of each type of material will be run at power while the second heater will be checked for resistance and continuity to determine its "cold life." The powered heaters will be run at constant and equivalent power densities (watts per unit surface area of wire). Thermocouples will monitor the temperature and flux measurements will be made.

Tensile testing of unirradiated Karma, Kanthal, Tophet A, and 406 stainless steel wires has been completed. These will be compared with their irradiated counterparts to determine the effect irradiation may contribute to heater failures during reactor operation. Irradiated samples have yet to be tested.

Present in-reactor creep capsules were shown, in the laboratory, to be capable of 900 C (1652 F) operation with minor modification. The modification consisted only of the addition of two concentric heat shields, accenting the design versatility and adaptability of in-reactor creep capsules to particular environmental conditions. Creep capsules modified to include heat baffles will be used to test stainless steels and other intermediate temperature structural materials in a neutron environment.

Tests have been started on tungsten wire configurations for use as heaters in a proposed high temperature in-reactor test capsule. Positioning of a tantalum heat shield is also being investigated. With the present design, a one-eighth-inch molybdenum rod has been heated to 1100 C (2012 F) under a vacuum of one micron for short periods of time. Long term tests are being started to determine the heater lifetime under the present testing conditions.

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 will be given to the determination of mechanical property changes produced in metals by irradiation at elevated temperatures.

A total of 88 specimens were tested during the month including bend tests on Zircaloy-2 and AISI 348 stainless steel; notch tensile tests on AISI 348 stainless steel; and tensile tests on Zircaloy-2 and AISI 304, 348, and AM350 stainless steels. These tests were performed on control specimens with the exception of six irradiated AM-350 specimens.

Approximately 300 Zircaloy-2 tensile specimens were fabricated from three typical process tubes. These tubes represent different manufacturers and different fabrication processes, and contain from 18 to 35% cold work. Since fabrication history, or more specifically preferred orientation, has been shown to have marked effects on mechanical properties, each tube is being characterized by x-ray pole figure diagrams, tensile tests, and metallography prior to irradiation. The tensile properties of specimens cut in the longitudinal direction compare favorably to longitudinal specimens cut from rolled plate having comparable cold work. Yield strength varies from 57,000 to 71,500 psi and ultimate strength varies from 79,500 to 92,000 psi among the three tubes. Two hundred sixty-five specimens were measured, stamped, and weighed and are being autoclaved for subsequent exposure in the G-7 and ex-reactor loops.

Tensile specimens of Zr-2 tested at -75 C (-100 F) and -196 C (-320 F) were examined metallographically to determine the combined effects of low testing temperature and preferred orientation on deformation twinning. At all temperatures twinning is pronounced in transverse specimens and present in minor amounts in longitudinal specimens. At room temperature and -75 C (-100 F) deformation twinning is by the $\{10\bar{1}2\}$ mode; whereas at -196 C (-320 F) both $\{11\bar{2}1\}$ and $\{10\bar{1}2\}$ twinning are important modes. Studies are continuing to determine where between -75 and -196 C (-320 F) $\{11\bar{2}1\}$ twinning occurs in significant amounts.

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. Tensile specimens of TZM molybdenum alloy and Cb - 1 Zr and Cb 752 alloys are being fabricated. These specimens will be irradiated in the ETR G-6 position to evaluate their property changes due to irradiation. Characterization of these materials has also begun with the preparation of specimens for recrystallization and grain size studies. Hardness, yield strength, tensile strength, and ductility measurements will also be made.

Refractory alloys will also be used in space applications utilizing liquid metal coolants such as potassium, lithium, and sodium. Since in-reactor liquid metal loop facilities do not exist, liquid metal environmental capsule irradiations will be made. All liquid metals to be used are reactive in air and must, therefore, be handled in an inert gas. An inert gas glove box has been designed which will contain facilities to purify both the inert gas and the liquid metal.

Various heating concepts such as electron beam, resistance, induction, and infrared radiation are being explored to provide elevated temperature (room temperature to 3000 C/5430 F) mechanical testing capability. Infrared radiation as a heat source is of interest for temperatures in the range room temperature to 1500 C (2730 F). By means of this energy source complicated shapes can be heated quickly in a variety of gaseous environments. A heating arrangement for use in tensile testing has been assembled and is being evaluated. Initial heating trials with this equipment have shown system feasibility, and refinements are being made to incorporate close temperature control and specimen testing fixtures.

Further examination of the fracture surfaces of notched beams of cold-rolled Zircaloy-2 broken at various temperatures from 22 to -196 C (72 to -320 F) have been made. The general appearance of fracture surfaces occurring parallel and transverse to the rolling direction are similar, except that the fracture surface parallel to the rolling direction exhibits a greater incidence of crystalline features. These features are attributed to twin-twin or twin-slip interactions with the fracture surface. At 22 C (72 F) these features are occasionally observed in fracture surfaces parallel to the rolling direction but are absent in surfaces transverse to the rolling direction. The degree to which this structural feature contributes to the ductile-brittle transition at about -75 C (-100 F) is being investigated.

Experiments are now under way to determine the change in compliance of notched beams in four-point bending as a function of notch depth. Compliance in this case is defined as the reciprocal slope of the moment versus angular deflection curve. This information is used to determine the elastic energy release rate as a function of notch depth. This relationship is essential for determining fracture toughness values of materials.

Damage Mechanisms. The objective of this program is to establish the nature of the interaction between defects present prior to irradiation and those produced during irradiation, with emphasis on the role played by interstitial impurities. The investigation is presently concerned with high purity iron and its low carbon and nitrogen alloys.

During this period, two iron samples with small amounts of Ti added have been rolled to foil for examination by electron transmission microscopy. In addition, 40 tensile specimens of Battelle zone refined iron and Ferrovac "E" were machined. No testing has been performed on either of these projects.

An electropolish with a perchloric acid-glacial acetic acid bath has been found to give a better surface finish than chemical etching for the purpose of cleaning the surface of iron tensile specimens.

A design was partially developed of a cryostat for tensile testing sub-size polycrystalline and single crystal samples. It is to incorporate continuous temperature control over the range 4.2 K - 273 K (-452 F to 32 F) and a maximum tensile load of 10,000 lb on the specimen.

In addition, design of a vacuum chamber and sample holder for internal friction measurements in the kilocycle frequency range was initiated.

Environmental Effects. In-reactor weight gain results for 24 coupons of Zircaloy-2 material irradiated in quadrants 87 and 88 during ETR cycles 43 through 49 were obtained during the report period. Both quadrants were exposed to a fast neutron flux of 7.6×10^{13} nv for 125 days at 540 F (282 C) for an integrated flux of 8.2×10^{20} nvt. The average weight gain for these coupons was 121.3 mg/dm^2 compared with 8.8 mg/dm^2 for equivalent material exposed to the same conditions in the CMO out-of-reactor loop. As with results previously reported for other quadrants there was no significant variation of weight gain with level of cold work. The coupons appear smooth, dark and glossy to the naked eye and show no signs of gross blistering, spalling or discoloration.

Irradiation Damage to Inconel. Metallography of the Inconel pressure tube from the DR-1 Gas Loop has shown that the sulfur contamination of the tube was located over a small area near the transverse break. The remainder of the tube has a three- to five-mil corrosion layer on both the inner and outer surface. Analysis of the reactor gas stream which was in contact with the outer portion of the pressure tube revealed a sulfur concentration of less than one ppm. Therefore, any sulfur contamination must have been introduced by some other method, possibly during heat treatment prior to installation in the reactor.

Oxidation of Superalloys. Oxidation testing of Hastelloy C, a nickel base superalloy, continues. As-received and surface-abraded specimens have been oxidized at 1000 C (1832 F) in pure oxygen and laboratory air.

Oxidant pressure has little effect on oxidation of abraded specimens; parabolic kinetics hold, indicating that diffusion through the oxide scale is rate-controlling.

Pressure has a profound effect on the oxidation of as-received specimens. At 3 mm O_2 the specimen rapidly gains weight, levels off, loses weight for a period of time, and finally resumes gaining weight to the conclusion of the test (24 hr). To a lesser extent the same sequence occurs at 25 mm O_2. For any given time, weight gain is higher at 25 mm than at 3 mm O_2 pressure. In one atmosphere air, weight gain is very high initially, but the scale formed is quite protective, and oxidation is slow thereafter. Further conclusions await completion of testing on electropolished specimens.

Microscopic inspection reveals that internal or sub-surface oxidation occurs extensively in air-oxidized samples of Haynes 25. No internal oxidation can be observed in air-oxidized Hastelloy X. Internal oxidation in Haynes 25 is apparently independent of surface preparation. In both Haynes 25 and Hastelloy X the metal oxide interface is considerably smoother and more uniform for electropolished specimens than for either abraded or as-received specimens.

Irradiation of Titanium. Two samples of titanium to be used in corrosion capsule construction were irradiated at KE-Reactor for 70 hours to compare activity and decay rates of pure titanium and a titanium alloy containing 6% aluminum and 4% vanadium. Each sample weighed 0.94 gram, and the decay rates were nearly the same over a period of eight days. Readings on a C.P. after eight days of decay were 15 mr/hr at six inches in open air.

Gas Loop Development. The regenerative heat exchanger in the model high temperature gas loop will use welded and subsequently cold reduced Haynes alloy 25 tubing. The quality of a five-foot tube section produced by the same basic process that will be used for the heat exchanger tubing was evaluated.

The tube had a clean polished outer surface and a smooth clean inner surface with no visible blemishes or defects. The tube was clear of defects as shown by fluorescent penetrant and radiographic techniques except for one small line indication observed in penetrant testing. This was shown by metallographic examination to be a crack approximately two mils deep by one-half-inch long in the weld heat affected zone. The tube dimensions were consistent, with the outer diameter varying from 0.5042 to 0.5055 inch and the wall thickness varying from 0.0817 to 0.0200 inch. Tensile and burst tests showed the base and weld metals to have ultimate strengths comparable to strip. The weld metal appeared to deform the same as the base metal. Expansion in the burst test and elongation in the tensile tests of better than 20% demonstrated the ductility. Carbide precipitation in the matrix and at the grain boundaries, present in both the base and weld metals, was more prevalent in the weld metal. The grain size of the weld metal was smaller than that of the base metal. Forty thermal cycles of a tube section consisting of heating to 2000 F (1093 C) and water quenching caused no apparent deleterious effects.

This type of seamed Haynes alloy 25 tubing appears to be of adequate quality for use in the regenerative heat exchanger. The shallow crack was apparently associated with grain boundary carbide precipitation in the weld heat affected zone. Close control of welding parameters and heat treatments are needed to avoid this type of defect. The metal exhibited considerable ductility but less than half that normal for solution annealed strip.

Gross effects of air, vacuum, and helium atmospheres on insulating material efficiencies were evaluated by tests at temperatures to 1900 F (1038 C). Results substantiated theoretical results. The major effect of the insulators appeared to be diminution of the amount of heat transfered by radiation. No insulation was indicated as markedly superior to another, on an insulating efficiency basis, although widely varying basic materials were tested (graphite fiber, silica fiber, and mixed alumina-silica fiber).

Neutron Dosimetry. Measurements were made of the fast ($E > 1$ Mev) and thermal-plus-epithermal flux in the core of PRTR in order to estimate the rate at which damage would accumulate in structural materials tested in the reactor. The test was conducted between February 13 and 23, 1963.

A tube approximately eight feet in length, containing flux-monitor wires, was suspended from the top shield of the reactor and extended down through monitor hole 1550 which is adjacent to the central channel in the reactor. Flux measurements were made from the top shield to within 11 inches of the bottom of the fuel. Nickel was used as the fast-flux monitor and cobalt was used to monitor the thermal flux. Cadmium shields were not used during the test because of the possibility of melting; therefore, the "cobalt flux" values cited include thermal and epithermal activation by the Co-59 (n, γ) reaction.

Data are presented in the table below. Calculations of thermal flux have not been made at this time because of uncertainties in the cadmium ratio and in effective cross sections which would be appropriate for the D_2O-H_2O system. It is noteworthy that the ratio, $\phi_{Co}/\phi\ (E > 1\ \text{Mev})$ for all monitor positions except the top and bottom averaged 12.3 ± 0.2, indicating a uniform thermal to fast neutron ratio over about 70% of the fuel length.

Distance from Top Shield(in.)*	$\phi(E>1\ \text{Mev})\times 10^{-12}$	$\phi_{Co} \times 10^{-12}$	$\frac{\phi_{Co}}{\phi(E>1\ \text{Mev})}$
17	1.4	29	21
25	4.3	52	12.1
33	6.2	74	11.9
41	7.3	89	12.2
49	7.9	98	12.4
57	8.2	103	12.6
65	8.0	98	12.2
73	7.2	--	
81	6.0	75	12.5
89	4.7	58	12.3
97	2.0	38	19

*The fuel extends from 20" to 108" from the top shield.

The flux, both fast and thermal, is skewed upward from the center-line of the fuel. This is reasonable since there is a large hole in the bottom reflector causing neutron leakage and shims are predominantly in the lower part of the reactor causing increasing neutron absorption. Both tend to depress the flux in the lower portion of the reactor.

6. Gas-Cooled Reactor Studies

EGCR Graphite Irradiation. The sixth capsule, H-3-6, in the series of EGCR graphite irradiations has operated successfully for one reactor cycle. The number 5 thermocouple became erratic at about mid-cycle and is now apparently open. The other eight thermocouples are operating satisfactorily.

Thermal Conductivity. The thermal diffusivity of a porous carbon, density 0.73 g/cm^3, was measured by the flash method. The thermal diffusivity was 0.011 cm^2/sec at 25 C. The thermal conductivity calculated from the thermal diffusivity was 0.014 cal/cm-sec-°C (a factor of about 30 less than a value typical of CSF graphite). This compared favorably with 0.015 cal/cm-sec-°C obtained by another method and indicates that the flash thermal-diffusivity technique is valid for porous carbon materials having a low thermal conductivity.

Work was continued on the use of a lead sulfide detector to measure the rear-surface temperature rise. The change in resistance of the detector was 315 ohm/°C in the temperature range 650-850 C. Although indications of temperature rise have been observed using this system, no quantitative results have been obtained so far because of a high noise to signal ratio and low detector sensitivity.

Graphite-Water Vapor Reaction Under Gamma Irradiation. Measurements of the rate of oxidation of TSX graphite by low concentrations of water in flowing helium in a Co-60 gamma flux have continued. In experiments conducted at 650-680 C with helium containing about 200 ppm of water, no significant variation in oxidation rate (about 1×10^{-4} to 2×10^{-4} percent per hour) with gas flow was observed. That there is little dependence on gas flow is perhaps not surprising in the light of previous work showing only a slight dependence, if any, on water concentration at this level but an apparent linear dependence on dose rate. These observations are in keeping with the hypothesis that the kinetic limitation is the rate at which active entities formed by radiation in the gas phase of the pores reach the surface of the graphite.

Inhibition of Graphite Oxidation. Graphite oxidizes in air at a rate which varies with the amount of prior oxidation. The rate of oxidation increases as surface area increases and tends to approach a constant rate which is taken as a characteristic rate. The effect of small amounts of CF_2Cl_2 on the air oxidation of graphite is complicated by the decomposition products of CF_2Cl_2, and rates of oxidation considerably higher than those achieved at steady state are sometimes observed initially. These high oxidation rates can be eliminated by purging with CF_2Cl_2 prior to heating in air.

After purging for at least three hours at room temperature, the initial rates at 600 C are low and increase slowly with time. At this temperature and 2% burnoff the weight-loss rate in air-0.5% CF_2Cl_2 is 0.4 to 0.7 times the rate in air alone. This ratio approaches 0.4 at approximately 10% burnoff.

A more exact method of determining the inhibition effects of CF_2Cl_2 has recently been tried in which tests were made on the same sample, thus eliminating effects due to sample variation. The oxidation rate was first established in air-0.5% CF_2Cl_2, and then changed promptly to air alone. At 600 C the ratio of rates was found to be 0.24 at 2% burnoff and 0.11 at 10% burnoff.

In a third experiment surface area measurements were made on one sample after 10% burnoff in air and on a second sample after 10% burnoff in air-0.5% CF_2Cl_2 to attempt to determine the effects of prior oxidation.

The surface areas were measured with nitrogen using the BET method. When the rates were expressed as fractional weight loss per unit surface area, the ratio of the rate in air-0.5% CF_2Cl_2 to the rate in air alone was 0.15. This ratio is in reasonable agreement with the value of 0.11 as determined on a single sample by the method described above. Specific rates were 0.0085 $g/m^2/hr$ at 10% burnoff in air at 600 C, and 0.0013 $g/m^2/hr$ at 10% burnoff in air-0.5% CF_2Cl_2.

The effects of iron and vanadium impurities, temperature, and gas composition are being investigated now that a reliable experimental method appears to be established. Interfering contaminants from corrosion in the halogen-bearing atmosphere are presumably avoided by using alumina and platinum in the hot zone.

7. Graphite Radiation Damage Studies

Effect of Impregnation on Radiation-Induced Contraction. Results have been obtained for a series of samples irradiated at approximately 650 C to an exposure of 1715 MWD/AT_K. The samples are intended to test the effect of impregnation and size of the filler particles on the extent of radiation-induced contraction. Three particle sizes, 0.015-inch, 0.030-inch, and 0.060-inch, were used and samples of each were given the following treatments: (1) a single impregnation with a well graphitizing pitch, (2) a single impregnation with a nongraphitizing binder, and (3) a double impregnation with nongraphitizing binder. Thus, nine test graphites resulted.

Post-irradiation measurement disclosed contraction in almost all samples, the largest effect being a function of impregnation treatment rather than particle size. The average percent parallel contraction for 12 samples of each type was: 0.011 ± 0.003 for samples with one pitch impregnation; 0.002 ± 0.003 for one impregnation with nongraphitizing material; and 0.023 ± 0.005 for two impregnations with nongraphitizing material. Thus, it appears that multiple impregnation leads to considerably higher contraction. It is surprising that the nongraphitizing impregnant led to lower contraction than pitch for a single impregnation. However, longer irradiations are necessary before definite conclusions can be drawn.

8. Aluminum Corrosion and Alloy Development

Corrosion of Aluminum in a Nonisothermal Loop. Tests are being conducted in a nonisothermal loop at 330 C (626 F) to investigate effects of dissolved corrosion product on the corrosion of aluminum. As reported previously, increasing the loop temperature drop from 8 C to 55 C (46 F to 131 F) approximately doubled the corrosion rate. However, in a subsequent run to confirm this observation, the rate dropped to a value observed in previous runs at low ΔT. However, during the second run at high ΔT, there was a concomitant decrease in pH (6.1 to 5.4), due apparently to regeneration of the deionizer resin.

In the third, ten-day run at large ΔT, both the pH (of the deionized feed water) and the rate increased, at least partially confirming the suspicion that pH was responsible for the unexpected decrease in rate during the second test at high ΔT.

Modification of C-1 Loop. Work on the C-1 Loop is about 90% completed. The shop work remaining consists of (1) completing the electrical hook-ups to the various pieces of equipment, (2) repairing three rejected welds, (3) installing the back-up accumulator and gas system to the accumulator, (4) completing the electrical panel, (5) painting the loop, (6) lagging the piping, and (7) calibrating the instruments.

A successful pressure test of the loop was completed. The test section in-reactor pipe was cut from the remainder of the test section so it could be tested inside with black light. A standard having the same dimensions as the in-reactor pipe was prepared for ultrasonic testing. All of the tubing connections on the loop were disconnected so they could be inspected.

The back-up accumulator was shipped May 8. As soon as it is received, it will be installed and the loop moved to 105-C.

9. Metallic Fuel Development

Effect of Thorium on Corrosion Resistance of Brazing Alloy. Autoclaving tests have been performed on brazed end caps which have been removed from thorium fuel elements. The cap was sectioned to expose the full length of Zircaloy - 5 Be braze to the high temperature water. Autoclaving conditions were 10 hours and 50 hours in 360 C water at 3000 psi. The braze nearest the thorium fuel showed white oxide corrosion gradually changing to the uniform black oxide coating on braze the furthest from the thorium. This shows that thorium contamination of the braze has a definite detrimental effect on the corrosion resistance of braze. A series of braze alloys containing varying amounts of thorium contamination are being made to establish what levels of contamination are allowable in the brazing alloy.

Two thorium - 2½ w/o U (normal) - 1 w/o Zr alloy billets were coextruded to 0.525" diameter Zircaloy-2 clad rod. Extrusion conditions were 2-hour preheat at 760 C with a 16.8 to 1 reduction ratio. The extruded surface before copper stripping appears satisfactory. The two 70-inch long rods will be fabricated into samples for visual autoclave defect testing.

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

19-Rod Bundle Burnout Data Comparison. Eighty-eight burnout data points obtained with 19-rod bundle test sections have been compared with a new empirical correlation of DNB (Departure from Nucleate Boiling) which was reported in the May 1963 Nucleonics. The correlation was developed especially for flow parallel to rod bundles. It resulted from a parametric study of about 3000 published DNB data points although the majority of these points were for circular or rectangular channels. The energy balance across a typical DNB test section was combined with the equation of motion and the continuity equation of two-phase flow, and the enthalpy rise at DNB conditions was obtained as a function of the nondimensional groups obtained by the combination of these equations. It was postulated that the DNB mechanism is hydro-dynamic in nature and that different mechanisms of DNB occur with different flow regimes, that is, subcooled or quality. By investigating each of the nondimensional groups systematically, an equation for describing the DNB condition in the subcooled region was obtained and another equation for conditions in the quality region. These equations were said to correlate the majority of the data within ± 25%.

These equations were applied to rod bundle burnout data by evaluating the enthalpy rise of a hot channel, i.e., for a flow subchannel in the rod bundle which has the smallest coolant flow area and the largest heat transfer perimeter. No mixing between coolant subchannels was assumed. The comparison between the equations and the available Hanford rod bundle burnout data shows that experimental values of burnout heat flux range from about 0.75 to 2.15 times the value predicted by the equations. The majority of the data points showed experimental heat fluxes higher than those predicted by the equations. The largest deviations from the predicted values were obtained with the 76-inch long test section which had 0.050-inch gap between rods of the bundle. Although this length of test section fell within the range of parameters used in developing the correlation, it is evident that the assumptions used do not apply for this long rod bundle test section. Hence, the application of these equations to fuel bundles on even longer lengths typical of power reactor applications would be questionable.

<u>19-Rod Fog Cooled Studies</u>. Forty-two experimental boiling burnout determinations were made with an electrically heated 19-rod bundle test section cooled with steam-water mixtures. The test section was made up of 19 Inconel tubes, each 0.587 inch in diameter and $19\frac{1}{2}$ inches long. A spacing between rods of 0.074 inch was maintained with two sets of warts or spacers, a set about $6\frac{1}{4}$ inches from each end. The warts were made of Al_2O_3 about 3/8 inch long, 0.10 inch wide, and 0.074 inch thick. The test section had an unheated length of 25 inches preceding the heated section. A flow diverter was located in the unheated portion about 14.5 inches upstream of the heated section. The flow diverter consisted of a

ring of Al_2O_3 segments around the bundle which blocked the space between the bundle and pressure tube wall and therefore diverted all the coolant flow to the inner part of the bundle at this location. Each of the heater rods had a thermocouple installed at its downstream end to measure the average inside wall temperature. The test section was installed in a vertical 3.25-inch ID pressure tube.

One of the two DC power sources of the Thermal Hydraulics Laboratory furnished power to the test section while the other power source was connected to an electrically heated steam generator which delivered steam-water mixtures to the test section.

Thirty-four of the experiments were made with steam-water mixtures at the test section inlet. The system pressure was either 1000 or 1200 psia. A summary of the approximate range of variables is given below.

Mass Flow Rate lb/hr-ft^2 x 10^{-6}	Inlet Quality %	Outlet Quality %	Boiling Burnout Heat Flux Btu/hr-ft^{2}x10^{-6}	No. of Experiments
0.5	0-30	27-48	0.28-0.58	15
1.0	0-27	21-35	0.25-0.80	13
1.5	0-12	14-23	0.48-0.77	5
2.0	0	12	0.79	1

In addition to the above experiments, eight were made with subcooled inlet coolant.

Analysis of the data has not been completed; however, several general observations were made during the course of the experiments which are worth noting.

The first seven experiments, which were at flow rates of 500,000 and 1,000,000 lb/hr-ft^2, had boiling burnout heat fluxes different from those predicted from other experiments with similar test section but with subcooled inlet conditions. In particular, the burnout heat fluxes at a flow rate of 500,000 lb/hrft^2 were lower than were expected in the heat flux range of 300,000 to 400,000 Btu/hr-ft^2. Furthermore, the occurrence of boiling burnout was sudden, occurred on most if not all the rods and the temperature excursions reached 300 to 400 F before the power to the test section could be reduced. Two experiments were then made at the flow rate of 500,000 lb/hr-ft^2 but with subcooled inlet coolant conditions. These gave boiling burnout heat fluxes of nearly 700,000 Btu/hr-ft^2, actually higher than were predicted from the other experiments, and much higher than those of these experiments with quality at the inlet. In fact, an experiment with about 10 F subcooling had a boiling burnout heat flux of about 660,000 Btu/hr-ft^2 at an outlet quality of about 35% which may be compared with an experiment with a nominal zero inlet quality (actually a fraction of a percent), which had a boiling burnout heat flux of only 340,000 Btu/hr-ft^2, and an outlet quality of 19%. In other words, increasing the inlet enthalpy by only a little more than 10 Btu/lb caused a reduction in the boiling burnout heat flux of nearly one-half, and, as a result of the lower power to the test section associated with the lower flux, the exit quality was reduced by nearly one-half. This gives a large discontinuity in the burnout flux-exit enthalpy relation at the zero quality inlet coolant point.

It is probable that this behavior lies in the fact that the steam generator and the piping from the steam generator have a large volume and when this volume contains some steam, which is compressible, it permitted flow surges in the test section which initiated boiling burnout. To evaluate this possibility, a valve was placed in the coolant line just upstream of the test section and across which a 150 to 200 psid pressure drop was established. Operation with such a pressure drop definitely increased the boiling burnout heat fluxes. At the flow rate of 1,000,000 lb/hr-ft^2, the burnout heat flux increased 5% at zero inlet quality and nearly 50% at 12% inlet quality. At the flow rate of 500,000 lb/hr-ft^2, the discontinuity was not eliminated but was apparently shifted to a higher inlet enthalpy point. The boiling burnout heat fluxes at inlet qualities beyond the discontinuity were increased slightly, about 5%, when the valve was used. Use of the pressure drop across the valve, however, did reduce the severity of the temperature excursion upon the onset of boiling burnout and operation of the test section was continued in boiling burnout.

The significance of the behavior just described is the fact that the coolant supply characteristics to a test section or a nuclear reactor fuel element have a strong bearing on their boiling burnout characteristics. This point must be considered in applying laboratory data to a reactor.

Subsequent experiments were all done using a 100 psid pressure drop across the valve. Eighteen of the experiments were done at 1000 psia system pressure; the other 24 were at 1200 psia system pressure. The data show an inverse flow effect, i.e., boiling burnout increased with decreasing flow. This effect was less pronounced at low outlet qualities, and a cross-over is indicated, although not conclusively shown, at about 5 to 10% outlet quality. The data show quite a pronounced outlet enthalpy effect, the boiling burnout heat flux decreasing quite strongly with increasing outlet quality over the range covered. The boiling burnout heat fluxes at 1200 psia were lower, about 10%, than those at 1000 psia.

Dome Seal Type Closure. Results of the first few "slow heat" cycles performed on the dome seal were reported last month but are repeated here for clarity of presentation.

Strong evidence points to the surge of high temperature water (570 F) as the major contributor to dome deflection. However, a minor cause of deformation remains, as indicated by the following data:

(a) "Slow heat" cycles 1, 2, 3, 6, 8, 9, 10, 12, and 13 had no detectable deformation under 10X magnification.

(b) Following cycles 5, 7, and 11, an increase of 0.007, 0.006, and 0.008 inch, respectively, were measured.

There was no measurement after cycle 4. Following the 13th cycle, the test series was interrupted to check a modified hold-down mechanism. It is believed that the intermittent deflection reported above is related to the method of retaining the dome and dome stem. This theory will be tested by modifying the present holding mechanism to incorporate a constant force, Belleville spring. As previously mentioned, the hold-down mechanism has not been considered as an item of evaluation since basic performance of the seal was desired and a horizontal, oriented seal would require a different holding concept than is used on this design.

The sealing properties continue to be outstanding. There is still zero leakage in the collection system. In other tests the leakage collection system has measured 0.05 ml of leakage over a 78-hour period, and it is believed that it could detect smaller leak rates.

The cause of increasing disassembly torques, mentioned in the last report, has been determined to be a result of lubricant breakdown and was easily controlled by application of new lubricant.

Testing on the modified hold-down mechanism is continuing.

11. **Advanced Reactor Concept Studies**

Fast Supercritical Pressure Power Reactor. Further 18-group diffusion theory calculations show that control and safety rods positioned in the moderator region will have sufficient strength. A control strength of 50 mk was obtained with a density of 0.007 gm natural boron/cc in the moderator region. Six control rods, 2 cm in diameter, positioned in the moderator region and constructed of stainless steel with 0.5 w/o natural boron, have a strength of about 13 mk which should be sufficient for shim control. It thus appears that there will be no difficulty in obtaining a desired control or safety rod strength.

Calculations still indicate that a partial loss of hydrogen in the moderator region causes a reactivity loss. A 6 mk loss in reactivity was calculated for a 50% hydrogen loss. This permits use of up to $\sim$ 50 v/o H_2O in this region for cooling without danger of positive density coefficients.

In making up the fuel elements for the second and third pass cores, the basic dimensions from the physics calculations were changed to obtain a proper power split. It will also be necessary to increase the enrichment slightly in the second core zone to flatten the power distribution in that region.

For the second and third pass cores, fuel elements are as follows:

	Second Core	Third Core
Total number of tubes	3576	7578
I.D.	0.2426"	0.2000"
O.D.	0.2875"	0.2500"
Mass flow rate lbs/hr-ft^2	3.4×10^6	2.25×10^6
Max. heat flux B/hr-ft^2	550,000	405,000
Avg. heat flux B/hr-ft^2	366,000	240,000
Percent power in dept. UO_2(%)	0	24.4
Dimensions:		
Base end	14.4"	10.95"
Truncated end		1.73"

Currently the physical characteristics for the first core and blanket fuel elements are being determined.

Much of the conceptual design effort during the month was related to control and safety considerations and the postulation of accidents to determine the effect of these on the reactor core and plant, and on containment provisions.

Each of the two moderator regions is composed of six slab-like tanks arranged to form a hexagon. Each tank is 3 cm thick and slightly longer than the core and extends the full width of the region in which it is located. Yttrium hydride bars, extending full core length, occupy approximately half the tank volume. Besides minimizing excursions which might arise from loss of liquid moderator, these solid moderator bars also serve as structural bracing and flow guides. The moderator regions are cooled by water which also provides half the neutron moderation.

A safety control rod and an operating control rod are located on opposite ends of each tank in the inner moderator region. The safety rods are located in the moderator water inlet tubes and the operating control rods are in the outlet tubes.

All the 0.5% boron stainless steel control rods are 3/4-inch OD and have active lengths equal to the height of the core. Three of the safety rods are of similar design except that they are loaded to (about) 1 w/o B-10. The other three safety rods are composed of small boron-containing ceramic balls in thin-wall metal tubes. The bottom closures of these tubes and, possibly, the tubes themselves, will be designed to "disintegrate" with very small strains, thus permitting (at least) some of the poison to be inserted in the core even if the safety rod channels collapse, and prevent rod insertion. A poison injection system for the moderator coolant is also being considered as a back-up safety device.

A rigid, two-foot thick, grid plate above the core serves to locate and support the core and blanket elements, the moderator tanks, and the pipe jumpers. A two-inch thick plate at the top of the fuel elements is bottomed against the support plate and then fastened by bolting from above.

The two reactor tanks (one "spare") are located in an oval-shaped pit. The pit contains borated or natural water which acts as shield and thermal reservoir for the vapor-suppression containment. Use of borated water is being considered to avoid the possibility of achieving thermal neutron criticality following a reactor accident in which the core might be dispersed into the water-filled volumes.

Plutonium-Fueled Rocket Reactor. Preliminary calculations are under way to set core parameters for a plutonium-fueled nuclear rocket reactor of about 1000 Mw power. An unclad tungsten-base cermet fuel is being considered; use of an unclad cermet would provide appreciable savings in reactor size and weight if it should prove to be a practical structural material.

To obtain the high power density required for reactors of this type (cross sectional area of the core will be on the order of one square foot), provision must be made for a very large number of small coolant channels. A hexagonal honeycomb fuel arrangement is being considered. It appears that such a fuel piece could feasibly be made by Dynapak techniques. To provide the requisite heat transfer characteristics, the hexagonal coolant channels would have to be on the order of 0.020" to 0.040" on a side; the cermet-fuel web between channels would be about the same thickness.

Plutonium Fuel Spacecraft Reactor. Document HW-77066, describing work done on the preliminary concept of the PFSR, was issued as an informal document.

The PFSR as described in HW-77066 is a compact reactor (a cylinder 20 cm in radius and 40 cm long) for use in space with a 20,000-hr core life. The control requirements to maintain criticality over the core life is 0.25 to 0.35 Δk. A spectral shift control region is being studied as one way to achieve the gross reactivity control necessary without using mechanical devices. The control region is a cylindrical shell, consisting of yttrium hydride with $B_4{}^{10}C$ coating each surface of the shell, which separates the reactor core into two parts. The yttrium hydride slows down the neutrons so that they may easily be absorbed by the boron^{10}. As the reactor fuel burns up, the hydrogen is released from the control region at a controlled rate reducing the amount of neutron moderation and, hence, reducing the neutron capture by boron^{10}.

HFN diffusion theory calculations, using 16 energy groups in one dimensional cylindrical geometry, have shown that the necessary control strength may be achieved. In these calculations the core or fuel volume remained equal to the original PFSR. The control strength was defined as the difference in k for calculations with and without hydrogen in the control region. Control strengths up to 0.35 Δk were obtained with control regions $\leq$ 4 cm thick. The core is described as follows:

		Radius
Reg 1	Core	~ 10 cm
Reg 2	$B_4{}^{10}C$	
Reg 3	Yttrium hydride	~ 13.5 cm
Reg 4	$B_4{}^{10}C$	
Reg 5	Core	~ 22.0 cm
Reg 6	Beryllium Reflector	32.0 cm

These calculations require verification with transport theory calculation. Burnup calculations are also necessary to determine the rate at which hydrogen should be released from the moderator.

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.

Single Crystal Molybdenum. Irradiated molybdenum single crystal tensile specimens were aged for 16 hours at 250 C (482 F) and tested at room temperature. A number of significant observations were made concerning these tests: (1) a pronounced yield point was developed only in the crystals containing 20 ppm carbon, and the magnitude of the load drop upon yielding increased with greater exposure, being > 30% of the load at the upper yield point at an exposure of 10^{19} nvt (E> 1 Mev); (2) the critical resolved shear stress of the intermediate- (150 ppm carbon) and high-carbon (450 ppm carbon) crystals was increased 25-50% as a result of aging, the increase being greater at the higher exposure; (3) the ductility of the intermediate- and high-carbon crystals was substantially decreased after aging, in concurrence with results obtained with irradiated and aged polycrystalline molybdenum; (4) gross slip traces were observed in the aged low-carbon crystals immediately after yielding; in one case of a crystal with an exposure of 10^{19} nvt, the deformation was so localized as to result in a displacement of one-half of the crystal of about 0.5 mm. These results are believed to be yet another manifestation of the interaction between interstitial impurity atoms and point defects produced by reactor irradiation. The manner in which the number of operative slip systems is reduced by irradiation and by post-irradiation aging treatments has not yet been resolved. Interpretation of results is continuing.

Lattice parameter measurements on irradiated molybdenum crystals have been continued using the modified single crystal diffractometer described previously. Variations in lattice parameter values from point to point on the crystal surface show that the starting material was not physically homogeneous. Diffraction maxima from many regions of the crystal consisting of two or more overlapping peaks separated by several minutes of angle indicate large subgrains. In an attempt to eliminate this substructure, an unirradiated crystal was annealed at 2000 C (3632 F) for one hour, cooled quickly to 1500 C (2732 F) to prevent carbon precipitation, then cooled slowly to room temperature. The crystal face was then electropolished and used for lattice parameter measurements. Six measurements at different points yielded an average value of 3.14707 A, with a mean deviation of 0.00003 A. Since this is within the predicted limit of precision of one part in 10^5, it is reasonably certain that the variations observed in the as-grown crystals are real variations and not experimental errors.

Measurements on low carbon (15 ppm C) and medium carbon (150 ppm C) crystals after an exposure of 10^{19} nvt (E > 1 Mev) show little if any change from the preirradiation values. For high carbon (450 ppm C) crystals irradiated to the same exposure the diffraction peaks are much broadened and diffused. Lattice parameter measurements on these crystals are not yet complete.

Polycrystalline Molybdenum. A preliminary investigation of the effect of reactor irradiation on the electrical resistivity and annealing kinetics of high purity (99.99%) molybdenum is in progress. Eight resistivity specimens, consisting of 30 cm lengths of 0.25 mm diameter wire, were annealed for one hour at 1300 C (2372 F) and encapsulated in a helium-filled aluminum capsule and submitted for irradiation to a goal exposure of 10^{18} nvt (E > 1 Mev). This investigation is intended to yield information on the nature of the complex recovery stages observed in irradiated impure molybdenum, with a primary purpose of establishing the mobilities of single vacancies and interstitials at the temperature of irradiation, i.e., 40 C (104 F).

Additional one-eighth inch diameter polycrystalline tensile specimens are being prepared for irradiation, as part of an experiment designed to establish the effect of irradiation on the yielding process. Planned exposures range from 10^{17} nvt to 5×10^{18} nvt.

Polycrystalline molybdenum, in the form of one-half inch diameter rod, has been successfully upset to 0.75-inch diameter in four stages at 500 C (932 F) and swaged to one-half-inch diameter at 600 C (1112 F) to produce light stored energy specimens. The high swaging temperature was necessary to avoid cracking. The possibility of preparing stored energy specimens from cold-rolled sheet is being considered.

Molybdenum foils having three controlled carbon levels, approximately 10 ppm, 150 ppm, and 450 ppm, were annealed two hours at 900 C (1652 F) after irradiation to 10^{19} nvt (E > 1 Mev). X-ray examination showed the lattice parameter and line breadths to be essentially identical to the corresponding values for unirradiated material. There were, however, pronounced changes in the relative intensities of the diffraction peaks in the samples containing 150 and 450 ppm carbon. A duplicate set of foil samples was annealed in the same manner, and the intensity changes were also observed in these samples. Electron microscope examination of the thinned foils showed a very small number of dislocation loops, in contrast to the large number observed to be present after a 750 C (1382 F) anneal. The marked decrease in observable defects and the change in diffraction peak intensities is presumed to result from a partial recrystallization. The irradiated foils containing 10 ppm carbon did not display the change in diffraction peak intensities and apparently did not recrystallize.

Defect structures have been observed in irradiated molybdenum foils by transmission electron microscopy. Deformation of the irradiated foils causes dislocations to move in straight channels. The moving dislocations interact with the defect structures and sweep them from the channels. Examination of the electron diffraction patterns has established that the channeling does occur in the close-packed direction, i.e., the [111] direction. A letter to the editor, entitled "Dislocation Channeling in Neutron Irradiated Molybdenum," has been prepared for submission to the Journal of Applied Physics.

Nickel. Nickel foils, 0.002 inch thick, of purities corresponding to 99.4, 99.6, and 99.97% have been prepared for quenching and irradiation studies. Specimens which have been annealed at 700 C (1292 F) for one hour as well as unannealed specimens with 30% cold work will be encapsulated for reactor irradiation. Techniques for electro-thinning these foils are being investigated.

Electrical resistivity studies on quenched-in defects in nickel wires have been delayed for repair and recalibration of instruments. The quenching chamber has been modified in the interim to permit operation at much higher helium pressures, i.e., 500 psig as opposed to 15 psig with the previous arrangement. This modification will permit attainment of much higher quenching rates of the order 10^4 degrees C per second (2×10^4 degrees F per second).

Additional nickel wire specimens are being prepared. Wires of 0.125 mm diameter and 1.0 mm diameter are being drawn for use as electrical resistivity and internal friction specimens, respectively.

Rhenium. Attempts to draw 0.125 mm diameter rhenium wires from as-received 0.5 mm rhenium wire have been unsuccessful. The cause for this was found to be die abrasion due to the presence of manifold transverse cracks in the surface of the wire. This cracked layer was removed by electrolytic polishing in an electrolyte consisting of 70 ml ethanol, 35 ml perchloric acid, and 10 ml n-butoxy ethanol (butyl cellosolve). The cracks were found to extend about 0.1 mm into the wire.

Experimental Techniques and Apparatus. An electrolytical machining device has been fabricated for the purpose of strain-free machining of reduced sections on single crystal tensile specimens. The device consists of a 12-inch diameter stainless steel wheel which rotates while partially immersed in an electrolytic solution. A potential is applied between the wheel and the crystal which rotates in opposition to the wheel and in very close proximity to it. The electrolyte is carried on the wheel and contacts the specimen, resulting in a polishing action. Two wheels are presently available, one a three-inch wide flat wheel for initial polishing of the crystal to an axially symmetric, uniform diameter. Final machining is performed with a contoured wheel having a one-inch flat section and one-fourth inch radii. This wheel produces the reduced section in the crystal. Evaluation of this device is continuing, and modifications of the crystal holder are being made.

An electrolytic jet polisher has been utilized for cutting wafers from a single crystal of molybdenum, a necessary initial step in the preparation of foils for transmission electron microscopy. Surfaces cut in this manner are free of cold work but are highly irregular. The smooth parallel surfaces necessary for final thinning could not be obtained by conventional electrolytic polishing. Sectioning of crystals by spark machining has therefore been investigated. This method will produce a smooth but worked surface on the specimen. The worked surface layer can then be removed by an electrolytic polishing wheel utilizing 10% HNO_3 with a current density of 16 amp/in^2. Sections of single crystals of molybdenum 0.020 inch thick have been produced in this fashion. The next stage of specimen preparation consists of "dimpling" the section with an electrolytic jet.

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.

Phase Transformation, Grain Growth, and Deformation and Fracture Studies. The effects of various variables on the alpha plutonium grain size are being investigated. Stress, both tensile and compressive, applied during the $\beta \longrightarrow \alpha$ transformation may affect grain size. An initial experiment in line with this will be to apply a bending moment to a plutonium rod while the $\beta \longrightarrow \alpha$ transformation occurs. The influence of both tensile and compressive stress should be readily apparent from examination of the longitudinal microstructure. Another technique which may result in grain coarsening involves progressive heating of a plutonium rod thru the $\beta \longrightarrow \alpha$ transformation (Bridgeman technique). A gradient furnace for this purpose is being constructed.

Tensile testing of 0.54 w/o Al stabilized plutonium is in progress. Results thus far show no evidence of the Portevin-Le Chatelier effect at 400 C (752 F). The specimen dimensions are one-half-inch diameter by $3\frac{1}{4}$-inch gage length. Specimens were given a two-hour 450 C (842 F) homogenization treatment prior to testing. Testing was performed with a 0.015 in/min crosshead speed.

It has been established that the creep rates of plutonium during the $\alpha \longrightarrow \beta$ and $\beta \longrightarrow \alpha$ transformations under compressive loads of 1000 and 2000 psi are much greater than the rates of either the alpha or the beta phase. The creep rates of the alpha and beta phases were 5×10^{-7} in/in/min, and 8×10^{-7} in/in/min, respectively, during a 1000-minute period, at 100 C (212 F) and under a compressive load of 2000 psi.

A 9% volume increase is associated with the $\alpha \longrightarrow \beta$ transformation and a similar decrease is associated with the $\beta \longrightarrow \alpha$ transformation. Therefore, the $\alpha \longrightarrow \beta$ transition is inhibited by a compressive load whereas the $\beta \longrightarrow \alpha$ transition is accelerated. The rates of both transformations rapidly decrease as the equilibrium transition temperature is approached. Thus, more creep might be expected during the $\alpha \longrightarrow \beta$ transformation than the $\beta \longrightarrow \alpha$ transition, at temperatures closer to the transformation temperature, and under a greater compressive load. As expected, the creep during the $\alpha \longrightarrow \beta$ transition was significantly greater than the creep during the transformation under compressive loads of 1000 and 2000 psi. Indications are that this is also true for 100 psi. Results to date are insufficient for evaluating the effect of transformation temperature on the creep. The creep is much greater for larger compressive loads during both $\alpha \longrightarrow \beta$ and $\beta \longrightarrow \alpha$ transitions.

Studies have been undertaken to determine the effect of grain size upon the compressive creep rates of both the alpha and beta phases of plutonium as well as those active during the phase transformation.

The completely indexed diffraction pattern of alpha plutonium has been made available by Zachariasen and Ellinger in a paper, the publication of which is currently in process. This has afforded the opportunity for a preliminary qualitative evaluation of the effects of severe deformation by rolling upon the crystallographic characteristics of this material. Data which have been derived from some of the stronger lines of the various patterns obtained to date are presented in the following table. The Zachariasen and Ellinger pattern was made from an as-cast button. A comparison of the observed and calculated line intensities indicates that very little, if any, texturing was present. The two patterns obtained from an etched cast plate and the etched surface of a similar casting after 90% reduction by rolling at about 25 C (77 F) are compared with the Zachariasen and Ellinger results.

Comparative Diffraction Data
of Alpha Plutonium in Three Different Conditions

hkl	Zachariasen & Ellinger				As-Cast (HL)		Rolled (HL)	
	2θ	I_{calc}	I_{obs}	I/I_o	2θ	I/I_o	2θ	I/I_o
013	31.08	56.3	56	20	31.02	8	31.08	12
$11\bar{3}$	32.20	128.7	122	43	32.18	16		
201	32.28	64.9	64	22.5			32.30	100
004	33.36	98.4	105	37	33.30	9	33.35	30
$20\bar{3}$	34.66	142.1	152	54	34.60	8	34.65	25
020	37.28	270.6	246	87	37.25	100		
211	37.42	301.4	284	100			37.38	56
014	38.36	105.6	121	43	38.37	13	38.40	20
$11\bar{4}$	38.60	72.6	86	30				
024	50.85	60.2	70	25	50.85	11	50.88	8
$22\bar{3}$	51.80	91.2	95	33	51.78	12	51.80	5

The inconsistencies noted in the low angle region of the pattern persist throughout. In addition, the back reflection region gives indications of significant changes in lattice parameter values resulting from severe deformation. A quantitative evaluation of this phenomenon must necessarily await the application of more precise techniques.

There is as yet no evidence of significant line broadening which would indicate the existence of residual stresses and/or extensive grain fragmentation in the deformed material.

A series of castings is in process which will afford material for the complete evaluation of the effects of varying amounts of reduction at three temperatures in the alpha range. Two different material starting conditions are planned. One of these will be a quench to the alpha from the epsilon region immediately after solidification and the other is to be an isothermal transformation in each phase after cooling from the melt. These extremes should result in a maximum difference in alpha grain size.

An initial attempt has been made to determine the effects of annealing cold worked alpha at temperatures above the normal stability maximum. A sample was hydrostatically pressed at 180,000 psi and then heated to 180 C (356 F) for seven hours. It was cooled under pressure to below 100 C (212 F). This treatment should not have resulted in any transformation to the beta phase. Metallographic evaluation is not yet complete.

Ten kilograms of as-reduced plutonium were ingotted and cast into appropriate shapes for deformation and phase transformation studies. The analytical data are not yet available. The quality of the metal was quite high as indicated by metallography and density measurements. The as-cast density was 19.59 - 19.65 g/cm^3. The metal contained few visible inclusions at 250X and appeared to be nearly free of the Pu-Pu_6Fe eutectic and Pu_3C_2 (zeta phase) which are commonly observed in plutonium. On the other hand, the metal contained an estimated one-half to one v/o of microcracks.

Four kilograms of the metal were used to prepare specimens for studying the mechanisms of phase transformations. A series of these specimens is to be used in establishing grain morphology and growth rates at different $\beta \longrightarrow \alpha$ transformation temperatures. A second series is to be utilized in determining the effect of beta plastic deformation on the $\beta \longrightarrow \alpha$ transformation and in studying the mode of transformation under applied stresses of 1000 to 12,000 psi. These studies will provide information on the mechanism of the $\beta \longrightarrow \alpha$ transformation. Five hundred grams of samples will be used for investigating crack formation and propagation during transformation.

Two kilograms of plutonium have been alloyed with 10 a/o aluminum for investigating the Portevin-Le Chatelier effect in stabilized delta plutonium. The remainder of the metal was prepared for alpha rolling, extended deformation during phase transformation, fractography, electron microscopy, and grain growth studies.

A standard stereographic projection of the crystal structure of alpha-plutonium was desired to aid proposed studies on this metal. A [010] projection was constructed from known unit cell data by calculating the length and direction of the appropriate vectors in reciprocal space. A total of 38 poles is represented by points on a stereographic projection, and a like number may be obtained by the symmetry relations. A table giving the angular relationships was also compiled.

A computer program for calculating reference powder x-ray patterns directly from single-crystal data has been obtained and will be utilized in the same proposed study.

E. CUSTOMER WORK

1. Radiometallurgy Laboratory

Examinations and Measurements

Samples were removed from NaK capsules 17 and 18 and were visually examined.

An N-Reactor outer tube element was measured for residual stress after post-irradiation examination disclosed the presence of a second phase in the uranium.

Metallography was done on the in-reactor part of the failed Inconel 105 DR Gas Loop. Metallography was also completed on eight nickel base alloy control samples.

Rockwell hardness tests were reported on three Zircaloy-2 samples from process tube 1643.

Metallography was used to determine extent of hydriding in fretting corrosion.

Three photomosaics and four diameter composites of cross sections of ceramic fuel elements were photographed at 75X (RM's B-603 and 678).

PRTR element #1501 (nested, tubular, UO_2) was examined for extent of sintering and degree of hydriding.

Microdrill samples were obtained from cross sections of samples from GEH-14-20, 22, 27, 66, 68, 69, 83, 85, 87, 88, and 91. Macrophotographs of each specimen were made after drilling (RM C-620).

Thirty-four samples were removed from 10 rods from Pu-Al PRTR element #5108. Samples will be processed to determine burnup and fission product distribution (RM C-757).

Burst test specimens were removed from sections of PRTR Zr-2 process tubes 0720, 5540, and 5702. Diameter and tube wall thicknesses of each specimen were recorded (RM C-853).

Forty irradiated metal or ceramic fuel samples were dissolved for burnup analysis, nitrogen analysis, or cladding recovery. Thirteen uranium samples were chemically declad. Densities of four samples were determined.

The cause of failure of the electrical resistivity leads in Capsule GEH-4-82 was determined (RM C-904).

Thickness of corrosion coupons from quadrants GEH-20-48, 64, and 82 was measured (RM C-113).

Metallography of two irradiated thorium samples B-2 and S-3 was completed after a 100-hour anneal at 700 C. Replicas were obtained for electron microscope studies (RM C-501).

Samples were removed from NaK capsules 3A24 and 3U31 and measured. Seven other capsules were stored for "cool down" because of high radiation readings encountered (RM C-554).

Twenty quadrants of structural materials were received, identified, and pre-test measured. Six stainless steel samples were tested. Tests were terminated because of inadequacy of the grips at loads exceeding 4000 lb. New grips have been received and are being modified for remote operation (RM's C-500 and 801).

The results and significance of these measurements will be found in the reports of the appropriate Research and Development organizations.

Equipment

High Temperature Tensile Grips. Fabrication of the Inconel-X tensile grips was completed in Technical Shops. Following a dimensional inspection, the grips will be tested in the Physical Mechanical Properties Testing Cell.

NaK Capsule Opening Equipment. Design drawings were completed for a revision of the milling machine in "F" Cell to make it suitable for certain NaK capsule opening operations. Equipment cost estimates were requested.

Orbital Grinder and Polisher E-65. The grinder has been completed and is now under test to determine optimum operating conditions, and limitations and capabilities. Preliminary tests are very promising.

Stereo Zoom Microscope. The stereo zoom, hot cell microscope drawings were received for Hanford approval.

Light Duty Master-Slave Manipulators. Fabrication of two light duty, master-slave manipulators continued in the vendor's plant. Delivery of the two manipulators is expected in June.

Irradiated Structural Materials Testing Facility. Preliminary scoping of the Irradiated Structural Materials Testing Facility, B-504-H, was initiated.

X-Ray Diffraction Cell. The lead brick shielding around the double crystal x-ray diffraction unit was completed this month. The unit has been operated with irradiated samples to establish operating procedures after the extensive modifications.

Vacuum Fusion Furnace. Demonstration of the remote induction furnace at the vendor's plant was witnessed May 1 and 2. Heating characterisitcs of the furnace were very satisfactory. An 0.5 in. diameter by 2 in. long uranium sample was heated to liquid state in 21 seconds. Equipment and controls responded quickly and there was no heating of the mounting plug or supporting equipment. Shipment of the equipment is expected during May following certification of the integrity of the vacuum system by an independent testing laboratory.

Building Operations

Cask #26 has completed two round trips between Hanford and NRTS in Idaho. Casks 16 and 22 were shipped to NRTS.

Local cask shipments were as follows: five to PRTR, fifteen to 325 Building, three to 105C, three to 105 KE, two to 105KW, and one to 105F. Forty-three "milk pail" and three "Gatling Gun" (30 cans) waste cask runs were made.

One storage rack was removed from the basin and stored in the burial ground to make room for construction of the Burst Test Facility canal. Also, in preparation for this work, a coffer dam was installed in the basin.

Both overhead cranes were inspected by an off-site certified inspector.

Miscellaneous

A paper titled "Criteria for Safe Handling of Irradiated Plutonium in Beta-Gamma Cells" was co-authored by C. L. Boyd, J. G. Bradley, and L. G. Faust, and submitted to the 11th Hot Laboratory and Equipment Conference Program Committee.

2. Metallography Laboratories

Metallographic assistance to NRD for the welding of supports to N-Reactor fuel elements has continued. Since process variables have been reasonably well defined, further work may be confined to quality control.

Two heli-arc welds, one of stainless steel to niobium and one of Zircaloy-2 to niobium, were examined for weld characteristics and micro-structure. The stainless steel to niobium resulted in brittle compounds in the weld which caused it to fracture as the piece cooled. This weld was uneven and of general, poor quality. The Zircaloy-2 to niobium weld appeared strong and uniform. The weld region consisted of a non-uniform mixture of inter-metallic compounds.

A gas-pressure bonded test sample of a simulated Zr-2 clad uranium fuel element with a niobium cap insert was examined for bond integrity. All bonds appeared good. A few unbonded areas were limited to the actual size and location of unwanted inclusions in the bond lines. The niobium to Zircaloy diffusion zone proved particularly difficult to etch for proper outline. The etch finally used was one containing 80 cc HCl, 20 cc HNO_3, and 5 drops of HF. The sample was swabbed for about 10 seconds with this acid mixture after receiving a mechanical polish. This same etching solution can be used electrolytically with 12 volts D.C. to electro-polish Zircaloy. The resultant surface shows all grain boundaries and also the concentration gradient of alloying constituents within the Zircaloy.

During vacuum cathodic etching the temperature of the specimen being etched rises, although an attempt is made to cool the specimen by cooling the cathode upon which it rests. During the month, an effort was made to determine the temperature that a sample might reach during the etching process. Samples of uranium shot were cast in a "Wood's Metal" alloy having a melting point of about 65 C. The 3/8 in. diameter by 1/2 in. thick specimen was then polished. During cathodic etching the voltage was advanced in 0.5 kv steps with five minute intervals between steps to allow the sample to reach thermal equilibrium. At 2.5 kv and 1.7 ma. no melting of the sample surface was detected. At 3.0 kv and 2.0 ma., however, the polished surface was observed to become slightly fluid. Examination of this specimen after removal from the etcher disclosed that no apparent melting took place on the surface resting on the cooled cathode. This information when coupled with the thermal conductivity and sample geometry may enable a rough estimate of the sample temperature during the etching process.

An examination of Edison-type Monel-R resistance temperature detectors (RTD's) was made in an attempt to determine the cause of the corrosion noted on them upon removal from K and DR Reactors. General pitting corrosion was located on the barrel although some pitting was located on the stems. A slight intergranular corrosion tendency was noted in some of the pits. Whether the corrosion is due to normal reactor environment or caused by a reactor decontamination process has not been determined.

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

3. N-Reactor Design Testing

N-Reactor Charging Machine. One drive roller, which had the rubber stripped from it during use, was replaced. Apparently the production drive rollers were not of the same quality as the shop fabricated prototype on which the testing was performed. The problem will be investigated further.

The end thrust stabilizer system has again been tested in its modified form to determine if it is adequate to resist a 20,000 pound end load on the machine. The movement of the main frame with respect to the lower frame was measured at 0.085 inch when the main frame was fully loaded. This amount is now considered acceptable in light of the modifications that have been made to the vertical lift worm gear transmissions.

A major portion of the testing performed during the month dealt with Design Test 22-B, which evaluates the effect of the charging machine on fuel elements.

When typical fuel elements were charged into the Test 22-B mockup with a back-pressure of approximately 500 pounds (75 psig) and a velocity of 60 fpm, flattening of the self-supports occurred up to a maximum of about 0.025 inch. Flattening of reduced severity also occurred when fuel elements were charged at (1) reduced velocity (15 fpm), (2) reduced back-pressure (30 psig), and (3) reduced velocity (15 fpm) and pressure (30 psig). The Test 22-B mockup was modified to a more prototypical condition by replacing the receiving magazine with an NPR process tube. Ten charges have been made into this modified mockup with no more than 0.005 inch flattening of the self-supports, the majority of which is attributed to measurement error. It should be noted that half of these charges were made into an empty process tube (no extra column load) and the other half into the process tube which contained the previously charged column (column load of a fuel column only). In both cases, the back pressure was much less than that used when the feet were damaged. Further investigations are under way to determine the cause of the flattening.

4. Special Plutonium Fabrications

Al-Pu Fuel for Corrosion Tests. Thirty-eight fuel elements were made to complete this work; a thermocouple well for in-reactor temperature measurements was placed in the center of six of the elements.

1103501

Corrosion which occurred during autoclaving of some of the X8001 cladding alloys was apparently a surface phenomenon since it did not recur when the alloys were re-autoclaved after removing 0.010 in. of the surface. In review, this work consisted of co-extrusion cladding an Al-8 w/o Pu alloy with three different Al-Ni-Fe cladding alloys. The cladding was metallurgically bonded to the core. The significant development in this work was cutting the long extruded rod into short lengths, counterboring the core material, followed by successful decontamination of the fuel prior to welding the end caps in place.

UO_2-PuO_2 Fuel for the PCTR. This fuel consists of 0.90 w/o PuO_2 in depleted UO_2 in the form of 1/2-inch diameter pellets. The pellets will be clad in Zircaloy; 4000 pellets were sintered and 2000 pellets were ground to final diameter.

High Exposure Al-Pu Fuel for Physics Tests. Work was completed on the 1000 rods required for this test. The alloy was 2 w/o Pu which contained 16.5 w/o Pu-240. Each three-foot rod contained 6-1/2 grams of plutonium and was clad in zirconium.

5. EBWR Fuel Elements

Preparation of PuO_2 from 60 kg of Pu metal was completed. This work involved burning Pu to oxide, ball milling, and screening to -325 mesh.

The first Zircaloy tubes for test standards were received. Standard notches will be machined in them, and they will be returned to the vendor for use during accpetance testing of the remaining tubing.

A pit was dug and concrete walls were poured to house equipment for the experimental fabrication of EBWR fuel rods.

EBWR Plutonium Fuel Elements. Uranium dioxide, prepared by burning depleted uranium (0.2% U-235) to U_3O_8, and hydrogen reducing to UO_2, was densified by pneumatic impaction. One-third more impact energy was required for impaction than with other previously tested types of UO_2. Because of the high impact force required and the limited capacity of the 308 Building impaction machine, it will be necessary to treat the material by pneumatic impaction in the large machine in the 325 Building prior to blending it with PuO_2 and again impacting it. This method of achieving high density material has been demonstrated for several types of UO_2, and is relatively insensitive to the properties of the starting material.

T. W. Woodfield
for Manager
Reactor and Fuels Laboratory

PHYSICS AND INSTRUMENTS LABORATORY

MONTHLY REPORT

MAY 1963

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

Analysis of N-Reactor Experimental Data

The draft of the formal document on N-reactor physics work (HW-72096) is complete except for a few graphs and pictures. The neutron temperature problem discussed last month has been resolved. Since a relatively small error in the measured 1/v flux in the graphite has a large effect on the resulting temperature but a negligible effect on "f", there are no problems in normalizing between experiments with different graphite densities. All experiments have now been normalized to a standard condition which corresponds to the PCTR mockup lattice.

Optimization of Re-tubed Lattices

Vertical traverses have been completed in the "C" exponential pile containing dry CVIN (overbore) fuel. The measurements were performed both without safety rods and with six rods. The data on the overbore experiments are being analyzed.

K Lattice PCTR Experiments

Measurements are being made in the PCTR to obtain absolute k_∞ values for KVNS fuel pieces and zirconium process tubing in the K lattice. The experimental work has been completed for the wet case using a three by three cell array. The correct mass of copper has been determined to poison the central cell to a k_∞ of unity. Foil irradiations were made for determination of fine structure of the flux in both poisoned and unpoisoned cases in which the cadmium ratios of 5 mil gold were matched in all cells. Copper foils were used throughout the center cell and on the thermal column, and in addition gold foils were used in the fuel and coolant channels. Detailed flux variations have been measured in the copper poison and near the end-caps of the fuel pieces. All foil data for the above work have been processed through the computer program APDAC and checked, and are now being analyzed.

Reactivity measurements have been made in the PCTR to determine (1) the negative Δk due to either a normal or grey spline being inserted into the center cell process tube in the wet K lattice, (2) the Δk for a normal spline as a function of total exposure in the production pile, and (3) the Δk for a normal spline as a function of corrosion. Measurements were made using several irradiated and non-irradiated spline samples. One non-irradiated sample was etched uniformly in a strong caustic solution to approximately 75%, 50% and 25% of the original mass. Reactivity measurements were made on this spline at each step in the etching process to obtain Δk as a function of corrosion. The data from these measurements are now being analyzed.

Computational Programming Services

Test cases have been successfully run on both LEAP and ADDELT, the thermal neutron scattering codes imported from Harwell, England, last month. Instructions for their use are being prepared for distribution to those interested.

Additional revisions to TRIP005 this month are probably the last before the program is frozen to conform with the descriptive document, which will be published next month.

A specialized version of the reactor kinetics code, called TRIP105, was prepared to compute the reactivity by a sine function.

The Fortranslations Index was extensively revised and updated this month.

Limited core space in a portion of the CALX chain forced preparation of a more compact version of LILLEY, the routine which positions the Composite Lilley Tape. The new version, called LILLY2, functions in the same manner as LILLEY except that one of the output options of LILLEY is omitted. LILLY2 is in debug.

Scattering Law for Graphite

Several experiments performed in the PCTR studying the moderating properties of graphite show promise for checking the various currently available scattering law models for graphite. To distinguish moderating from transport effects, it is desirable to analyze the experiments with computer program S. Since a large number of transfer cross sections must be calculated as input for program S, SPECTRE is being modified to give a punched card output that is suitable as input to program S. This mechanization will also reduce the possibility of error so essential to avoid in such a comparison study.

Instrumentation

Circuit modifications were carried out by GE-APED on the gamma energy spectrometers planned for use in the N Fuel Rupture Monitoring System. The changes involved the count rate meter circuit and the DC amplifier circuit. A trip to GE-APED to witness final tests on a production model instrument was made the last week in May.

Cooperative testing was carried out with Instrument and Electrical Development, IPD, on several possible N Reactor startup instrumentation systems. Tests included pulse height distribution measurements at various neutron counting rates for three BF_3 proportional counters. Other tests were carried out at the 305 pile with the spare source range log count rate and period instruments with substitute fission counters and preamplifiers. Test results suggested a need for further instrument evaluation studies prior to N Reactor startup. Drawings of the ion chambers for the N Reactor intermediate and power range equipment were reviewed.

An oscilloscope camera and two integral assembly scintillation detectors were received for use in the N Fuel Testing Loop being installed at PRTR. An order was placed for the X-Y recorder, and requisitions were prepared for the rack cabinets and cooling blowers. Revised flow equipment bids which require control valves capable of taking full loop pressure are being reviewed. Special valves were specified in order to prevent inadvertent flow stoppage which would place full loop pressure on the sample cells and flow transmitters. Specifications were started for the two delayed neutron monitors and for the scintillation spectrometers. The solid state multichannel analyzer was returned to the manufacturer for repairs. Scheduling problems were discussed with IPD personnel.

System Studies

The N Reactor kinetics and primary loop simulation for the power range was used to determine the effects of varying the reactivity ramps, inlet temperature, instrument trip settings and time constants, initial power level, power setback rates, and scram reactivity time functions on a number of system variables. The simulation was described and demonstrated to interested NRD and HL personnel.

A three-pump model of the injection system was set up on the computer to study the effects on system stability of varying primary loop operating pressures and different controller types. Optimum controller settings were found by minimizing the integral of the squared error resulting from a systems disturbance function. Primary loop pump suction pressures were varied over the range of 100 to 1250 psia. Both proportional-plus-integral and pure integral control modes were used.

An analog simulation has been prepared for studying N Reactor kinetics over several decades of power, including startup. This simulation requires the use of analog memory techniques which do not ordinarily allow system variables to be initiated. A circuit has been devised which automatically selects either initial conditions or range scaling, depending on how far the problem solution has progressed in time.

The two-dimensional D Reactor simulation which has been under investigation for the past few months has been found to be basically in error; hence, earlier tests using the automatic D Reactor prototype controller are invalid. The simulation was being used to determine the best methods for gathering data from a production reactor. A digital computing program has been established which allows a simple measurement on the D Reactor to be used for measuring system parameters.

A D Reactor production test made to determine the effects of control rod perturbations on reactor performance may prove to be of only limited value. During early test runs involving small changes in control rod position, reactor power did not reach equilibrium in one instance following a one-inch rod movement. Although subsequent tests did not duplicate this unusual reactor response, for safety reasons continuing experiments were restricted to smaller changes in rod position. Preliminary evaluation of the data suggests that the measurement uncertainty resulting from the restricted rod movements may make the data of limited value.

Three consultations were held this month with AEC Legal and Purchasing people to write the terms and conditions of the process control computer contract. The bid package is now in its final draft, and will be sent to the prospective sellers upon its completion and our approval.

SEPARATIONS

Experiments with Plutonium Solutions

Criticality experiments with plutonium solutions were temporarily suspended in order to facilitate the final checkout of the split table machine, which is located in the second hood of the critical assembly room. The same neutron detection system is currently being used for both solution experiments and those with PuO_2-plastic mixtures. Safety requirements and equipment limit operations to one critical assembly and hood at a time.

Measurements were completed with two $Pu(NO_3)_4$ solutions during the month. From these and previous data, the critical concentration for a homogeneous Pu-water mixture (zero nitrate) is estimated to be ~ 63 g Pu/ℓ, for a critical mass of 816 g Pu.

Some comparisons have been made among the current experimental critical data and previous values obtained from theory and other methods, such as buckling conversions. At 12.95 liters, for the case of zero nitrate, the measured volume is 12-13% larger than previously assumed. For Pu nitrate solution at a concentration of $\sim$435 g Pu/ℓ (0.8 molar solution) the measured volume is $\sim$40% larger. As a result one may anticipate revisions in the critical data, whereby critical volumes and masses will be increased by $\sim$20-40% at higher concentrations depending on the nitrate concentration.

Experiments with Plutonium Oxide-Plastic Mixtures

The final checkout of the split table machine was completed, and this device was made ready for the experiments with PuO_2-polystyrene compacts. The initial experiments are being restricted to subcritical neutron multiplication measurements and are designed to obtain preliminary critical mass data and provide operating experience with the split table machine preparatory to its use in critical experiments.

The first subcritical neutron multiplication experiments utilizing the remote split table machine began on May 24. The fissile material is in the form of PuO_2-polystyrene compacts (small cubes up to two inches on a side). The concentration of Pu in the mixture is 1.14 g/cc, and the H/Pu atomic ratio is 15. An unreflected rectangular parallelepiped of cross sectional dimensions 12 inches by 12 inches was found to be well subcritical when four inches in height, and containing about 10.5 Kg of Pu; in further experiments the height of the stack will be increased up to a maximum of 85% of criticality. Preliminary estimates of the control and safety rod worths will be determined. Critical masses and volumes will be estimated from the subcritical multiplication measurements for various configurations of the fuel (principally rectangular parallelepipeds) under different conditions of neutron reflection.

Critical experiments will not be undertaken until final approval of the hazards summary report has been received.

GAMTEC - A Neutron Slowing Down and Thermalization Code for Lattices

The GAMTEC Code, which gives multigroup constants for heterogeneous systems for use in multigroup diffusion calculations, has been completed and debugged. The code has been modified to provide multigroup constants for homogeneous systems as well. GAMTEC also computes the individual lattice physics parameters, i.e., η; ϵ, f, p, τ, L^2, k_∞, and B_m^2.

The lattice parameters for a natural uranium, water cooled, tube-in-tube fuel assembly have been computed with the code. The quantities ϵ, f, p, and k_∞ were determined experimentally by D. E. Wood, et al., (HW-67094) for this fuel assembly in a $10\frac{1}{2}$-inch graphite lattice. The calculated results obtained this month are in good agreement with experiment.

It is planned to use this code in certain nuclear safety applications involving heterogeneous systems of slightly enriched uranium in water. Input data instructions are being prepared for distribution.

The experimental value of the minimum enrichment for criticality of a UO_3-water mixture has been determined by experiment to be 1.034 (HW-70310); this enrichment occurs at an H-to-total U ratio of about 4.8. Several cases have been run on the GAMTEC Code with the 1.034% enrichment at different H/U ratios. The k_∞'s of the homogeneous GAMTEC cases do not peak at the correct H/U ratio. The results seem to indicate that some adjustment is needed in the uranium cross sections.

Buckling of Partially Filled Spheres

In an effort to predict the analytic value for the buckling of a bare hemisphere more accurately (.6% last reported) by difference methods, the lattice of two-dimensional mesh points is being varied. Thus far, some small trends have been apparent as changes in either the r or θ spacing have been made. Furthermore, change from an equal spacing in θ to an equal spacing in cos θ have given numerical values on either side of the analytic value. The lattice with best value for B^2 has not yet been determined, however.

If the accuracy of the present approximation is insufficient (error in B^2 of about ±.01) a double Lagrange interpolation function will be tried. The flux is written as

$$\phi = \sum_i L_{r_i} L_{\theta_i} \phi_i$$

for each of the i points in the mesh. L_r and L_θ represent the Lagrange functions for r and θ coordinates, respectively. That is:

$$L_{x_i} = \frac{\prod_{j \neq i} (x - x_j)}{\prod_{j \neq i} (x_i - x_j)} .$$

In anticipation of this possibility, some of the basic programming has already been accomplished. One subroutine calculates the matrix of coefficients of the Lagrange function for each degree at each lattice point. This enables a term by term integral evaluation of the polynomial fit. Subroutines have also been written for the analytic value of

$\int \theta^m \sin\theta \, d\theta$ and $\int \sin^m \theta \, d\theta$ for arbitrary m.

Subcritical Interactions

INTERSET code reached a usable level; hence, an informal document was issued (HW-77279) for local distribution. A document and program request from Douglas Aircraft Corp. was filled.

A set of criticality data from Oak Ridge has been used to increase the number of problem types which can presently be handled. These experiments consisted of arrays of cylindrical vessels in three-dimensional geometry. The critical separation between the vessel faces was measured for cubic arrays of 2, 3, 4, and 5 on a side with no reflector. Similar measurements were taken with various external reflectors. Data for the bare systems have now been reasonably predicted theoretically. Two new methods for calculating the necessary constants were obtained. Interaction of vessels in the vertical plane (i.e., circular face to circular face) was accounted for by iterating k_{eff} of this system, from which an axial buckling was obtained. The average radial interaction between sets of these "equivalent cylinders" was found from new integration limits in SOLAN. Approximations were made for the effectiveness of the plexiglas cylinder walls and modified one-group theory was used for k_{eff} (reasonable in this H/U range). Results of the calculations are as follows:

Units/Side	F-F Separation	B_H^2	Eigenvalue Albedo	Pred. k_{eff}
2	1.43 cm	.006103	.4247	1.044
3	6.48 cm	.010696	.4860	1.012
4	10.67 cm	.01291	.5581	1.028
5	14.40 cm	.014247	.5278	.985

Constant flux in the void approximations for outside reflectors appearsto be too conservative. A new method which solves the reflector equations in terms of the cylinder fluxes, then solves the eigenvalue matrix, will be attempted on the reflected criticality data.

Instrumentation

Circuit changes have been made to improve the neutron pulser at the Critical Mass Laboratory. The pulser has been used to pulse an array of neutron film badges, a stainless steel sphere containing different levels of water, and an array of plutonium impregnated plastic blocks.

The speed limiting device on the split-half assembly was fabricated and tested. It operated satisfactorily.

The channel one instrumentation at the Critical Mass Laboratory was improved to provide an extended range and a more reliable signal. The channel one fission counter system was replaced by a Beckman electrometer with an uncompensated ion chamber as a detector.

Meeting of Industrial Nuclear Safety Group

A meeting of the Industrial Nuclear Safety Group was held in Washington, D. C., May 7-9, with the AEC as sponsor. Messrs. Brown and Clayton attended from HL, and Kiel and Stevenson from CPD.

Material for discussion included nuclear safety practices and methods of criticality control, together with reviews of recent experiments and theoretical work performed at the various sites. The recent criticality incident which occurred in the Lawrence Radiation Laboratory on March 26 also was described.

NEUTRON CROSS SECTION PROGRAM

Scattering-Law Measurements for Light-Water at Elevated Temperatures

The measurement of the slow-neutron inelastic scattering from H_2O at 95°C was terminated by the beginning of an extended reactor outage. This measurement series is now complete except for such further measurements as will be indicated from the analysis of the present results. Measurements have been made for eleven different values of the neutron energy-change on scattering which range from 0.0375 to 0.25 ev.

The computer programs LEAP and ADDELT which were obtained from AERE, Harwell, have been adapted, debugged, and tested. These programs calculate the scattering law for a Gaussian space-time correlation function for the motion of the atoms which is derived from the observed experimental data.

Rotating-Crystal Spectrometer

Development of apparatus to measure slow-neutron inelastic scattering by time-of-flight was continued. The phase stability of two test rotors operated at 12000 rpm was studied during the month. Measurements were also made of the time jitter of several neutron detectors using a delayed-coincidence technique with a Pu-Be source. The de-excitation γ-rays from C^{12} which result from the neutron-producing reaction are used as the zero-time marker for the origin of the neutron-induced pulse.

Construction continued on the 6144-channel time-of-flight analyzer. The digital functions of the system were debugged and further tested. Input and output electronics for the memory drum were developed and tested. Complete system checkouts are now in progress.

Fast-Neutron Cross Sections

The data which were obtained in April on the total cross sections for neutrons from 3 to 15 MeV for 14 samples were processed by the data-reduction program. Work continued on methods of removing systematic errors in the data-reduction program, and several methods of unfolding and smoothing results were studied. Work also continued toward the procurement of samples of the remaining natural elements.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

High Exposure PuAl Lattice Studies (2.2 w/o % Pu Containing 20.6 Atom % Pu-240)

The foil activation and mass of copper poison data from the measurements in the PCTR, on the three cell by three cell array of 19 rod clusters on 6-1/2, 8-3/8 and 10-1/2 inch pitch in graphite, is being compiled for inclusion in the Physics Research Quarterly Report. The copper and gold foil data for all three lattices has been processed using computer program APDAC. The plutonium and uranium foils used in the experiment have been internormalized using irradiations in the TTR. Thus, the Pu and U-235 foil data may now be processed by APDAC.

Plutonium Recycle Critical Facility

Startup measurements have continued during the month. Experiments to calibrate the safety rods by the rod drop method have been completed. Interaction between the rods and dependence of detector position on the measured worth of the rods was found. An absolute value for the worth of the rods will have to be determined by further analysis of the data taken. The values obtained by preliminary analysis indicate the worth of each of the three safety rods to be 25.6 mk, on the average, and the value of all three inserted at one time to be 105 mk, with an accuracy of 20%.

The time response of the safety system has been measured. All of the safety rods and control rods were found to be completely inserted in the reactor in less than 500 milliseconds after a high level scram or a short period scram.

Comparisons of the observed spatial distributions of the reaction rates of a 1/v detector and Au-197 in the PRCF with those calculated with the IBM 7090 code SWAP are in poor agreement. The calculated Au-197 reaction rate is 5% low at the reactor axis and the 1/v comparison is low, also, by approximately 20%. These calculations are based on an assumed spectrum consisting of a Maxwellian thermal component (neutron temperature equal to physical temperature) and 1/E epithermal spectrum.

The PRCF has been exposed to highly irradiated PuAl fuel elements placed in the loadout basin near the transfer lock. The neutron flux increases in the PRCF were observed as a function of fuel element position. These experiments are being analyzed and hardware for the first irradiated fuel experiments is being designed.

The initial experiment in the program will be a study of the (γ, η) reaction in PRCF due to single irradiated fuel rods that are to be placed on the axis of the PRCF. These results are to be extrapolated to experiments with complete clusters. Normalizations due to difference in exposure between the single rod experiments and complete clusters are to be based upon experiments recently completed in PRTR loadout basin.

Computations with the SWAP code have been made to estimate the reactivity changes to be expected in measurements in the PRCF of PuAl samples which have been irradiated in the MTR. Three different experimental situations have been considered in the calculations: in the first one the central cell of the reactor contains a PuAl 19-rod cluster in an aluminum thimble; in the second one the fuel is replaced by UO_2; in the third one the thimble contains only D_2O. In each case, the irradiated samples are positioned only in the region which would normally be occupied by the center rod of the 19-rod cluster. The rest of the reactor simulates the present two zone loading of the PRCF.

Reactor Kinetics Experiments

A comparison has been made of the results of the hand method of fitting flux decay data and the machine method (Glex Subroutine) in order to determine the relative accuracies of the two methods, both of which use two groups of delayed neutrons. The comparison was made using the standard deviation of results from six sets of independent readings of a given flux decay curve. The standard deviation was of the same order of magnitude for both the hand and machine method. The advantage of the machine method over the hand method is in its time saving feature.

The Glex Subroutine which will be used in analyses of kinetics experiments has been modified to accept data as it is read from a digital voltmeter. When the Glex Subroutine is used to fit flux decay data from the TTR, parameters of three out of six groups of delayed neutrons were not discernible. A program is now being developed which applies a linear least squares fitting method successively to the data. It is expected that this method of analysis will allow more groups to be determined. Also, the analysis can be done with one set of input data whereas Glex requires a series of runs with a set of data for each run.

Approach-to-Critical Measurements Using High Exposure PuAl

Approach to critical experiments in H_2O similar to the ones performed with 1.8 w/o PuAl fuel rods are planned for 2.0 w/o PuAl with 16% Pu-240 content. The remainder of the 1000 fuel rods which were ordered were delivered at the end of the month. Critical approaches with lattice spacings of 0.66, 0.75, 0.80, 0.85, 0.90, and 0.95 inches center to center are to be made. The template with a 0.66 inch spacing has been loaded with 850 fuel rods. An extrapolation from this point indicates approximately 1300 fuel rods would be required to go critical at this high fuel-to-moderator ratio.

Theoretical Scattering Laws for Water

The current program using modified Gaussians to treat the resonant structure of water in calculating thermal scattering cross sections treats only one resonance at a time exactly. In the regions between resonances, a detailed comparison of results obtained by treating the lower neighboring resonance exactly with results obtained by treating the higher neighboring resonance exactly shows large discrepancies. Consequently, it appears useful to treat at least two resonances exactly. The analysis of such a treatment is complete and coding of the analysis for the 7090 has started.

Proposed Fast Spectrum H_x Pu Experiments

Some exploratory work is under way to determine the feasibility of carrying out reactivity coefficient measurements of some H_x Pu samples in a suitable fast spectrum critical facility. These experiments should help to resolve the considerable uncertainties which exist in the high energy cross section data of the higher Pu isotopes. The required amounts of Pu needed for these experiments are being estimated via a series of perturbation calculations.

Plutonium Utilization Studies

The survey of UO_2, PuO_2, and PuN as fuels for small, compact, fast reactors has now been extended to include 30 v/o tungsten fuel elements in addition to the 0 and 50 v/o concentrations previously considered. The reactor configurations and methods of calculations are the same as reported during the previous reporting period. Additional check calculations are now underway to determine the accuracy of the methods and cross sections used in this study.

Phoenix Fuel Studies

(1) ARMF-MTR Experiments with Plutonium Fuel

Values of the sensitivity of the ARMF to fission and absorption cross sections have been used to calculate values of the fission and absorption cross sections for the PuAl samples irradiated in the MTR. The range of cross sections which were calculated was compared to the ranges used in the sensitivity measurements. As a result of the comparison it is concluded that the final set of sensitivity measurements should include a measurement with 26 mg of boron. Also, changes in the sensitivities affect the results sufficiently so that measurements of the sensitivities should be made for changes more nearly like those encountered in the measurements of the irradiated samples. The measurements of the ARMF spectrum are completed except for foil irradiations in a thermal column.

(2) Control Effectiveness of 1/v Absorbers

"Blackness Theory" is being used in the investigation of control rod effectiveness of various neutron absorbers in a series of Pu environments. The investigations to date have concentrated on 1/v absorbers of various strengths. The Pu environment is characteristic of the so-called Mark I geometry (Zr-2/H_2O ratio = 1, 70 kg Pu loading in 500 lit. core). Pu compositions have been varied from 5 a/o Pu-240 up to ~ 30 a/o Pu-240. Rod effectiveness generally decreases with increased Pu-240 content.

Group-dependent rod transmission coefficients have been obtained from numerical averages over various spectra. The thermal transmission probabilities were averaged over Wilkins spectra. The epithermal T values were obtained for 1/E spectra or from spectra derived by means of the GAM slowing down code. Even though departure from 1/E spectra is quite pronounced, the difference in rod effectiveness is not too great for these two cross section averaging schemes.

Calculations for Hf control rods are presently under way.

(3) Comparison of Pu Group Constants

The GAM and TEMPEST cross section updating reported last month has resulted in a considerable reactivity increase for one of the "Phoenix Study" reactors. The comparison was made on a light water, zirconium, M/W = 1.0, 70 kg Pu load consisting of 50.02 w/o Pu-239, 31.18 w/o Pu-240, 16.59 w/o Pu-241, and 2.21 w/o Pu-242. The rise in k_∞ was from 1.1573 to 1.2164 or 0.0591 Δk. The only cross sections changed were for Pu-241 and Pu-242; however, the spectrum change resulting from the substitution of the new data produced different group cross sections for the other isotopes as well. A table of one group burnup cross sections for the two cases follows:

	σ_a		σ_f	
	Old	New	Old	New
Pu-239	60.76	63.84	37.90	39.73
Pu-240	54.67	52.49	0.56	0.56
Pu-241	58.06	51.02	38.49	40.33
Pu-242	29.13	29.40	0.55	0.51

PRTR Burnup Experiments

Data from the analyses of physics trial test element 5042 have been received. Burnup analyses of this data are currently being done. Data are also presently being received from the analysis of physics test elements 5051, 5095, and 5092. Elements 5051, 5095, 5092 have exposures of 13.1 MWD, 40.2 MWD, and 71.1 MWD, respectively.

A thermal g factor correlation study, where $g = \frac{\bar{\phi}_{fuel}}{\phi_{mod}}$, is being done for the PRTR configuration using various codes. The purpose of the study is to establish correct flux depression factor, g, to be used in burnup calculations. Also included in the study is the calculation of epithermal monoenergetic resonance flux depression factors using either narrow or wide resonance recipes. The latter are being compared with THERMOS calculations.

Code Development

RBU

A major error in the construction of tables describing the distribution of target velocity magnitudes has been detected. Correction of the error resulted in Monte Carlo generated neutron spectra which are in excellent agreement with theory. Studies of non-absorbing hydrogen gas and non-absorbing graphite, both infinite media, produced neutron spectra having no detectable departure from the Maxwell Boltzmann distribution. In addition, the results of two spherical uranyl-nitrate solution experiments indicate that the thermalization calculation in the Monte Carlo is generating the gas scattering law properly. Using the latest uranium nuclear data in the RBU Basic Cross Section Library, the system predicted an eigenvalue of 0.9999 for the first experiment 0.9867 for the second, which contained a small amount of boron poison. The eigenvalue discrepancy in the latter calculation is thought to be due to inaccurate data.

Plans have been made to begin the theory-experiment correlation work with RBU once the modified gas model, which is to account for chemical binding effects during thermalization, is rechecked.

A program for updating the Basic Library is now complete and following adequate documentation, the library is to be published complete with support updating and editing programs.

CALX

Changes have been made in the SIGMA-3H code which allows a simpler input. This deck is currently operational. All endpoint searches for CALX seem to be working satisfactorily. A new code, KONTROL, has been written and 90 percent debugged. It provides over-all control for the CALX system by generating a new data tape at specified times during a burnup so that the CALX system can redo the spectrum calculation. Programs TEMPEST and GAM, using this new data tape with current concentrations, then provides new average microscopic cross sections after which the burnup resumes in program CALX.

HRG

Revisions in the GAM slowing down code have been in process for some time. Some of these revisions extend and modify the resonance integral calculations to permit more isotopes than U-238 and Th-232 to be included. Other revisions redefine the P-1 scattering cross sections in standard form. These revisions have now been incorporated but have not yet been evaluated. Because these changes give results differing from the unrevised GAM code, which is being used by several groups at HAPO, and because the cross section data for the revised code are being obtained from the RBU Basic Library and will be incompatible in form with the data for the unrevised code, it seems advisable to reduce confusion by renaming the revised version. The proposed name is HRG, for Hanford Revised GAM.

The last major change in BARNS-H, the code to prepare the 68-group cross section data for HRG from the RBU Basic Library, has now been coded. This change provides the base cross sections for the resonance region for those isotopes which will have resonance integral calculations done in HRG. This change has not been debugged and an elusive bug in the BARNS-H calculations in the resonance region has not been found.

ALTHAEA-MELEAGER

A substantial improvement has been made in the usefulness of the ALTHAEA-MELEAGER one-dimensional (diffusion) burnup code, in that the thermal reactions and the epithermal reactions can now be conveniently "zeroed" at the beginning of the analysis to agree with codes treating the energy spectrum in greater detail than Westcott methods allow. GAM-I and SPECTRE have been chained or linked together to provide the reference for the epithermal reactions and the thermal group may be compared with P-3 or S codes.

RBU Cross Section Updating

A document, entitled, "Updated RBU Basic Library," HW-75716, has been prepared and will be distributed upon completion of printing.

The document, "Westcott Parameters Derived from the RBU Basic Library for a Δ-4 Cutoff," HW-77389, has been completed and distributed.

Mass Spectrometry

Isotopic analyses in support of Plutonium Recycle Program studies were continued. Analyses were provided for 13 macrodrill samples from fuel element No. 5051, 4 macrodrill samples from fuel element No. 5095, and

6 burnup samples and 3 macrodrill samples from fuel element No. 5092. Informal reports were issued which contained the summary of results of isotopic analyses that are complete for individual PRTR fuel elements, No. 5042 (HW-77652), No. 5075 (HW-77653), and No. 5095 - Burnup Samples (HW-77651).

Instrumentation and Systems Studies

A number of multiplier phototubes were ordered from various manufacturers for testing in the PRTR Fuel Element Rupture Monitor system. These will be tested for stability characteristics in an effort to replace the presently used phototubes.

The mechanical design of the collimator of the PRTR Underwater Gamma Scanner for fuel rod measurements was completed and is being reviewed for approval. Scintillation detector bids were reviewed and requisitions were completed for system line filters and isolation transformers.

The modified prototype for the PRTR Liquid Effluent Monitor for gamma emitters was completed and is essentially ready for calibration. A work order was placed for fabrication of three final model instruments which are to be direct copies of the modified prototype. The three units will be used at PRTR and the prototype will be used as a reference spare.

Bid reviews were carried out regarding instrumentation for use in FRPP.

Assistance was given to PRTR personnel on the instrument requirements for a testing device which will be used to cycle test a salt-bridge resistance temperature detector in the 100-K reactor.

HIGH TEMPERATURE REACTOR LATTICE PHYSICS PROGRAM

The preliminary proposal for the HTLTR project (HW-76408), which includes the preliminary design criteria, has been approved by the AEC Richland Office. The design criteria (HW-76928) have been reexamined, and re-written wherever necessary. The criteria are at present under approval examination. Although there were many changes made, the main features of the reactor and facility, as outlined in the April Monthly Report, have not been altered.

The nature of the reactor requires that some developmental work be done during the design period. This work includes tests of the electrical heating elements, the control and safety rods, and thermocouples for use up to 1500°C. The construction of a mockup of a portion of the HTLTR graphite, 18" x 18" x 120", surrounded by thermal insulation and a gas tight containment sheath is proposed. Into this structure prototype heaters, control and safety rods and thermocouples can be inserted. The mockup would serve two main purposes; (a) to provide high temperature test and operational information about components of the HTLTR, and (b) to provide similar information about experimental equipment which is to be put into the reactor at any time during the period of its use. The proposed mockup structure is thus intended for use over a long period of time. The limitations to be placed on the size of the structure and the extent of the work are under discussion.

A descriptive formal report on the HTLTR is in preparation. It is intended for the use of the AEC in acquainting potential users of the HTLTR with the characteristics of the reactor and the experimental equipment associated with it.

NEUTRON FLUX MONITORS

The neutron spectral parameters r and T have been experimentally evaluated in the water-cooled KE test facility which is to be used for irradiating the regenerating detectors. The value of r was determined from the relative radioactivity of bare and cadmium-covered cobalt samples. To eliminate background effects and possible errors due to instrument non-linearity, the cobalt plated aluminum foils were dissolved and diluted to a similar specific activity. The spread in the five data points was greater than anticipated and limited the accuracy of the determination of r; however, the results were acceptable. The integrated neutron exposure was determined by measuring the isotopic composition of a uranium sample. The neutron temperature was based on the change in isotopic composition of a plutonium sample and was measured to an estimated 5°C, which is well within the requirements of the regenerating experiment. The results of this experiment will be reported through classified channels.

Based on the experimental values of spectral parameters r and T, a computation was made to determine the optimum initial composition of regenerating plutonium detectors for irradiation in the water-cooled test facility. Chemical Analysis, HL, was furnished with the necessary materials for mixing and encapsulating experimental detectors to these specifications. In addition, uranium samples will be encapsulated for integrated flux monitors. All of the material for the planned irradiation of the regenerating detectors has been fabricated. When encapsulated, the samples will be sealed in aluminum containers and made ready for reactor charging.

Components were received for the on-site fabrication of experimental in-core neutron detectors using Boron-11 as the sensing element. Off-site fabrication of three similar prototype units is meeting with difficulty because of the manufacturer's inability to achieve proper packing density with amorphous boron. Although this difficulty would be expected to reduce sensitivity, such chambers should be adequate for demonstrating the technique.

Work proceeded on the microwave method of measuring in-core neutron flux. Additional equipment was ordered. Four gas tubes for ion plasma tests were fabricated and several experiments conducted. In addition, several experiments were made using different interferometer configurations, and two low noise preamplifiers were designed and fabricated for detecting the small signals from the crystals. For gas tubes, 0.8 db attenuation of microwave energy has been measured at ion densities of 10^{17} ions/cm^{-3}, which corresponds to a tube current of 12 milliamperes.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

The ability of the multiparameter eddy current nondestructive testing device to separate four test specimen parameters was demonstrated. The performance of an eddy current tubing tester was improved by making modifications in the test coil a-c bridge circuit. The design of a polar type graphical nulling unit has been finalized and working drawings for its fabrication have been prepared.

Evaluation tests showed that the multiparameter eddy current tester was able to separate four parameters. The test involved a three-layered sample composed of stainless steel on brass with steel below the brass. The four parameters involved were probe spacing, stainless thickness, brass thickness, and steel thickness. Separation of parameters was good over the following ranges:

Probe spacing	0.0035 to 0.0115 inch
Stainless thickness	0.001 to 0.006 inch
Brass thickness	0.004 to 0.006 inch
Steel thickness	0.002 to 0.006 inch

In general, the work and time involved in setting up a four parameter system is nearly an order of magnitude greater than that involved in setting up a three parameter system.

The possibility of obtaining three parameter separation with a single test frequency is being evaluated. For this purpose two additional 22 kc plug-in amplifiers were constructed and tested. With these units in the tester a simple test was run with only the three 22 kc channels in use.

1103521

Preliminary tests indicate there is a possibility of separating three parameters. However, since this contradicts some of the basic theory, more evaluation is required. It is known that under some conditions many parameters, occurring one at a time, can be identified using a single frequency system. However, this type of test is limited in its ability to separate parameters occurring simultaneously. There are some subtle relationships in this area involving modulation and information theory which are being studied.

During the evaluation of a single frequency eddy current test for inspection of Zircaloy tubing, it was observed that electrical "noise" signals occurred with certain samples which could not be discriminated against using the common phase discrimination technique. This difficulty was traced to a differential sensing coil unbalance, even though the a-c bridge in which the coils were connected was adjusted to give an over-all bridge balance. It was found that by rearranging the bridge circuit and including an additional bridge balance adjustment performance of the tester was improved. With these changes a No. 80 drill hole through the .035 inch wall of a 0.565 diameter Zircaloy-2 tube gives a signal well above the noise level.

A design of a polar type graphical nulling device for use in eddy current testers has been established and working drawings were prepared for construction of the device. It is planned to use devices of this kind for a-c balance adjustments in a multiparameter eddy current tester where several balancing circuits are required.

Heat Transfer Testing

Constant, uniform heat input to the sample is desirable during heat transfer tests. Tests made to determine causes of variations in heat input from the experimental plasma arc heat source show that changes in the heat input to the sample are accompanied by variations in current supplied to the plasma generating head, while the gas flow, gas pressure, and voltage remain constant.

Connection of telephone lines between the 314 Building heat transfer testing instrument and the 3707-C Building analog computer facility is complete. These lines will allow the tape delay unit in the computer to be used during continuous computation required for emissivity compensation. Tests, made to determine noise pickup and attenuation in the lines, indicated that 2 kc signals could be transmitted through them at a level of one volt with a signal-to-noise ratio of at least 100. No appreciable attenuation occurred.

Zircaloy-2 Hydride Detection

Eddy current measurements on butt-welded N Reactor process tube samples showed that electrical resistivity decreased in the weld zone by about the same amount that 2000 ppm hydride would increase the resistivity. N Reactor process tubes are composed of 10 to 20 foot long sections but welded together.

A temperature compensated eddy current probe is being developed. Several possible concepts are being explored. Silver plated Nilvar wire, which has a low thermal coefficient of expansion, is presently being evaluated for this application. Probe coils made from this wire are expected to undergo a smaller change in inductance than copper coils during equivalent temperature changes.

Two frequencies are being considered for the eddy current hydride test. 455 kc has been applied to date, and has been found satisfactory for detecting defective surface areas. However, tests have shown that frequencies as low as 40 kc can be used on 0.25 inch thick Zircaloy to obtain deeper eddy current penetration, without the test being excessively sensitive to changes in tube wall thickness.

Scanning requirements of in-reactor inspection of process tubes may require a slip-ring connection between the probe coil and lead-in wires. Measurements in the laboratory indicate that a slip ring, contributing .001 ohm variations to the series resistance of a 100 micro-henry, 19 ohm probe coil, could be used at 450 kc without seriously degrading the signal.

Incipient Failure Detection

Microcracks in metals can propagate slowly under conditions of creep or reversing loads. When they reach a critical size, however, they become unstable and propagate rapidly to cause failure of a metal component at a much lower stress than would be required to produce failure by plastic deformation alone.

Formation of microcracks can initiate at sites of massive dislocation pile-ups. Dislocation arriving at such a pile-up region are tenaciously held there by interaction with other dislocations until a fissure is formed. Microcracks can also be initiated due to impurities and fabrication flaws which act as stress risers.

Techniques capable of providing an indication of impending failure during the initial stages of dislocation pile-up or microcrack formation would be of value in preventing in-service failure. A research program oriented toward this objective has been organized. Initial plans have been made to monitor with eddy current and ultrasonic sensors carbon steel samples as they are mechanically stressed to fatigue.

Fundamental Ultrasonic Studies

Experimental measurements on the kerosene-gylcerine boundary are partially complete. Wave amplitude distributions in and normal to the propagation directions were measured. In addition, reflection coefficients and phase velocities were measured over a range of incident angles. All experimental data were in close agreement with the predicted values and boundary waves were observed at incident angles exceeding the critical angle. Boundary wave phase velocities were shown as predicted to decrease as the incident angle was increased from the critical angle to ninety degrees. An energy exchange phenomenon, not explained by the original analysis, was observed during measurements of boundary wave phase velocities.

The experiments were conducted at an ultrasound frequency of 10 Mc using the beam from a one inch diameter lithium sulphate transducer incident on a kerosene-glycerine boundary, kerosene being the incident liquid. The analytic model assumed that the incident waves would be plane and infinite in both time and space. To experimentally approximate infinite time, narrow frequency-band pulses of about seven microseconds duration were used. The near field of the one inch diameter transducer was used to approximate plane waves infinite in space. The degree with which these approximations were valid was indicated by good agreement between predicted and measured results. The exponential amplitude distributions in, and normal to, the propagation directions of refracted waves were measured at several refraction angles which demonstrated that inhomogeneous waves were generated as a result of attenuation.

The boundary waves formed at the kerosene-glycerine interface were detected at several angles above the critical angle and their measured phase velocities were in excellent agreement with predicted values. The phase velocity change from a value equal to the compressional velocity in glycerine, to a value equal to the velocity in kerosene for incident angles exceeding the critical angle, was clearly demonstrated. An interesting behavior was noted during boundary wave phase velocity measurements. As the wave sampling probe was traversed parallel to the interface the amplitude of the boundary wave was observed to pass through periodic maxima and minima. This effect was not predicted in the original analysis and is believed to be associated with the exchange of energy across the interface. The exchange mechanism is being studied.

Theoretical work continued on Lamb waves in a hollow cyclinder, with a view to finding a satisfactory calculation procedure for the frequency equation. As mentioned last month, the Bessel functions involved are of higher order than those for which published tables exist. Several methods of approximation were tried, and one was found that gives an answer that is apparently usable, although only numerical calculations will show how accurate it is. To obtain this it was necessary to expand the product of two Bessel functions in a power series in the reciprocal of the order and evaluate three terms (when substituted in the frequency equation, the first two terms cancelled). To determine the accuracy, at least one more term must be found.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Special emphasis is being given the development of optimum transducer arrangements for purposes of inspecting thin-walled tubing. Most favorable results have been obtained to date with spherically focused lithium sulphate transducers measuring 3/16 inch in diameter. Plans have been made to evaluate these transducers more fully in a complete prototype test station now under fabrication.

It is now clearly established that test data suitable for comparison between separate test stations cannot be obtained unless tranducers of similar operating characteristics are employed. A means of assuring uniform transducer performance is therefore essential. A method under development which appears to provide this assurance measures the radiated beam field strength by monitoring the ultrasonic pulse amplitude reflected from a 30 mil diameter ball as the ball is moved through the center of the field in two perpendicular directions. A third measurement determines the center frequency, band-width and damping factor (Q) of the crystal. The experimental arrangement employed establishes these parameters on an oscilloscope tracing which may be photographed to provide a permanent record of the transducer characteristics. These procedures have been communicated to a prominent transducer vendor who has agreed to apply these tests prior to shipping transducers to HAPO.

Although the 3/16 inch diameter crystal detects 1/20 inch wall thickness defects very well, it presents other problems in that signals from the tube surface comparable in amplitude to bonafide defect signals are also present intermittently. Since the tube wall is thin, the defect and surface signals appear very close together in time. If the electronic defect gate is positioned with respect to the transmit pulse, small changes in crystal to tube spacing would occasionally place the gate over these surface signals and would, consequently, be interpreted as defect signals. If a constant amplitude surface signal were available, however, one gate could be used to position a second gate precisely behind the surface signal, thus obtaining a gate which would follow the tube surface and accurately select out only the defect signals. This constant surface signal could be obtained in two ways: (1) by high gain amplifiers to bring the varying surface echo up to a usable minimum level, or (2) by redirecting a portion of the ultrasonic beam to strike the tube surface; thereby gaining a high amplitude constant surface signal.

The high gain amplifier would require more complex circuitry in the tester, while redirecting the beam would offer no additional circuit complication. Consequently, a novel method of redirecting the ultrasonic beam has been developed. An outer lens was machined into a 1/4 inch diameter focused crystal leaving the 3/16 inch diameter inner portion focused and the outer lens flat. This produced a bull's-eye type of ultrasonic beam with an inner focused inspection beam and an outer ring shaped surface echo beam. The surface echo remains constant in amplitude and precedes the inspection beam information. The inspection beam is focused to 9 mils diameter at its focal point. Notch defects down to 1/20 inch of the wall thickness are still easily detected as with the 3/16 inch diameter crystal.

A prototype tester which will utilize the constant surface signal from the bull's-eye beam is being designed. The basic timing and pulser circuits have been fabricated and all additional circuit components have been received on site. This instrument will be versatile with both red light and analog recorder outputs. A prototype mechanical tube scanning system is also being set up. Both good and reject tubing will then be tested. The test information will be recorded in analog form for easy reference in evaluating equipment performance.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Analysis of atmospheric dispersion data collected over the past three years continued. Combining the earlier results,which permitted prediction of the lateral growth of the plume, using the product of the standard deviation of the wind azimuth variations and the mean wind speed, with the Richardson Number, peak exposures were calculated as a function of distance from the source. As a check on the accuracy of predictions, ten Hanford experiments which were not adequately contained within the sampling grid for use in development of the prediction model, were used as an independent sample to test the model. Using a "miss-factor" defined by the greater of predicted or observed exposure divided by the lesser (a number always greater than 1.0), a range of values of 1.0 to 4.0 was found with an average less than 2.0 over the distance range of 200 meters to 3200 meters. A similar calculation for the plume width yields a miss-factor rarely exceeding 1.25.

Development of an atmospheric dispersion prediction scheme which permits more realistic accounting for the period of release of contaminants by superposition of incremental releases will permit improved appraisals of the consequences of reactor accidents. Relationships between Hanford ground source diffusion data have been evolved, making it possible to estimate cloud growth and exposure solely from wind and temperature data during the release period. By further considering the duration of release to be a series of incremental releases, the Gaussian crosswind distribution assumption can be employed to yield a resultant distribution that compares well with those observed in experiments. Independent verification of the method has been obtained for 10 experiments where the release time was as long as 3-1/4 hours and the resultant distributions were multi-modal.

Five field tests were conducted during the month. Two of these were run during unstable meteorological conditions with the source height at 61 meters to simulate a stack discharge. Three tests were conducted using a 6.7-meter stack height during stable conditions with sampling on both the horizontal and vertical grids to a distance of 3200 meters.

In our precipitation scavenging work, comparison of the sizes of zinc sulfide particles found in raindrop autographs to the sizes found on air filters for three experiments showed consistently that large particles are scavenged most efficiently. Corrections for anisokinetic errors in sampling must still be applied, however.

Dosimetry

Samples have been received from all three of the Public Health Service hospitals in Alaska that are cooperating in our survey-by-urinalysis program. Data are on hand for 19 subjects from Fort Yukon, 28 from Tanana, and 37 from Bethel. The average body burden indicated for Fort Yukon is about 120 nc of Cs-137. At Tanana and Bethel the bulk of the people fall in a range with an average of 25 to 50 nc. At Bethel, however, five subjects were up in the 200-300 nc range. Two of these were from Nunivak Island and list reindeer and fish as their principal foods. In general, those from Fort Yukon list moose as their principal food; those from Tanana and Bethel list moose or fish or both.

Preparation continued for this summer's whole body counting study in Alaska.

A measurement of the sensitivity of the shadow shield whole body counter for Cs-137 was made with our new plastic phantom. Contrary to the results for potassium where good agreement was found, the result for Cs-137 was 20% different from that found in our other work. We feel that this is the fault of the phantom in not giving a good simulation of the human body.

Study of the detection of beta rays emitted from the surface of the body continued. A new counter was calibrated on our cooperating P-32 patient following his most recent injection. Work to lower the background of the counter continues.

Assistance was given RPO in preparation of their truck-mounted shadow shield whole body counter. The background and the counting efficiency of the 11-1/2 x 4 inch scintillator in the shadow shield turned out to be about as expected, namely the sensitivity for whole body counting is about the same as that of a 9-3/8 x 4 inch crystal in the iron room. Resolution of the large crystal was found to be poor at first operation. This was traced to an inoperative photomultiplier tube. When this was repaired the resolution was found to be as good as that of the smaller crystals. RPO calibrated their counter for potassium and a partial check was made of our shadow shield; it agreed with our previous results. We calibrated RPO's shadow shield and a new two-crystal assembly for the regular whole body counter and our shadow shield for cesium; for our counter the results were within a few percent of the value that we have been using.

More supplies were received for assembly of a large scintillation counter for X-rays.

The positive ion Van de Graaff operated satisfactorily during the month except that it was necessary to limit beam energies to 1.8 Mev or less to prevent breakdown.

The possibility that a temperature effect had caused the shift in calibration of the precision long counter that is being used for inter-laboratory calibration was investigated. In a temperature range of ten degrees a very slight temperature coefficient was observed. It was not enough to explain the shift.

Measurements were completed of the energy sensitivity to neutrons and to photons of our large tissue-equivalent ionization chambers.

Instrumentation

Each of the three individual sensors which together comprise a composite test probe for the detection of neutrons with energies ranging from thermal to fast have been used to measure flux levels in laboratory experiments. Some difficulties were encountered in obtaining a uniform, adherent coating on one detector. Difficulties in coating U-238 in place of Am-241 on the fission detector led to the use of Np-237 plated as a sulfide. Plans were made to test this detector with the positive ion Van de Graaff.

Two experimental rechargeable pocket gamma dose meters were fabricated with the sensor located in a plastic half cylinder which was positioned on one face of the dose meter box. A newly printed circuit board layout was also prepared. Proper molding of a 0.125 inch diameter conducting Teflon rod for use in the sensors was achieved. The molded rod provided 1.2×10^6 recycle operations before sticking occurred between the recharging fiber and the rod. Tests were carried out by Radiation Protection, HL, on one of the two new prototypes. Most tests seemed to be acceptable with the worst case angular dependence being about a twenty percent drop in sensitivity, which is considered to be well within acceptable limits. For reasons not yet understood, however, the unit proved to be sensitive to mechanical shock. This problem was not apparent in earlier model instruments. It is suspected that center rod flexure is causing the difficulty.

Circuit development work was carried out on the HAPO Radiotelemetry System with emphasis on providing a new line-operated data station. The new regulated power supply was completed. In addition, a voltage controlled oscillator, which incorporates a novel count rate circuit, was developed and tested. For an input voltage of 0-5 VDC, the output frequency extends from 300 cps to 3.3 kcs. The input current required is 0.2 microamperes. The differential linearity $\left(\frac{\Delta f}{\Delta v}\right)$ is low by 1% at 300 cps and is ± 2.5% from -25°F to +140°F as referenced to 75°F. The discriminator characteristics of the commercial receiver incorporated in the data station were investigated and found to be unacceptable. New audio amplifiers were then designed to drive the handset and Vibrasponders with the Vibrasponder amplifiers having frequency response roll-off above 300 cps to reduce false call-ins due to noise. A new squelch amplifier was designed, and a new time delay circuit was designed to eliminate the high current required for the original thermal time delay unit. Considerable chassis rewiring was also completed.

All test work was completed on the 100 r high level emergency type pocket indicating and signaling dose meters. A rough draft formal report was prepared.

An improved method of obtaining accurate and reliable differential alarm trip signals from two solid state count rate meters was developed and tested. The circuit modifications were installed in the experimental beta-gamma hand and shoe counter which employs gamma background compensation. Tests proved to be satisfactory.

Test chassis fabrication was nearly completed for an experimental logarithmic response area radiation monitor which uses a scintillation detector, vibrating capacitor, and solid state circuitry.

The new light trap design for the Atmospheric Physics experimental, in-field, real-time phosphorescent particle detection instruments was tested in comparison with the original type. Tests with varying airborne concentrations of zinc sulfide showed the advantage, in increased sensitivity, of the new design with a 15 CFM air flow versus the 4 CFM of the original. The original prototype was used in a novel experiment to demonstrate how radioactive particles can be dispersed during a fire. Zinc sulfide was placed on zirconium turnings enclosed in a room and the zirconium was ignited. The zinc sulfide carried aloft was detected. Fire extinguishing foam was then applied to stop the fire, which also prevented further release of zinc sulfide. The fire did not appear to affect the phosphorescence properties of the zinc sulfide.

Reliability and general performance tests were carried out on the experimental radioactive liquid sample changer for use with large scintillation well crystals. A number of mechanical modifications were made.

Wind speed and direction are being recorded from eight levels on the 400 foot Meteorology Tower in the 200-W Area. From these data, one hour averages are estimated for maintaining climatological records. These hourly estimates are punched on IBM cards for later programming on the IBM 7090. It is desired that these data be more accurately processed in a form which can be programmed on the IBM 7090. A secondary requirement is to obtain data for calculating the diffusive capacity of the atmosphere. In this case the data required are five to ten second rms deviations of the rectilinear components of wind velocity from a five to twenty minute mean velocity vector.

The results of an analog solution for atmospheric diffusion have shown that the accuracy with the analog method is not within acceptable limits. The inaccuracy results from the method for determining the deviations of the mean, which incorporates an exponentially weighted time averaging device. To obtain the correct results a delay of one-half the long period averaging time (five to twenty minutes) would have to be incorporated in the system. Two delay units per power level would be required. Due to the number of delays, and cost per delay unit for the long delays, the analog system does not appear feasible.

The required data can be acquired by using a digital computer but the cost of a complete digital system with sufficient memory makes this system unfavorable. The use of a data phone with phone lines directly to the IBM 7090 was considered. Due to the memory required for a fast sampling rate, this method proved to be inadequate. The use of analog averaging circuits with a digital readout was also considered. This method seems to be the most practical way for obtaining the required data. Wind speed and direction from the eight levels of the tower could be punched out on paper tape. A paper tape punch with a punching rate of 150 punches per second will result in a sampling rate of about 0.75 samples per second input for the 16 inputs. The averaging circuits would consist of a simple RC filter with a time constant of 15 minutes. This time constant would result in approximately the same statistics as a 30 minute average sample period would. The combined analog-digital system then would give the required data for the climatological records. In addition, the system could be used to record instantaneous values of wind speed and direction; then, at a later time, these data could be programmed on the IBM 7090 to calculate the diffusive capacity of the atmosphere.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples as received during the month. The mass spectrometer for this program was out of service for four days due to instrument component failure.

Considerable progress was made on the design of a vacuum-lock sample changer intended to increase the sample load capacity of the mass spectrometer. It is planned to construct the sample changer with a sliding shaft which introduces the sample into the spectrometer through a manifold with three stages of differential vacuum pumping. Both the shaft and manifold are to be made of high-strength alumina. Final design drawings are being prepared.

Construction continued on a scintillation-type prototype ion-detector for the mass spectrometer.

TEST REACTOR OPERATION

Operation of the PCTR continued routinely during the month except for a scheduled three-day maintenance outage. There was one unscheduled shutdown caused by faulty bypassing technique. The k lattice experiments - k_∞, KVNS fuel wet and dry, spline worth measurements were partially completed. The gas door seal was replaced during the maintenance outage. The reactor room will be pressure tested to determine if a satisfactory seal has been obtained.

Operation of the TTR was on an intermittent basis during the month. Several series of irradiations for foil normalizations were made. The reactor was operated for the University of Washington Graduate Center one day during the month. The gas door seal was replaced.

The last 150 Pu-Al 16% 240 rods for the critical approach experiments have been received, and the approach tank equipment is being reactivated.

CUSTOMER WORK

Weather Forecasting and Meteorological Services

Further consultation service was rendered on meteorological and climatological aspects of 1) oxides of nitrogen release in 300 Area to IHO, 2) exhaust fume release from the Bioassay Building to CET, and 3) stack requirements for the Low Level Radiochemistry Laboratory to CR&D. Off-site requests included 1) selected monthly and annual climatological summaries for the U. S. Weather Bureau and 2) psychrometric data for E. Bollay Associates, Inc., Santa Barbara, California.

The mid-May forecast of the Columbia River at Hanford during the remainder of the water year, 1963, was prepared and distributed.

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

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	93	84.9
24-Hour General	62	91.0
Special	162	95.7

The wet spell, which brought a record amount of precipitation during April, ended abruptly after the first week of May. Temperatures were below normal during the first half of May, but generally above thereafter. The over-all monthly average was slightly below normal.

Instrumentation and Systems Studies

The Boonshaft and Fuch's transfer function analyzer was used to obtain frequency response data on a uranium swelling capsule in an effort to determine the transfer function of the closed loop temperature control system. Preliminary data indicates the capsule transfer function is of the form:

$$\frac{\text{Temp}}{\text{Power}} = \frac{K}{(1+T_1 s)(1+T_2 s)}$$

where T_1 = 120 sec.; T_2 as yet undetermined, but near 2-3 sec.; and K = gain; non-linear. Although the capsule transfer function itself is straightforward, control problems are encountered because of the many variable parameters affecting total power input such as gamma heating, inlet water temperature, and water flow. In addition, although the static control signal-to-heater voltage signal is linear, the voltage-to-power relationship is a second degree function and thus non-linear. These effects will all be determined along with the basic capsule transfer function and the system will be simulated on an analog computer complete with three-mode controllers.

Similar tests were performed on a creep capsule and preliminary data indicates a transfer function of the same form as for the swelling capsule. However, in addition to the same parameters affecting control as for the swelling capsule, the creep capsule transfer function is very dependent on the helium pressure within the capsule. Also, the creep capsule has three heaters, separately controlled, and thus temperature interactions. The primary time constant in the creep capsule transfer function is about 360 seconds, but varies with helium pressure. The high frequency breakpoint has not been determined. The creep capsule temperature control system will also be simulated.

Advice was rendered Physical Metallurgy concerning a temperature control system for an in-reactor test capsule now under development. The capsule will contain five metal specimens each to be simultaneously irradiated at a different temperature.

Reactor Metals Research was provided advice concerning a system to measure the dynamic stress-strain characteristics of various reactor metals. The requirements of the system are to display and record the stress and strain of a specimen as a function of time as the specimen is tested to destruction. The experiment is of a one-shot nature and takes place within a few milliseconds.

Advice was rendered Physical Metallurgy concerning a cryogenic temperature control system. Control temperatures would be in the range 20°K to 200°K. Methods and hardware were recommended.

A method of controlling an in-reactor specimen at a constant strain rate and reading out stress was proposed to Reactor Metals Research. The strain rate was to be easily changed and have good stability and linearity.

A temperature control system designed to control a high vacuum-high temperature (3000°C) oven was recommended to Reactor Metals Research.

Information on the performance of a specific differential amplifier was given Chemical Research. The amplifier was one which has been evaluated for possible use in the creep program. Information on the same amplifier was rendered Chemical Development, HL. The amplifier did not meet the requirements of the specific application, so another amplifier was recommended and subsequently purchased.

Specifications for a precision solid-state control and data acquisition system were written for a new test to be performed by Reactor Metals Research in the ETR.

Specifications for a precision solid-state over temperature protection system were written for Physical Metallurgy. The system will be installed at the 100-KW in-reactor test facility.

A paper tape reading and programming system was suggested to Reactor Metals Research for use on the creep program data logging system.

Construction of the creep program 96 point data logger is continuing. Debugging is also in progress. Target date for completion of the system is mid-August.

The remote demand-log system designed and built for use with the swelling program data logger was installed at 100-KW.

Work on the 333 Building autoclave temperature control problem has stopped pending the solution to certain mechanical problems on the autoclaves. Work will be resumed when the autoclaves are working properly.

A firm understanding of the theory and derivation of the C-column mathematics model was obtained. The purpose of the associated analog simulation will be to optimize the interphasial mass transfer constant and the back diffusion coefficients with organic and water phases in the column. Two models will be used for describing the C-column. One will describe the stripping action below the feedplate, and the other one will describe the extracting action over the remainder of the column.

Basic design was established for a special soil moisture content measuring probe for Advanced Technical Planning, CPD, and Chemical Effluents Technology, HL. A neutron proportional counter and a solid state preamplifier will be used to work directly with available commercial instrumentation. The necessary Pu-Be neutron source was ordered.

Fabrication was completed on the conversion of an alpha-beta-gamma air filter sample counter to the coincidence count method. The instrument is in use at the 231 Building and the work was done for Radiation Protection Operation, HL.

The coincidence counter for air filter samples was completed and tested for use in the 327 Building and is ready for installation. The counter was assembled for Radiation Protection Operation, HL.

Necessary instrumentation for an air sample filter coincidence counter for use at the Fuels Recycle Pilot Plant has arrived and the required interconnecting coincidence circuitry is being fabricated. The detector heads have been ordered.

A design cost estimate was prepared for the acquisition of an air sample filter coincidence counter for use by CPD personnel at the 234-5 Building and a similar estimate was made for CPD for an instrument to be used at Purex.

Specifications were prepared for detectors and instruments for use in a plutonium waste container measurement system for Plutonium Process Engineering, CPD.

Work has continued on the Eddy Current Motion Analyzer (ECMA-I) which is being developed for use in studying the fuel assembly vibrations that have been encountered in the PRTR. Fabrication of six channels of amplifying and detecting circuitry has been completed and these chassis are presently being checked out and installed in the equipment cabinet. Fabrication of the transfer panel has begun and is expected to be ready for installation and final checkout early in June. Prototypical portions of the specially designed fuel element assembly are being fabricated and tested to insure component compatibility and reliability. Assembly of the sensor equipped fuel element cluster is expected to begin in June.

Optics

The readout system of the optical process tube traverse mechanism has been replaced with a Linear Variable Differential Transducer (LVDT) readout to reduce the total inspection time and provide readout data in a form suitable for digital or analog recording. Evaluation tests were conducted on a transistorized excitation and demodulation system during the month. Results of these, and other tests, indicate that a system of maximum utility and accuracy can be obtained by utilizing these excitation demodulation modules in conjunction with a self ranging digital voltmeter with a built-in binary coded decimal (BCD) output for automatic recording on a paper punch tape. The information thus obtained would be available for immediate processing through the IBM 7090 system. A memo proposing such a system is being prepared.

The April monthly report discussed the least squares analysis of eight calibration runs of the Electrical Readout Traverse Mechanism in a full length reactor process tube at 189-B Building. These fits were good to less than 0.080 inch and the over-all accuracy of the device using averaged values for the calibration constants use ± 3/8 inch. Six tests run later with tube bendings of about five inches yielded least squares fits good to about 0.080 inch. In all, 18 test runs were made. A modified computing program was effective in reducing error in calculated displacements. An average absolute error of 1/4 inch was obtained for these 18 runs.

All the 18 runs were made using an electromicrometer as a readout device. Shop tests were also conducted using a d-c translator and millivoltmeter which offers the advantage of no electrical deadband and more rapid response. The chief disadvantages are that more electrical equipment is required and the response is not as linear as with the electromicrometer. More test runs are planned for the next few weeks.

Tests using CdS photoconductors in combination with a projection comparator to read out fuel element rail heights indicated that positioning accuracies of 0.0001 inch are feasible. A servo positioner is being assembled to maintain the reference location while a second photocell is used to read out rail height on a Honeywell recorder. The object of this project is to demonstrate the feasibility of production line gauging of rail heights at the rate of one fuel element per six seconds with an accuracy of ± 0.0002 inch.

Two laser rods, one a calcium tungstate rod doped with neodymium ($CaWO_4;Nd^{3+}$) and the other a calcium fluoride rod doped with uranium ($CaF_2;U^{3+}$) have been received. Attempts will be made to induce lasing in these rods using gamma scintillation as a light pumping source. Before this is attempted, however, a control series of tests will be conducted with a conventional Xenon flash tube as the light source.

While the equipment for these tests is being fabricated, some pilot experiments are under way using photo flash lamps. Calculation of the Boltzmann factor for Nd^{3+} at room temperature leads to a terminal state population of .0000698 relative to the ground state, so it is expected that four-level action can be accomplished at a low threshold.

Investigations are also under way to find the best available scintillator and/or phosphor with appreciable spectral output in the respective pump bands. Since phosphors are usually opaque to their own emission, their effectiveness as pump sources is questionable. Equipment is being fabricated for testing materials in a 10^6 r/hr gamma field for spectral emission characteristics and for relative conversion efficiency.

During the five-week period (April 14-May 19) included in this report, a total of 600 manhours shop work was performed. The work included:

1. Repair of one crane periscope head each for Redox and Purex.
2. Change-out of all lenses in two army M-2 borescopes for Irradiation Testing, IPD.
3. Fabrication of one set of eight glass bearings for CPD.
4. Repair of three camera shutters and modification of four shutter release cords.
5. Fabrication of a borescope camera adapter to permit use of motion picture camera in photography of reactor graphite channels.
6. Assembly and adjustment of one metallograph.
7. Silvering of ruby laser rod and reflector.
8. Silvering $BaTiO_3$ crystals for Radiological Physics.
9. Aluminizing 10 mirrors for optical comparators.

Physical Testing

Testing service increased at an accelerated pace during the month, necessitating an additional shift crew at the tubular testing facility to meet a deadline requirement. The number of tests performed during the month increased to nearly 30,000; one-half that number being on separate items, an increase of nearly 25 percent over the previous month. Thirty-nine different organizational components purchased testing services, seven more than before.

A deadline of May 6 and a goal of 1200 inspected and tested tubes for retubing the K Area reactors were met by the tube testing operation. To meet this goal over 1600 tubes were inspected and tested and an additional 150 tubes were made available for the reactor retubing in excess of the required 1200. Approximately 40 tubes of the remaining 400 were returned to the vendor for replacement, and the rest are either being held for the results of corrosion tests or for different locations in the reactor. A dry honing process developed by Physical Testing was utilized by IPD to virtually eliminate the Vanstone cracking problem. This extra processing step in the inspection process created the necessity of adding an additional shift and extra personnel to meet the scheduled delivery of tubes to the reactors.

Approximately seventy Zircaloy-2 sheath tubes, 0.442 inch in diameter with a twenty mil wall, were tested during the development of ultrasonic inspection equipment for in-plant testing of similar tubing. Fifteen of the tubes tested indicated defects greater than the limits specified of 1/20 the wall thickness. In each instance the tubes were either visually or metallographically examined and all but one of the indications were verified as defects ranging from 0.001 to 0.008 inches in depth. The same type of commercial equipment will now be modified at Harvey Aluminum Company in California for the ultrasonic inspection of over 2000 tubes being purchased for fuel element manufacture. Training will be offered to all available offsite General Electric inspectors for ultrasonic testing to Hanford standards while these tests are in progress at Harvey.

A list of nine possible methods was presented to the Fuels Section of N Reactor Department to nondestructively test the welds on some 2000 suspected N Reactor fuel elements. The specific welds in question were produced from a spot welder that may have been out of control and produced such a large weld nugget that uranium was alloyed with the Zircaloy in the weld nugget. Of the methods suggested, at least three will be evaluated on purposely defected specimens.

Other tests were performed on the high pressure C-1 loop to insure its integrity as well as measurements for the bond quality of the plutonium-aluminum fuel elements scheduled for the loop. A series of tests are in progress to determine the cause of failure of the Hansen brass fittings installed on the front face of F Reactor. A routine survey of pressure vessel thicknesses was conducted for F Area as well as ultrasonic thickness measurements on 190-F Building process piping. A series of acceleration and vibration measurements have been made at selected points on the PRTR during various phases of startup and reactor operations. Radiographic inspections of welds for code compliance and pressure vessel certification were conducted on enough footage to represent nearly ten miles of weld, with the results in each case reported before the next shift.

ANALOG COMPUTER FACILITY OPERATION

The analog computer problems considered during the month include:

1. D Reactor Control Problem.
2. NPR Kinetics.
3. NPR Pressurizer.
4. NPR Injection Pump System.

The computer utilization was as follows:

EASE (Days & Swing)	GEDA (Days Only)	
300	144	Hours Up
22	22	Hours Scheduled Downtime
30	10	Hours Unscheduled Downtime
0	0	Hours Idle
352	176	Total

Considerable difficulty with the EASE computer is presently being experienced. These difficulties include relay contact failures, trouble with the readout units (DVM, panel meter, printer) and intermittent troubles with the servo-set pot system. Some of these troubles may be due to a relaxation in the routine maintenance program. This relaxation occurred as a result of a lack of trained maintenance personnel due to temporary transfers and vacations.

Purchase of the new analog computer is being delayed pending completion of negotiations by Company and Commission purchasing groups. At the Commission's suggestion, discussions were held with two bidders. A purchase contract is expected to be placed during June.

Preliminary arrangements were made for setting up an analog computer applications course. The course is tentatively set up to accommodate 15 members and will constitute ten two-hour sessions. One-half of the time will be spent on basic analog theory and the remainder of the time will be spent discussing and working simple applications problems.

INSTRUMENT EVALUATION

Temperature tests on the experimental solid state portable rechargeable survey meter showed calibration errors of about 10% from -10°F to +125°F.

All evaluation tests have now been completed and only the instrument drawings remain to be done.

Calibration testing and minor circuit modification were carried out on the experimental beta-gamma hand and shoe counter which uses gamma background compensation circuitry. Most testing is now complete and a final background probe is being fabricated for installation. Following minor retesting, the instrument will be moved to the reactor areas for field tests.

Acceptance testing was started on 30 scintillation solid state portable alpha "poppies" which were fabricated offsite.

RS Paul

Manager
PHYSICS AND INSTRUMENTS LABORATORY

RS Paul:mcs

CHEMICAL LABORATORY

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Reactor Effluent Radioisotope Studies

During a week of this report period the water treatment pilot plant produced water with a turbidity of 0.001 by close control of the alum addition, pH, and zeta potential. During this week reductions in many of the radionuclides in reactor effluent water by as much as a factor of two were noticed. This test confirms the importance of zeta potential control. Only 10-12 ppm alum were required when water was being produced at a pH of 6.6.

During the present reactor outage a pump will be installed in the raw water line to provide positive control of the feed water and to allow capacity operation during reactor outages. The alum supply system will be altered to provide a means of more accurately measuring feed rates. At reactor startup water furnished by this plant will be at a pH of 7, using aluminum nitrate as the flocculating agent and nitric acid to control the pH. This test is designed to determine the reduction in P-32 obtainable when the sulfur parent material is limited to the extent practically feasible.

Deionized Water Studies

For a period of three days prior to a shutdown, process water was used to cool two reactor tubes whose previous coolant had been deionized water. Both tubes had fuel charges clad with zirconium but one tube was zirconium and one aluminum. Although three days is not sufficient time for a steady state to be attained, some information of interest was obtained. The Na-24, P-32 and Cu-64 concentrations in the effluent increased sharply while the Mn-56 and As-76 concentrations increased much more slowly. The Ni-65 and Ga-72 concentrations remained essentially constant. These latter two radionuclides are related to the corrosion rates of the tubes and jackets. Since the change in their concentrations observed when process water was added was very slight, this indicates that the contributions from parent materials in process water are small. Ni-65 and Ga-72 are thus true indicators of corrosion not complicated by interferences from water salt activation products.

Film Adsorption Studies

Aluminum coupons coated with various agents which had previously shown promise in laboratory tests were placed in reactor tubes downstream from the flux zone. After 30 days' exposure the amounts of adsorbed radionuclides were measured. Preliminary data show that a Flo-Master Black Ink coating reduced adsorption of P-32 and As-76 by factors of 2-4. If similar reduction can be obtained with parent materials on fuel element surfaces in the flux zone, then a reduction of these radioisotopes in reactor effluent water would be expected.

Effluent Monitor

Several pipetter breakdowns and a high voltage chassis failure resulted in a poor record of operation by the iodine monitor during the month. The pipetter was replaced by a Mini-pump and reliable operation should result.

A sampling system was installed which causes a sample of the effluent to be retained on signal from a high reading on the iodine monitor.

Release Studies

Tests to determine the release of polonium from irradiated bismuth were completed. Aluminum clad specimens held at 1300 C for two hours showed releases of 16 and 23 percent, not significantly different than the previously reported release of 17 percent in one hour. The formation of a sintered oxide skull completely covering the liquid metal retarded the oxidation rate and, consequently, the polonium release rate.

Polonium releases of 50 and 60 percent were obtained with bare specimens of irradiated bismuth held at 1300 C for one-half hour in helium and air atmospheres; however, the amount of bismuth volatilized was three times greater in a helium atmosphere than in air. Consequently, the release of polonium and bismuth are unrelated and volume diffusion predominated, although distorted by eddy currents.

SEPARATIONS PROCESSES

Disposal to Ground

The concentration of gross beta emitters in the ground water beneath the Purex process condensate crib, 216-A-10, dropped from 7×10^{-3} μc/cc in April to 3×10^{-3} μc/cc in May. Isotopic analyses of the ground water beneath the crib previously showed the presence of Sr-90 at a concentration of 3×10^{-8} µc/cc; analysis of the May 5 sample showed the Sr-90 to be less than detectable, $< 2 \times 10^{-8}$ µc/cc. Soil column studies were started at the Redox Laboratory to evaluate the decontamination benefits and prolonged crib life which might result by adding caustic to the crib influent. A Sr-85 spike is being employed to facilitate obtaining rapid analytical results.

Scintillation probe logging of all 200 Area wells is in progress to monitor the vertical and areal spread of ground and ground-water contamination from waste disposal sites. Data comparison will be made with previous surveys conducted in 1956 and 1959 to determine radionuclide migration rates. Preliminary evaluation of several out-of-service cribs and trenches revealed very little movement (on the order of several feet) of radioisotopes due to soil column drainage during the period from 1959 to the present.

A two component drive-point was designed and is being fabricated for use with a drive casing to obtain thermal conductivity and soil moisture measurements in tank farm subsoils. This work is a joint venture with the Chemical Development Subsection to provide the Chemical Processing Department with soils information pertinent to in-tank solidification heat dissipation problems. The neutron-neutron probe will be used to measure soil moisture content, and a sensor will be used to evaluate thermal conductivity. If good correlation exists between these two parameters, it may be possible to use soil moisture, an easier parameter to measure, as an index to soil thermal conductivity.

Ruthenium Removal Studies

Sorption methods of removing ruthenium from Purex process acid recovery condensate were scouted. Sorption of the untreated (pH 2) condensate on eleven types of activated charcoal (using one gram of sorbent per 100 ml of solution) gave removals of 18-66, 18-77, 27-89 and 51-93 percent in 24, 48, 144 and 312 hours, respectively. Neutralizing the acid waste to pH 7 increased ruthenium removal from 66 to 96 percent in a 24-hour equilibration period with the most efficient charcoal.

Ruthenium sorption on activated oxides produced by heating geothite, gibbsite and brucite gave removals from neutralized waste of 63, 78 and 85 percent, respectively, in one hour. Removals increased to 75, 96 and 97 percent after 26 hours. Removals on geothite and gibbsite at pH 2 were only 8 and 22 percent after two hours.

Four anion exchange resins (Dowex-1, WCR, A30B and IRA410) were tested for ruthenium removal from the raw waste. Ten percent breakthrough was reached after 160, 80, 50 and 30 column volumes, respectively. Due to the complexity of the system (electrophoregrams indicate at least three anionic species) appreciable loading occurs after 10 percent breakthrough. As an example, Dowex-1 maintained about one percent ruthenium breakthrough up to 30 column volumes where it rapidly increased to 10 percent breakthrough, continued at this level for the next 130 column volumes, then gradually increased to 25 percent breakthrough in the next 70 column volumes. Elution with 6 N NaOH was not too successful. For the most favorable case 10 column volumes of eluant removed only about 20 percent of the ruthenium loaded in 200 column volumes.

Plutonium Electrowinning

Continued experimental investigation of plutonium electrowinning, using the small cell described in the April report, has revealed some general trends. In successive runs at various temperatures, it was observed that an increase in temperature above 800 C resulted in a loss in current efficiency (from 45 percent at 800 C to 24 percent at 900 C and zero percent at 1000 C). An increase in the anode-cathode distance (from 1.0 to 3.5 cm) resulted in a small increase in current efficiency (from 35 to 45 percent).

Solid State Electrolysis

An electrolysis run of 77 hours' duration at 600 C was made on cerium metal containing a Zr-Nb-95 tracer and molybdenum as impurity. A current of 70 amperes at 0.90 volt yields a current density of 350 amps/cm^2 in this specimen.

Radiochemical assay on the 256 channel analyzer shows Sb-125 and Co-60 in addition to Zr-95. The counting data show only Co-60 migrates. The samples were analyzed spectrographically for non-radioactive impurities. If one considers that a change in concentration from end to end by a factor of four is certain evidence for migration, then iron and possibly nickel are the only elements which migrate. On this basis no migration was observed for silicon, manganese, magnesium, copper and molybdenum.

Electrolysis runs on Fe-59 and Co-60 traced cerium at 650, 600, 550 and 495 C are now complete and samples from these runs have been counted on the 256 channel analyzer. All runs were of 23 hours' duration.

The results can be conveniently expressed in terms of a transport number, T, defined as follows:

$$T = \frac{(C_t - C_o)\, 10^2}{C_o \text{ (time)}}$$

where C_t is the concentration of transported solute to the anode side of center at time t and C_o is the initial concentration to the anode side of center.

If the logarithm of the transport number, T, is plotted against current density for the Fe-59 data at 600 C, a good straight line results and this is also the case if potential drop is the variable instead of current density. The data at 495, 550 and 650 yield points off this straight line. Parallel lines drawn through these points and parallel to the curve at 600 C result in a family of straight lines.

The transport number at constant current density or at constant voltage drop can now be estimated from these curves. If the numbers so obtained are plotted on semi-log paper as a function of reciprocal temperature (an Arrhenius plot) a nearly straight line results, but the slope is negative.

A negative activation energy is not possible, and the explanation for this result appears to lie in the nature of the temperature dependence of two competing processes which occur. As electrolysis proceeds, the initially uniform distribution of the solute becomes non-uniform under the driving force of the potential field and electron flux, and this results in an opposing chemical gradient for back diffusion. If, as it appears, the back diffusion process accelerates more rapidly with increasing temperature than the electrolysis does, the net effect will be a decrease in electromigration with temperature. This is, in fact, observed.

WASTE MANAGEMENT AND FISSION PRODUCT EXTRACTION

Storage of Cesium and Strontium on Zeolites

Observation in B-Cell of the strontium loaded zeolite cartridge continued through the month, without evidence of radiolytic pressure buildup. The sudden appearance of pressure buildup last month when the temperature was increased above 550 C, and minor excursions observed this month, are believed to have been due to residual moisture in the sealed system exceeding the capacity of the zeolite for moisture retention. Both the number of moles of water available to form steam pressure in the system and the pressure per mole of steam increase with temperature, with the former effect being much more important in the region of interest. The proposed plant-size storage cartridges will be adequately stressed to contain the observed steam pressures. In addition, the pressurization phenomena should be less pronounced in the large cartridges since the zeolite near the outside of the container will be at a rather low temperature and will, therefore, have a high capacity for moisture retention. Failure to observe a time-dependent radiolytic pressure buildup may be due to recombination of radiolysis products, absorption of radiolysis products on the zeolite, or possibly to an undetected leak in the equipment.

<u>Isotopic Analysis of Strontium from Purex 106A Tank Sludge</u>

A sample of suspended sludge from the 106A tank was obtained. Besides determining promethium content, chemical composition, and defining a dissolution procedure, an isotopic analysis of its contained strontium was carried out. The Sr-90 content was found to be 52 percent, which compares favorably with about 55 percent for current Purex production and only 17 percent for the 103A tank, the only tank previously analyzed. The 106A strontium, which has cooled approximately two years and whose Sr-89 content is therefore negligible, is thus of high quality and attractive for recovery. A sample of sludge from the 104A tank, which also contains a soft and easily slurried sludge, is being similarly analyzed.

<u>Strontium Extraction from Dissolved Sludge</u>

Present CPD Waste Management plans are to mine the sludges from the Purex (and Redox) waste tanks, dissolve these in dilute acid, and to process the resultant solution in the future B-Plant complex to remove the heat-generating strontium followed by in-tank evaporation of the waste to a stable salt cake. Major drawbacks to this scheme are (1) very high chemical costs due particularly to the large quantities of complexant required to keep dissolved iron and aluminum in solution, and (2) the very large volumes involved, which require oversizing the B-Plant equipment. Schemes for processing the sludge which would avoid or minimize these problems are accordingly being sought.

One approach is to separate the strontium and iron ("iron extraction") prior to B-Plant solvent extraction. A precipitation flowsheet for doing this, based in part on earlier strontium recovery work, is being studied and appears quite promising. Acid-dissolved sludge is adjusted to pH 1 to 2 with sodium carbonate, the sulfate concentration butted to 0.7 to 1.0 M, heated very briefly to about 80 C, and lead nitrate equivalent to 0.02 moles per liter added. Ninety-five to 97 percent of the strontium is precipitated in less than one minute. The precipitate is washed with sodium sulfate solution, metathesized with sodium carbonate, and dissolved in a small volume of dilute acid for solvent extraction. Cake washes and metathesis wastes are recycled to reduce losses. Advantages of the process are elimination of organic complexing agents, short time cycles, and process simplicity. Because of the very short mixing and digestion times, a continuous or semi-continuous method is being considered, with the bulk of the precipitate to be removed by decantation through a tank equipped with a dipstick filter.

Promethium Isotopic Content

The discovery of Pm-146 in fission-produced promethium and a Pm-146/Pm-147 ratio for both Brookhaven and Purex 103A promethium were reported several months ago. Significance of Pm-146 is that its 0.45 and 0.75 Mev gammas would cause a slight, but significant, radiation dosage from massive sources of well-aged Pm-147, requiring somewhat greater shielding than would otherwise be needed for Pm-147 bremsstrahlung alone. The Pm-148 content of the promethium purified in January (and recovered about two years ago from current Purex 1WW) has now decayed sufficiently to permit measurement of its Pm-146 content. The ratio was found, unexpectedly, to be at least a factor of three lower than in the earlier material. The reason for a discrepancy of this magnitude is difficult to explain and suggests that some of the fundamental physical constants of Pm-146 may be seriously in error.

Extraction of Neptunium and Plutonium from Purex 1WW

Analytical data are now complete for experiments on the extraction of neptunium and plutonium from Purex Plant 1WW solution. As reported last month, following reduction of neptunium with hydrazine, 87-89 percent of the neptunium and 98-99 percent of the plutonium were extracted in an equal volume contact (10 minutes at 25 C) of the 1WW with 0.3 M trilauryl amine (TLA)-Soltrol. Further experiments indicate the extraction of neptunium can be increased to greater than 98 percent by increasing contact time to 30-60 minutes. Decontamination factors for the extraction contact were 300-1600, 16-20 and 40-160 for zirconium-niobium, ruthenium and cerium, respectively. When the organic phase was contacted with one-half volume of 0.1 M oxalic acid, about 80 percent of both neptunium and plutonium stripped. Overall decontamination factors (extraction and strip contact) were about 700, 600 and 100 for zirconium, ruthenium and cerium, respectively.

Observations from further experiments with synthetic 1WW were:

1. Extraction of neptunium and plutonium were not impaired when the extractant contained up to five volume percent tributyl phosphate or when the uranium concentration in the 1WW was increased from 0.00034 to 0.027 M.

2. Third phase formation occurred when 0.1 to 0.6 M Adogen 368 - Soltrol solutions were contacted with synthetic 1WW. Adogen 368 (40 percent C_8, 25 percent C_{10}, 30 percent C_{12}) is a new tertiary amine from Archer-Daniels-Midland Co.

3. Extracted cerium can be scrubbed effectively from the organic phase with 0.5 to 4 M HNO_3 at 60 C without significant stripping of plutonium.

4. Contact of the Np-Pu-bearing organic at 60 C with an equal volume of 1.0 M HNO_3 - 0.1 M $Fe(SO_3NH_2)_2$ stripped about 95 percent of the plutonium but only three percent of the neptunium. Cerium was also stripped.

5. Ruthenium left in the solvent after oxalic acid stripping was removed effectively with an alkaline permanganate wash.

CSREX Process - Laboratory Studies

Laboratory tests in which CSREX solvent (D2EHPA-BAMBP-Soltrol) was shaken with an equal volume of aqueous phase showed that sulfamic acid, hydrazine or urea in the aqueous phase prevented degradation of BAMBP when in contact with 1 M HNO_3 at 50 C for at least one day. This temperature and acidity exceed expected operating conditions in the 1B and 1C columns. Addition of sulfamic acid to the CSREX process feed solution has been shown to be satisfactory for preventing nitrite degradation to BAMBP under extraction column (1A) conditions. These results indicate means are available to prevent short-term damage to BAMBP by nitrite or nitric acid in the CSREX process. However, observations made during laboratory studies indicate that, once some degradation of BAMBP has occurred in the solvent, further BAMBP degradation occurs during long-term standing of the solvent out of contact with any aqueous phase. Long-term protection against BAMBP degradation may depend on devising a solvent wash procedure which will remove completely any BAMBP degradation products.

CSREX Process - Engineering Studies

Pilot plant studies were devoted to further investigation of the tartrate-complexed feed flowsheet described last month and to determination of the purity of the strontium-cesium product. Highlights of the results are as follows:

1. Strontium losses in the 1A column were generally two percent or greater, apparently reflecting a decrease in the strontium distribution ratio due to using tartrate instead of citrate to complex the feed and/or to iron and aluminum loading in the solvent. The latter reached 0.4 and 0.2 g/ℓ, respectively, after several passes. Cesium losses of less than one percent were achieved.

2. Cerium losses increased from three percent to 18 percent when the 1A column interface was lowered from the top to the bottom, resulting in decreased aqueous holdup time.

3. The concentrations of feed impurities reaching the 1BP (strontium-cesium product) were determined for a number of runs. The values varied, apparently randomly, over the indicated range:

Fe	0.0003 to 0.01 g/ℓ (10^{-5} to 0.002 M)
Al	0.004 to 0.04 g/ℓ (0.0001 to 0.001 M)
Cr	0.0002 to 0.005 g/ℓ (< 10^{-5} to 10^{-4} M)
Ni	0.01 to 0.06 g/ℓ (0.0002 to 0.001 M)

Except for nickel, the 1BX contained significant quantities of these impurities and therefore was a major contributor to the 1BP contamination. The nickel was carried over by extraction in the 1A column at a distribution ratio on the order of 0.1 to 0.2. For comparison, the strontium content of the product was nominally 0.0003 M.

Treatment of Purex Stored Waste Sludge

Development of procedures for removal of sludge from Purex waste storage tanks and removal of strontium from it were continued. Procedures involving an initial water leach followed by leaching with 1-2 M HNO_3 remove 75-80 percent of the strontium and 20 percent of the iron in the acid leach. Use of complexing agents such as citric acid and EDTA instead of the nitric acid did not improve strontium removal but did result in less iron being dissolved. Addition of oxalic acid to the nitric acid leach improved strontium removal (85 - 90 percent) but also resulted in greater iron dissolution. A procedure was tested in which the carbonate in the sludge was first destroyed with nitric acid; the slurry was then made alkaline (pH 13.5) and digested for 25 minutes at 80 C; supernatant liquid was removed and the solid was digested with nitric acid. Iron dissolution (33 percent) was significantly increased over that attained without prior caustic treatment. This approach will be studied further to see if a more rigorous alkaline pretreatment will so condition the sludge that all of the iron and the strontium associated with it can be dissolved in a subsequent nitric acid leach.

An experiment was performed in laboratory scale equipment to see if sludge disintegration could be hastened by use of a low pH leach liquor. Simulated sludge in a container representing an underground tank was covered with about five volumes of water. This solution was then circulated through a smaller container where acid was added to maintain the pH at about 5.5. Phosphoric acid was added first (as an acid carrier) until the solution was about 0.1 $\underline{M}$ in phosphate. Then nitric acid was added. Acid addition was controlled such that the solution flowing from the acid addition tank to the sludge tank was not more acidic than pH 5.5. Slurry in the sludge tank was not agitated; it was maintained at 80 C. About ten days were required to completely neutralize the carbonate in the sludge. During this period the simulated sludge, initially hard walnut-sized lumps very slowly attacked by hot water, disintegrated completely to a pumpable slurry. The pH of the slurry ranged from greater than nine at the start of the run to about seven at the end.

<u>In-Tank Solidification</u>

A new pilot scale model of an in-tank solidification system for intermediate level wastes has been designed and installation is proceeding. Initial tests are designed to confirm calculated heat transfer rates and scaling effects. The process as presently planned consists of passing heated air downward through an annulus and into the solution; as the air rises it transfers its remaining heat to the solution and also effects circulation of the tank contents. In the model design 100 cfm of air at 1200 F is used, with the annulus formed by concentric 1-1/2 and 3-inch pipes.

An alternative approach is being explored on a laboratory scale: that of direct immersion of heaters inserted in the air lift to avoid heating of excessive quantities of air. Using cold coating waste, a 2.5-to-1 volume reduction has been attained without scaling of the heater. The test is being continued to the point of solidification or heater failure.

Thermal Conductivity of Soil

Determinations of the thermal conductivity of tank farm backfill soils are being made in support of the Waste Management Program. The results of the tests will be used in estimating the heat loss rates from underground tanks of solidified waste.

Four in situ measurements have been made at a depth of four feet over Tank-101 in the 241-SX Tank Farm. Conductivities (Btu/hr/ft/°F) and corresponding temperatures were 0.33 (62 F), 0.43 (91 F), 1.2 (100 F), and 0.36 (73 F). The unexpectedly high conductivity of 1.2 is attributed to the presence of a buried object, such as a stone, of high thermal conductivity in contact with the probe. A deep scratch found on the probe after the test supports this theory.

Cesium Removal from Wastes

Column experiments with clinoptilolite and Linde AW-400 (20 to 50 mesh) were run with a synthetic Purex formaldehyde-treated waste simulating the current waste composition (1.5 M HNO_3, 0.6 M Na^+, and 0.4 M Fe^{+3}). The objectives for the experiments were to observe visually the exchanger and to determine the pressure drops across the columns in preparation for tests to be made soon with actual wastes. A significant pressure drop was not detected in the 20 cm long columns, at a flow rate of 3.3 ml/min/cm^2. One hundred column volumes of waste were passed through each exchanger, and the exchangers were then allowed to be in contact with the acid waste for 30 days. At the end of this time, no significant pressure drop was measured, and no visual degradation of either exchanger was observed. Either exchanger appears to be suitable for the plant tests.

Duolite C-3 resin is the most suitable exchanger for removal of cesium from Redox supernatant wastes. Fourteen load-elute cycles of a flowsheet were completed in an automatic column apparatus. The major difficulty encountered was precipitation of aluminum hydroxide in the lines and exchanger. To avoid this precipitation, washes of either 0.01 M NaOH or 2 M NaOH were used after loading to wash aluminate ions out of the bed and after elution to wash the ammonium carbonate (pH 9.4) out of the bed. Precipitation was avoided by a 0.01 M NaOH wash following loading and a 2 M NaOH wash after elution, prior to reloading.

The cesium concentration of Redox supernatant wastes is so low that it is beneficial to recycle the eluate from the ion exchange column for cesium removal to the column feed to increase the cesium concentration. Thus, the eluate from five column runs with Duolite C-3

[redacted] [redacted]

and synthetic Redox SX farm waste was boiled to remove ammonium carbonate, and to concentrate the cesium. The concentrate was then added to the influent of a sixth column run. The throughput capacity and kinetics of this sixth run were decreased by about one-third, and the sodium to cesium ratio in the eluate from the column was reduced to a satisfactory level.

Fission Product Packaging

Laboratory experiments were conducted to determine possible causes for the poor strontium loading on the Linde 4A zeolite used in the full-level Sr-90 packaging demonstration. The results of tracer experiments with a duplicate cartridge indicate that poor strontium loading may result when pumping the feed (0.2 M Na^{+}, 0.035 M Sr^{++}, and 0.010 M Ca^{++}) upflow through an initially dry bed of 4A. It is quite probable that channeling occurred along with a considerable evolution of gas from the bed of Linde 4A. Although much care was taken to fill the cartridge, there is no assurance that the bed is packed tightly enough to prevent channeling. The passage of gas bubbles upflow through a bed of zeolite, which is not held firmly in place, is almost certain to cause channeling.

Three experiments were run using the duplicate cartridge and the same lot of 14 x 30 mesh Linde 4A zeolite that was used in the full-level demonstration run. The feed was pumped upflow through dry 4A in the first experiment. The 4A had been previously dried in the cartridge at 600 C for 16 hours and allowed to cool. Considerable gas evolution occurred as the feed passed through the Linde 4A and strontium breakthrough exceeded 20 percent within two column volumes. The amount of strontium loaded through 30 liters was 1.4 meq/g. A second run was made under the same conditions except that the Linde 4A had been soaked in water for more than 5 days and loaded into the bed wet with no subsequent drying. Some gas evolution also occurred in this run which was probably caused by insufficient deaeration of the feed. Strontium breakthrough exceeded 5 percent within two column volumes. The strontium loaded through 30 column volumes was 2.0 meq/g. On the third run feed was pumped downflow through wet Linde 4A at a somewhat slower flow rate (10 cv/hr instead of 12 cv/hr). Breakthrough did not exceed 5 percent until 15 column volumes. Strontium loading was 2.2 meq/g through 30 column volumes. The temperature in all cases was 80 C. The prior heat treatment of the 4A in the Cold Semiworks did not appear to have a significant effect on the 4A used in these experiments.

EQUIPMENT AND MATERIALS

Scaling of Titanium in Concentrator Service

Titanium tube heat exchangers exhibit low heat transfer coefficients after use for concentrating Purex acid waste solution. Laboratory tests are underway to determine if titanium is more susceptible than stainless steel to scaling in these solutions. Heat transfer equipment (heat transferred to a solution in a reflux flask through a metal wafer clamped to flask bottom) is used. Simulated current Purex 1WW was refluxed in flasks having titanium or stainless steel wafers as the heat transfer medium with no apparent loss of heat transfer capability with time. When refluxing 1WW concentrated 20 and 30 percent, the solutions failed to boil through titanium wafers in 24 and 12 hours, respectively. No loss of heat transfer was noted for stainless steel wafers in either solution. A six-mil-thick scale of what appears to be ferric sulfate formed in four days on the titanium wafer boiling 30 percent concentrated 1WW. A similar scale on the wafer boiling 20 percent concentrated 1WW was less than one mil thick. Analyses to determine the scale composition are in progress. Other experiments in which titanium wafers are boiling simulated 1WW periodically butted with additional ferric sulfate are in progress.

Loss of heat transfer capability has been noted in titanium heat transfer tube bundles in the Purex waste back-cycle concentrator (H-4). Silica is the suspected scaling agent in this case. Solutions six and 10 molar in nitric acid and butted periodically with sodium silicate or preformed colloidal silica are being boiled through titanium or stainless steel wafers (bulk metal temperature 145 C). Heat transfer capability has been maintained in all cases to date (30 days exposure) except one. In this case, 10 M HNO_3 butted with preformed silica was boiled through a titanium wafer. A similar experiment utilizing a titanium bayonet to provide a vertical instead of horizontal surface is in progress.

Stress Corrosion Cracking of Mild Steel

Stressed (bend beam) and notched specimens of mild steel ASTM A-283 Grade C were exposed to 50 w/o $NaNO_3$ at 90 C with and without cathodic protection. Both specimens exposed without cathodic protection were cracked after 30 days' exposure. Specimens cathodically protected (30 μamp/sq cm) have not cracked to date (55 days exposure). A potentiostat which will permit cathodic protection experiments on large (3 feet by 3 feet) weldments has been ordered.

Non-Metallic Materials

Chemical compatibility tests (in nine typical plant solutions) were completed on eight silicone rubbers and one each adduct, styrene, hypalon and natural rubbers.

A glove port ring attached to a 3/8-inch sheet of polymethylmethacrylate with sheet metal screws in untapped holes was examined in polarized light for strain patterns. The ring showed significant strain pattern but this was not affected by the screws. It appears this would be an acceptable method of attaching glove port rings to hood panels.

PROCESS CONTROL DEVELOPMENT

Chemical Compatibility of Scintillating Glass

A sample of Nash and Thompson Type GS1 scintillating glass was tested in a 20 percent TBP - 80 percent CCl_4 solution which had been equilibrated with 60 percent HNO_3 and 98 percent H_2SO_4. After 528 hours at 50 C and 192 hours at room temperature there was no visible change in the appearance of the glass. This same glass devitrified in 0.25 M HNO_3.

Purex Canyon Instrumentation

A new type of jumper containing two electrical connectors and two specially designed coaxial connectors was successfully installed in Purex E-Cell. The jumper was developed to alleviate problems caused by the lack of spare electrical connectors and the need for remotable coaxial instrument connectors to permit installation of electronic gear on the cold side of the canyon. In conjunction with the jumper installation a BF_3 tube and pre-amplifier were installed to monitor plutonium concentration in the E-6 tank, which contains feed for the first cycle solvent extraction battery.

Air Pulser Amplitude Control

Development work on control instrumentation for the air pulser in the new Plutonium Reclamation Facility was concentrated on the development of techniques and instruments for controlling pulse amplitude. Following up earlier work to evaluate several methods of bi-directional flow measurement, studies have been made of the accuracies attainable with the various instrumentation systems. Circuit design and testing have demonstrated that the average value of an irregular pressure valve can be determined to an accuracy of about one percent. Using a strain gage flow meter the pulse amplitude-frequency product can be controlled within two percent. With frequency compensation, pulse amplitude can be controlled to ± 15 percent. Some improvement can be expected with careful circuit design; however, control of amplitude is inherently less accurate than that of the amplitude-frequency product, due to the change in wave shape with both amplitude and frequency.

Testing of a diaphragm-type differential pressure cell under conditions expected to yield a signal relating to the amplitude-frequency product actually gave results more nearly correlated with the square of product. Analysis indicates that this effect is due to mass acceleration in the measuring system. Close coupling of the D/P cell to the orifice is expected to reduce or eliminate this effect.

Remote Weight Indicator and Alarm

Instrumentation for monitoring the accumulation of plutonium oxalate in the button line hood during the filtration step has been developed and tested. Circuitry was designed to read out the weight of oxalate collected on a tray supported by a cantilever beam. Deflection of the beam is measured with a linear variable differential transformer. The transformer output is fed to a magnetic amplifier whose output drives a contact meter relay. The meter indicates the weight of accumulated oxalate and has an adjustable alarm level. The full scale span of the device can be selected in the range of 0.5 to 5 kg. The completed system is not ready for plant installation.

Advanced Process Control Development

The program directed toward development of a mathematical model of the C-Column progressed to the point where all scheduled runs of the experimental column have been completed. Detailed analyses and correlations are underway to check the adequacy of the data and to determine the values of the model parameters. Adaption of the column and associated instrumentation for demonstration of on-line computer control have been initiated. Based on present schedules, this phase of the work should be completed well ahead of the estimated delivery date of the process control computer.

A detailed investigation was made of an apparent discrepancy between product stream (ICU) uranium analyses made by the mid-column photometer and by the gamma absorptiometer. The photometer analysis is made at a point in the column between the feed plate and the ICU exit; the absorptiometer sample is taken external to the column. The difference in uranium concentration is real and can be readily explained by a revision in the model which allows for mass transfer between phases in the region below the feedplate. The practical significance of this effect is being explored, in terms of optimum feed points and effluent stream take-off locations at both the top and bottom of the column. A computer program has been written to investigate this phenomenon, and to compare the results with those obtained by another investigator[1] under similar feed conditions.

---

1. Miyauchi, Ind. Eng. Chem. Fund. 2 #2 (May 1963) p. 113.

REACTOR DEVELOPMENT - 04 PROGRAM

DECLASSIFIED

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Chlorine Liquefaction Studies - Projected reprocessing of 15-day-cooled PRTR type fuel elements by the Salt Cycle Process yields an off-gas stream laden with chlorine and iodine-131. The present off-gas treatment method includes a caustic scrubber which generates a large volume of caustic waste solution. Study has commenced on an alternative, low-temperature liquefaction (-100 C) method which offers the possibility of efficient iodine-131 removal, allows reuse of the chlorine, and hopefully will result in negligible waste volume.

Two experiments have been completed on the liquefaction of chlorine from a chlorine-air mixture. In the first run chlorine at 300 ml/min and air at 100 ml/min were passed through a small cold-finger trap immersed in a dry ice-acetone bath (-78 C). The amount of chlorine liquefied was 96 percent of the total passed through the trap and 99.5 percent of the total possible at the temperature employed as determined by vapor pressure considerations.

In the second run a 50 percent chlorine - 50 percent air stream was passed at four liters per minute through a modified 15 lb. chlorine cylinder immersed in a dry ice-trichloromonofluoromethane batch (-78 C). The theoretical recovery of chlorine at the above temperature and flow rate is 92.5 percent. The actual recovery was at least 80 percent but not more than 88 percent based on a material balance using an analysis of a caustic scrub solution in the exit line. The liquefied chlorine was retained in the cylinder for four days without leakage before discharge, reaching a pressure of 125 psig at 27 C (chlorine vapor pressure at 27 C is 100 psig) possibly indicating the presence of dissolved air.

Crushing and Sizing of Electrolytic UO_2 - A commercial paint conditioner was modified and tested in the dual role of sieve shaker and pulverizer. The shaker was tested with -3 +200 mesh UO_2 feed and a stack of three 8-inch-diameter sieves. Proper operation in 15 minutes was achieved only at 200 shakes per minute with no part of the sieves on the centerline of the rotary shaking motion. At 100 shakes per minute many small particles remained on the top sieve and at 400 shakes per minute there was degradation of the largest particles. A 30-minute pulverizer run was made at 1610 shakes per minute in a pint container loaded with steel balls

[redacted] [redacted]

and a feed of one pound -3/4 inch +1/2 inch UO_2 particles. The product consisted of 51 percent -200 mesh material with no particles larger than 28 mesh.

Fuel Processing Economic Study - A comparison of the relative economics of close-coupled Salt Cycle fuel reprocessing with central plant reprocessing was reported last month for the case of a graded fuel cycle reactor operating with a 14,500 MWD/T fuel exposure. Efforts have since been devoted to extending this comparison to higher fuel exposures (up to 40,000 MWD/T) to evaluate the effect of the fuel exposure parameter on the relative fuel cycle costs for the two reprocessing concepts and to enable a comparison at optimum exposure in each case.

These computations have been completed for the central plant cases and indicate optimum fuel exposures around 35,000 MWD/T for the leased uranium cases and around 30,000 MWD/T for the purchased uranium cases. The fuel cycle cost was decreased anywhere from 0.15 to 0.5 mills/kwh at optimum fuel exposure. These optimums were calculated assuming constant fuel fabrication costs. A correction was estimated for increased fabrication charges with increased exposure and this resulted in reducing optimum exposures about 5000 MWD/T and increasing the minimum fuel cycle cost at optimum exposure from 0.1 to 0.2 mills/kwh.

The computations for the close-coupled, Salt Cycle cases were halted to modify the computer program to handle a wider range of make-up uranium enrichments. This modification is nearly completed and the program should be ready for use again early next month.

Salt Cycle Flowsheet Studies - Laboratory studies of the parameters governing the preparation of UO_2-PuO_2 solid solutions by electrocodeposition from chloride melts have continued. A plutonium enrichment factor of 40, with a promethium decontamination factor of 6 (plutonium basis), has now been achieved, with the preparation of a deposit containing 23 weight percent PuO_2. In this run, a platinum cathode was used at a current density of 0.05 amp/cm^2; the melt was LiCl-KCl at 575 C; and a 50 percent O_2 - 50 percent Cl_2 sparge was used. Even though some of the deposit fell from the platinum, over 50 percent of the plutonium was removed from the melt.

From the observations to date, it appears increasingly certain that the reaction proceeds via the formation of plutonyl(V-VI) species as a result of reaction with the oxygen-chlorine sparge gas. These penta- and hexa-valent plutonium species may then be cathodically

[redacted] [redacted]

reduced, along with uranyl(VI), to form the codeposit. Obviously, control of the PuO_2 enrichment in the deposit requires regulation of the relative rates of PuO_2 and UO_2 deposition on the cathode. One such means of regulation is through the current density, since changes in current density have a much greater effect upon the rate of deposition of UO_2 than of PuO_2. Accordingly, the highest enrichments have been achieved at the lowest current densities. The PuO_2 deposition rate is closely linked with those variables which influence the rate and extent of oxidation of plutonium to the plutonyl(V-VI) states. These variables (chloride activity of the melt, temperature, oxygen/chlorine ratio in the sparge, cell geometry, etc.) interact with each other, and have limits imposed upon them by other considerations, to the point that it is difficult to generalize about their effect. For example, the PuO_2 deposition reaction is promoted by increasing the oxygen content of the sparge gas, at least to the point at which PuO_2 precipitates by chemical action. The critical oxygen/chlorine ratio depends upon the melt composition, being lower in KCl-2.5 LiCl than in KCl-LiCl, for example. Plutonium enrichment in the deposit increases with temperature, apparently up to the temperature at which interference from chemical reaction involving the graphite anode becomes excessive (625 C for the LiCl-KCl system). Plutonium enrichment has been found to be greater on the side of the cathode nearest the anode, and is strongly affected by the quality of the graphite anode, deposits obtained in runs in which the anode disintegrated having PuO_2 contents about half those of normal deposits. Thus it is evident that, although the way appears open for the production of a wide range of PuO_2 - UO_2 solid solution compositions by electro-codeposition, rather careful choice and control of the conditions will be required.

In the course of this study, the following samples of PuO_2 - UO_2 were prepared for use in in-reactor fuel tests:

1. 580 grams, ranging from 0.34 to 0.95 w/o PuO_2 and averaging 0.53 w/o. The target was 0.5 w/o.

2. 110 grams, ranging from 1.72 to 3.40 w/o PuO_2 and averaging about 2.7 w/o. The target was 5 w/o, but PuO_2 precipitated, resulting in poor efficiency and a poor deposit.

3. 475 grams, not yet assayed. The target was 2.5 w/o.

<u>Chloride Contamination in Electrolytic UO_2</u> - During the month, increased attention has been given to the problems of measuring and reducing the extent to which electrolytic UO_2 is contaminated

with chloride. The chloride determination problem has been attacked via the coulometric titration of chloride ion with silver ion, using an amperometric indication of the end-point. Preliminary results indicate that the method should give reasonably good results for as little as 8 ppm chloride in solid UO_2 and at somewhat higher concentrations should be capable of an accuracy of ± 5 percent.

One test has been made of a pyrohydrolysis technique for the removal of chloride from UO_2. Using unwashed, -150 +200 mesh electrolytic UO_2, a three-hour treatment with a 4 percent H_2O, 5 percent H_2, 91 percent N_2 gas mixture at 950 C reduced the chloride content from a level of 1000 to 125 ppm.

Electrochemistry of Uranium in Molten Chloride Salt Solutions - Recent receipt of analytical data has made it possible to calculate additional thermodynamic values for the reaction

$$UO_2 + Cl_2 = UO_2Cl_2.$$

The values of the free energy change for various salt systems containing 0.1 molal UO_2(VI) are tabulated below, in a form which reflects the effect of the melt composition:

Melt Composition	$\Delta G^{570\ C}$ (kcal)	$\Delta G^{670\ C}$ (kcal)
0.42 KCl - 0.58 LiCl	-24.8	--
0.50 KCl - 0.50 LiCl	-25.8	-24.2
0.26 NaCl - 0.74 LiCl	-18.9	-18.2
0.50 NaCl - 0.50 KCl	--	-26.3

It was recently realized that incorrect values of ΔS and ΔH had been reported for this reaction in previous monthly reports (April, August, and December, 1962). In order to correct this situation and to provide an up-to-date summary of the thermodynamic data accumulated thus far, as calculated for 1.0 molal UO_2Cl_2, the following table is provided:

Melt Composition	T^oC	ΔG (kcal)	ΔS (e.u.)	ΔH (kcal)
0.42 KCl - 0.58 LiCl	550	-25.1	-14.2	-36.8
0.50 KCl - 0.50 LiCl	550	-26.2	-15.3	-38.8
0.26 NaCl - 0.74 LiCl	670	-17.9	-10.1	-27.4
0.50 NaCl - 0.50 KCl	720	-25.9	-15.8	-41.6

It is apparent from the revised values that previously-drawn conclusions that uranyl(VI) complexation decreases with increasing uranyl(VI) concentration are no longer valid.

Chronopotentiometric studies of the electroreduction of UO_2(VI) have been made in various melts to determine whether or not the reduction mechanism and the diffusion properties of UO_2(VI) change with melt composition. The diffusion coefficient for the UO_2(VI) species in NaCl-LiCl at 704 C was found to be about two-thirds what it is in NaCl-KCl at 713 C; at 650 C in NaCl-LiCl, it is about three-fourths what it is in KCl-LiCl (eutectic) at the same temperature. The reaction appears to be reversible and diffusion controlled at temperatures above about 550 C and at UO_2(VI) concentrations above about 6×10^{-3} molal.

Preparation of UOS - Three samples of mixed UOS and β-US_2 were prepared by precipitation from a chloride melt. These samples have since been washed with dilute HCl solution in an effort to remove the β-US_2. X-ray diffraction patterns of the washed samples indicate that the treatment was successful in producing crystalline UOS uncontaminated with either β-US_2 or UO_2.

RADIOACTIVE RESIDUE PROCESSING

Eighteen-Inch Radiant-Heat Spray Calciner

During the month six runs were made in the Cold Semiworks reactor to (a) produce powder product for anticipated corrosion testing studies, and (b) continue evaluation of column capacity and filter blow-back arrangement.

Two short runs of about 120 liters each were run with (a) simulated neutralized waste with 150 g/ℓ sugar added, and (b) acidic waste adjusted with phosphoric acid, calcium ion and additional sodium ion to produce a phosphate melt. Powders from these runs will be used in studies on materials of construction.

In the run with neutralized waste the sugar addition was sufficiently low that oxidizing conditions were maintained in the reactor in order to avoid the suspected conditions for rapid corrosions due to sulfidation. Very few wall deposits were noted although the deposition of powder on the filter elements was not removed by the pulse blow-back conditions used. It is anticipated that a higher blow-back pressure would have been satisfactory as the powder was completely removed by blow-back on cooling.

In the other short run, phosphate, calcium and sodium were added to simulated acidic FTW such as to satisfy the following ratios $Ca/SO_4 = 0.8$; Na/(Fe + Al) [normality] = 1.0; PO_4/(Na + Fe + Al) [normality] = 1.3. The resultant feed was a slurry which was atomized without incident after initial difficulty with partial calcination in the nozzle due to excessive preheating of the feed. The powder product melted to a free flowing liquid below 950 C. Weight loss on melting (probably resulting from driving off sulfate) was less than two percent. Sulfate in condensate was less than 0.02 percent. Solubility and structure (glass vs. microcrystalline) have not been evaluated.

A run at 25 gph was made with an external-mix nozzle and runs at 20 and 25 gph were made with an internal-mix nozzle. These represent the highest rate obtained as yet in the reactor. Operation with both types of nozzle was satisfactory although the buildup of powder on the wall was less with the internal-mix nozzle as indicated by the stability of the various internal temperatures. Steam consumption was also markedly less [steam to feed weight ratio = 0.25 (internal-mix) and 0.43 (external-mix)] and consequently the filter pressure drop was less [12 vs. 18 in. water]. With the present arrangement of thermocouples it is not possible to detect precisely the point at which the column becomes overloaded. However, outlet temperatures of gas were at about 225 C and it seems unlikely that capacity could be increased very much above the 25 gph rate.

One run at about 12 gph was continued for 24 hours, the limit of present feed tank capacity. Principal purpose was to observe behavior of the filter blowback system. Pressure drop initially plateaued at 10 in. water but then started climbing after 12 hours and reached a value of 17. in. water for the last three hours. Superheated steam at 40 psig was used for blow-back. The use of 60 psig steam at the end of the run reduced the pressure drop to 15 in. water. The venturis on the blow-back assembly are being altered to conform with the configuration of the more favorable venturi design of two of the existing arrays. One notable observation during the run was the very small amount of wall deposits (agglomerates) noted in the powder product. Operating temperatures and conditions remained essentially constant throughout the runs with very little attention being required to maintain stable operation.

Waste Calcination Instrumentation

It was previously reported that spaced thermocouples can indicate pot powder levels by means of the temperature drop which results when cooler powder covers a thermocouple. It has been found that this technique is not adaptable for melt level measurement. The difficulty seems to be that an insufficient difference in temperature exists between the melt itself and powder which floats on the melt and is about to fuse.

Electrical conductivity was investigated as a melt level detection method because previous fused salt work and handbook data had indicated that fused salt resistivities are on the order of 0.1 ohm - cm. This value would yield adequate sensitivity for melt level detection. The simulated calciner product powder obtained from the radiant spray calciner had a resistivity of about 1 ohm - cm which, although 10 times the predicted fused salt value, will provide adequate sensitivity for use as a melt level indication. Development work to date indicates that each level detection method is specific in that thermocouples will indicate only powder level and conductivity probes will indicate only melt level.

Full-Level Pilot Plant Calcination Studies

Analytical work was completed on two pot runs reported last month (#14 and #15) and verified earlier conclusions, i.e., that there was a large cesium carry-over in the high-sodium-nitrate run and that calcium was at least as effective as magnesium in preventing sulfate volatilization. A pot run, #16, was made duplicating run #14 in an attempt to determine whether cesium actually volatilized or whether it reached the condensate via a boilover associated with plugging of the off-gas line. No plugging or apparent boilover occurred in run #16, but cesium was observed in the condensate and off-gas particulate sampler during the high temperature heating phase of the run. No cerium was detected in the above samples which shows that no gross boilover occurred. The fact that cesium was picked up on the filter samplers and the absolute filters suggests that it came over in a particulate form, although volatility is not ruled out. This particulate could have resulted in the final drying or calcining of the aqueous liquid or the melt which was enriched in cesium because of the higher solubility of cesium salts relative to the other fission products. The total amount of cesium evolved from the pot in run #16 was small and essentially none penetrated the absolute filters in the off-gas train. Run #16 has shown that a small amount of cesium carryover results during high sodium nitrate waste calcination and that the large carryover in run #14 was caused almost entirely by boilover.

The non-radioactive 8-inch diameter by 10-foot tall spray calciner has been repaired and returned to service for piloting of runs proposed for the hot-cell unit. Several runs were made with sugar-containing acidic and alkaline Purex waste to study corrosion. Although metallographic examination of corrosion coupons is not complete, no catastrophic corrosion was observed and it is now thought that the leak in the in-cell calciner was probably due to a brittle weld. Efforts to calcine a tartrate complexed, alkaline Purex waste (plus sugar) were less successful than with sugar alone. Apparently the tartrate complex resists pyrolysis and tends to result in "tars" in the nozzle and unreacted organics in the product. Pumping and atomizing a slurry, which has given no difficulty either in the small hot-cell calciner or in the large 321 Building unit, would appear preferable to use of tartrate.

Exchanger Properties

Work continued on the zeolite binary equilibrium systems. Mass action quotients of univalent-divalent, univalent-trivalent, and divalent-divalent systems have been corrected for solution activities and are in the process of being corrected with zeolite-cation activity coefficients. The results will enable the determination of several thermodynamic exchange constants.

A quantity of $K_2CoFe(CN)_6$ was synthesized. X-ray diffraction patterns of this material were identical to a sample of $K_2CoFe(CN)_6$ obtained from the Savannah River Laboratory. Ion exchange capacities of the two materials were identical at about 2.0 meq/g. The equilibria in a potassium-cesium system was determined. At $K_2CoFe(CN)_6$ cesium loadings of greater than 10 percent, cesium selectivities were no better than several of the zeolites. Below 10 percent cesium loading, however, $K_2CoFe(CN)_6$ was more cesium-selective, especially in acid solutions. The $K_2CoFe(CN)_6$ tended to be unstable in basic solutions. The above behavior suggests that $K_2CoFe(CN)_6$ would be of little value for a radioisotope packaging media, but could be of interest for cesium extraction from some waste solutions.

Condensate Treatment

Installation of the engineering scale steam stripper and associated equipment was completed except for trim in the feed flow control valve. The equipment was satisfactorily operated (manually controlling the flow rate) using distilled water as feed. The 5000 gallon tank trailer was received. Minor modifications will be made to the outlet piping to improve operability before placing it in service.

Electrostatic Bubble Scrubber

Studies continued of the parameters controlling electrostatic charging and deposition of 0.3 μ particles on dry collectors and onto bubbles formed from air containing the charged aerosol. Experiments were carried out which demonstrated that the fraction of particles collected in the particle charging section increased with charging current. Particles not retained in the charging section were readily deposited in a simple collector held at the potential of the charger, supporting the conclusion that all or very nearly all particles were charged with the system used. When this charging section was used in connection with the bubble scrubber, a large fraction of these particles failed to be collected on the bubble wall. Some 98 to 99+ percent of the particles, after passing through the scrubber, were found to retain some charge. This was shown by their efficient deposition in an electric field after having passed through the scrubber.

Tests of scrubbing efficiency were made with conducting and non-conducting orifice plates used for forming the bubbles. A brass sieve plate with 0.016-inch holes yielded higher efficiencies than a similar acrylic plastic plate. The brass plate with 0.016-inch holes gave higher scrubber efficiencies than a brass plate with 0.032-inch holes. At the higher charging currents, however, the advantage of the smaller orifices was not apparent.

Making the solution conducting with sodium hydroxide did not change the scrubbing efficiency.

Tests of a single orifice scrubber with a needle corona charger oriented centrally with and near to the orifice were made for several liquid depths. The scrubber was 98 to 99.9 percent efficient. The efficiency did not depend on liquid depth. The fraction of particles deposited on the back surface of the orifice used was not measured in these tests. Electrostatic deposition directly from the corona region onto the forming bubble surface does give high removal efficiencies in the ensuing passage of the bubble through the liquid.

<u>Columbia River Sediment Studies</u>

A river sediment core sampler was designed, fabricated and tested which will facilitate obtaining sediment samples from the Columbia River. The sampler is 6 inches in diameter and can take cores up to about three feet in length. It incorporates a liquid-filled plastic ball which operates as a ball check valve, allowing the sample to enter without restriction and then sealing. To maintain the seal during withdrawal of the sampler from the river bed, a vacuum is drawn in the space between the sample and the ball. This effectively prevents any washing of the sample during the trip to the river surface. The larger diameter cores provide larger samples suitable for more sensitive analyses.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

The electrical analog grid spacing formulation made last month was modified, and the programming needed for functional evaluation is in progress. This study will yield information on the error introduced by going to unequal grid spacings (warping) on the three-dimensional resistance network.

Rather significant progress was made in obtaining an integral expression for the permeability along a streamline as a function of ground-water potential, ϕ. The resulting mathematical expression involves integration along a streamline commencing at a point of known permeability, K_o, (associated with ϕ_o) and ending at ϕ to obtain the permeability, K, at that point. The equation provides the relationship needed to deduce the permeability distribution from the measured potential distribution and the boundary condition in permeability.

Improvements were made in the Am-241 soil column moisture detecting electronic system to lower the background count and achieve better stability. In a 2-inch I.D. Lucite cylinder with a wall thickness of 1/4-inch, the degree of soil saturation can be measured to ± 1 percent expressed on a pore-volume basis. This method of measuring moisture content (gamma ray attenuation) will permit the measurement of permeability, capillary pressure, and moisture content to be determined in the laboratory from a single-step operation rather than a two-step operation.

A modified method for installing piezometer tubes in wells is being evaluated theoretically. The method involves using only sand to fill the casing between tubes rather than cementing-off a narrow zone between the sand fill. This would simplify and speed up piezometer tube installation considerably. If the evaluation results show that sand is effective in dissipating the head differences between zones, a field test will be run employing sand only in a well immediately adjacent to a well in which piezometers are already cemented.

Geological studies in Cold Creek Valley disclosed that the valley form is only locally and incidentally the result of erosion by surface streams, for no single stream course follows the valley floor. Movement of contaminants, as potentially from the Radioecology

Site at Rattlesnake Springs, will be minimal by surface runoff and their potential movement to the Yakima River by such mechanisms is practically non-existent.

Surface runoff is appreciable only in the uppermost part of Cold Creek Valley, west of the Yakima Barricade, where the structural (synclinal) form of the valley concentrates runoff. The valley west and southwest of 200-West Area is the result of deposition of fine materials by runoff as the water infiltrated into the ground. From Rattlesnake Springs to the Yakima River the valley is dominantly the result of wind erosion of fine-grained sediments initially deposited as a belt along the flanks of the Rattlesnake Hills. Surface stream courses exist only where local concentration of runoff occurs. Runoff from Rattlesnake Springs to the Yakima River evidently has not occurred within historic times, if even within more than 5000 years.

RADIOLOGICAL AND HEALTH CHEMISTRY

Iodine Studies

Reports that silver and copper mesh are good adsorbers for I_2 have been confirmed in the laboratory. At a flow rate of 25 cfm through 4-inch diameter, 2-inch thick beds of silver or copper (preceded by an AM-1 filter to remove particulate material and followed by a charcoal bed to remove the remaining iodine) 96 percent and 50 percent, respectively, of the I_2 was removed. In a second test HI-131 was generated and found to be removed by these materials with an efficiency of 99 percent and 90 percent, respectively. This collection apparatus was used on the gas from the Redox Plant at a time when there had been no operation for two weeks and the I-131 emission was very low with little accompanying NO_2. Only 7.4 percent and 2 percent of the non-particulate I-131 was removed from the gas stream by silver and copper mesh, respectively. It thus appears that the I-131 from this gas stream is in chemical forms other than I_2 or HI. Further studies of these forms are planned.

Radiation Chemistry

Radish seeds which had been soaked overnight in either 1 percent erioglaucine solution or in plain water were prepared with water contents of 1.4 percent ("dry" seeds) or 20.1 percent ("wet" seeds). These seeds were irradiated in groups of 10 with Co-60 γ-radiation and then allowed to germinate immediately on paper pads soaked with 10^{-3} M potassium gibberellate. No protective effect of the dye was observed. The data indicate an interesting difference between radiation sensitivity of wet and dry seeds. The wet seeds show an abrupt loss in germination ability at 1.0×10^6 rads whereas dry seeds lose this ability more gradually over the dose range of 0.5 to 1.5×10^6 rads. The presence of large amounts of water appears to make their radiation response more uniform. Due to the procedure used to incorporate erioglaucine into the seed most of the dye is concentrated near the seed surface. The more uniform radiation sensitivity of "wet" seeds may mean that seed constituents throughout the seed are equally important in the germination process and that the entire seed must be protected. A microscopic examination of the distribution of dye within the treated seeds is in progress to determine whether other modes of treatment might provide more uniform dye distribution.

Oxygen has been observed to accelerate the rate of disappearance of the free radical signal observed in the ESR spectrometer when a solution of p-nitroaniline is studied following irradiation with cobalt-60 gamma radiation. This observation has led to improvement in the liquid handling system so that air is more rigidly excluded. The resulting measured lifetime of the free radical produced by irradiation of p-nitroaniline solution at room temperature was found to be 35 seconds, about double the previously measured 14 seconds.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Aerosol Sampling Study

The two-foot square wind tunnel was used in initial experiments designed to measure sampling errors when sampling rates are much smaller than isokinetic. This work is being done in support of Atmospheric Physics research. These initial experiments were to evaluate uniformity of particle concentration in the test section and to establish proper techniques in sample collection and sample evaluation. These experiments indicated that particles may be stratifying or streaming in preferred areas in the cross section, even after several circuits in the wind tunnel loop. Further study of this condition will be needed.

ISOTOPE DEVELOPMENT - 08 PROGRAM

Pyrochemical Cerium - Trivalent Rare Earth Separation

Preliminary scouting studies, based on work by Johnson and Olson[1]

1. Johnson, R.W. and E.H. Olson, Separation of Cerium from Other Rare Earths by Ignition of the Nitrates, ISC-1069, September 1958.

were initiated on a pyrochemical process for the separation of cerium from the trivalent rare earths contained in the CSREX process 1CP stream. Pyrochemical processing may permit more compact equipment and eliminate radiolysis problems expected to be severe in the aqueous batch solvent extraction process currently considered for the Hanford Isotope Plant. A conceptual process would employ evaporative concentration of the CSREX 1CP stream with added $Mg(NO_3)_2 \cdot 6H_2O$ as a fluxing agent to a boiling point of 200 to 275 C.[3] Thermal decomposition of cerous nitrate to CeO_2 occurs in this temperature range and the resulting slurry would be diluted with water for filtration of the precipitated CeO_2. Hopefully, 99 percent of the cerium could be recovered at about 95 percent purity.

Initial studies have been aimed at determining fluxing agent concentrations necessary for maintaining a fluid, non-mastic system throughout the evaporation and at the reaction temperature. For the expected rare earth composition in the CSREX 1CP, the addition of 50 w/o $Mg(NO_3)_2 \cdot 6H_2O$ (based on the hydrated rare earth nitrates) results in a fluid melt throughout the operating temperature range. Addition of 25 w/o fluxing salt was insufficient to maintain molten conditions. Oxidation of the cerium was found to occur below 200 C with rapid reaction in the 260 to 275 C range.

The cerium oxide formed by thermal decomposition of cerous nitrate in the nitrate fusion can be separated (after water dilution of the reaction mixture) by filtration either on 10 micron sintered metal filters or on No. 1 filter paper. Use of the latter might result in equipment simplification by permitting ashing of the filter element during subsequent drying and firing of the CeO_2 product.

Attempts to form a cesium borosilicate glass by the Dynapak compaction techniques which were so successful with strontium titanate and rare earth oxides have been unsuccessful. In all cases, plugging of the off-gas line followed by distortion or rupture of the can occurred when it was heated in the furnace prior to compaction. Removal of residual nitrate (using cesium nitrate as a starting material) was particularly difficult and required temperatures in excess of 800 C. It is concluded that high energy compaction is not an attractive or practical process for forming cesium glass (although it may be applicable to salts such as cesium sulfate).

W. H. Reas
Manager
Chemical Laboratory

WH Reas:cf

[redacted]

BIOLOGY LABORATORY

A. ORGANIZATION AND PERSONNEL

No significant changes occurred during the month.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

Young chinook salmon, after being reared for 17 weeks in reactor effluent, are currently under swimming tests to determine the effect of effluent concentration and type of corrosion inhibitor on swimming performance. In these tests the fish are stratified by size and distributed into treatment groups which are scored for performance at three different water velocities. Within a particular size group, the early results show no apparent difference due to treatments; however, it is clear for the zero-age chinooks (body length about 56 mm) that a difference as small as 5 mm in body length affects swimming ability. The size of the fish is one of the most important factors on swimming ability; i.e., the larger fish swim better than the smaller fish. Therefore, the average fish reared in 3% effluent would be expected to swim better than the fish reared in 6% effluent because the effect of being reared in higher effluent concentration was a significant growth depression.

Vertan 690

To help evaluate the biological significance of river disposal of Vertan 690 on fish life, young trout and chinook salmon are under test. The purpose of the bioassay is to determine the acute toxicity of this proprietory material which is to be used to clean and condition the NPR secondary system. Initial tests show total mortality at 3000 ppm (defining Vertan 690 as received as 100%) and 4000 ppm for young trout and chinook salmon, respectively. First estimates of LD_{50} will probably lie between 2000 ppm and 3000 ppm. If an arbitrary safety factor of 1% of LD_{50} concentration is applied, an upper limit to avoid acute toxicity of about 20 ppm is computed.

Columnaris

Tests for heat killing of columnaris showed that 50 C for one hour was the minimum required for 100% mortality. With increasing temperature, shorter times were required.

In the course of the temperature tests a high frequency of a new colony form of the organism was observed, a smooth as opposed to the usual rough form. Some features of this smooth form make it ideal for laboratory work.

Chinook fingerlings, survivals from previous exposures to columnaris, were re-exposed to the organism and their survival compared with a group of fingerlings having no previous exposure history. There appeared to be a slightly lower mortality in the previously exposed group, but no clear cut resistance had been produced.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY AND TRANSFER OF RADIOACTIVE MATERIALS

Zinc

Results of trout killed 164 days after a single oral dose of 46 µc showed that among the tissues sampled the gill filaments still have the highest concentration and about 1.8% of the given dose is estimated in the gills. Autoradiographs of gill filaments are being prepared for further study. The body burden declines little from about 25% of dose at 30 days to about 20% at 164 days. The amount found in the gastrointestinal tract is surprisingly stable. About 8% of dose is found in the gastro-intestinal tract at 30, 64, 94, and 164 days. The distribution at 164 days within the gastro-intestinal tract, on the average for ten fish, was 1.6, 1.6, 0.5, and 0.5 µc for stomach, pyloric caeca, mid-gut and hind-gut, respectively.

For further study, a group of 100 trout were administered a single oral dose of 200 µc each. These fish are to be utilized for more detailed studies on the deposition of Zn^{65}, in particular in the gastro-intestinal tract and gill filaments.

Strontium

The weekly massive bleeding of five miniature swine on the study to determine the reserve capacity of the hematopoietic system was discontinued. During this first phase of the study, the animals were bled eight times at weekly intervals with a total volume of blood equal to 8% of the animal's weight removed. Preliminary analysis of the results indicates that of the parameters studied no marked differences were observed between the control animals and those fed 125 µc Sr^{90}/day. (This level of Sr^{90} feeding results in an average radiation dose rate to the bone marrow of ~15 rads/day.) As expected, a marked increase in the rate of erythrocyte production was observed as measured by rate of plasma clearance of Fe^{59} and erythrocyte uptake of Fe^{59}. The red cell count and hemoglobin content of blood decreased slightly in all animals during the study. Due to an increase in red cell volume, the packed cell volume of the animals was only slightly depressed. Marked increases in the reticulocyte counts were observed in all animals.

(In a continuation of this study after blood values have returned to normal, the animals will be subjected to several massive bleedings at short intervals and their response to this acute blood loss determined as contrasted to the chronic blood loss in the first phase.)

Iodine

Five of 17 ewes given 3 mc I^{131} five years ago failed to conceive or experienced early death of the embryo. All the control ewes conceived at the first mating and had an uncomplicated pregnancy. The gestation period in the experimental group was 146 $\pm$ 4.4 days, while that in the control group was 143 $\pm$ 2.7 days. There was no difference in the birth weight of the lambs; however, in the animals sacrificed at birth the lamb thyroids of the experimental group were larger than the controls, suggesting a compensatory hyperplasia. There was a suggestion of an increased thyroid uptake of a tracer dose of I^{131} in the lambs born in the experimental group.

After 45 days of feeding 2 g of stable iodine to a cow on a regimen of 5 µc of I^{131}/day, the I^{131} concentration in the milk was reduced over 50% and the I131 uptake in the cow's thyroid was reduced about 95%. (The calculated stable iodine requirement for a cow is 5 mg/day.)

Plutonium

Data on plutonium tissue concentrations have been obtained over a 17-day period following injection of Pu^{238} or Pu^{239}. This study will ultimately extend to 64 days post-injection and will include hematological and histopatholcgical studies. As indicated in earlier preliminary studies, Pu^{239} deposition in liver, spleen, and adrenal gland is 2 to 3 times greater than Pu^{238} deposition, while the deposition of Pu^{238} in bone is greater than that of Pu^{239}. Addition of a large citrate excess did not appear to affect the distribution of Pu^{239}.

Copper, Radiation and Molybdenum Inter-relationships

Further results from the paired feeding study comparing the effects of high copper and molybdenum intakes on irradiated and non-irradiated rats support the earlier indications that the decrease in urinary copper excretion following 850 r total-body irradiation is due to a decreased food intake subsequent to irradiation. Liver copper concentrations are increased on a high copper diet in both irradiated and non-irradiated rats. Of perhaps greater interest is the observation of increased liver copper concentrations in animals on a high molybdenum diet.

Inhalation Studies

One dog died 1200 days after a single inhalation exposure to $Pu^{239}O_2$. Body burden at time of death was 2.1 µc, distributed as follows: lungs, 49%; bronchial and mediastinal lymph nodes, 23%; muscle, 3%; liver, 5%; bone, 18%; pancreas, 0.8%; heart, 0.4%; and spleen, 0.3%. The bronchial lymph nodes had the highest concentration of plutonium, 2.4 µc/g, compared with 0.005 µc/g in lung, and 0.0005 µc/g in bone. However, the 500-fold difference in concentration of Pu239 in bronchial lymph nodes compared with lung was attributed in part to a 3-fold increase in lung weight due to patholcgy, and at least a 2-fold decrease in lymph node weight due to atrophy.

One dog died eight months after depositing about 2 mc $Ce^{144}O_2$. Total radioactivity in the dog at death was about 1 mc divided between the lungs, bronchial lymph nodes, and liver. Radiochemical analyses of the tissues are not complete.

Six beagle dogs were flown to the Nevada Test Site and exposed to plutonium released from a one-point detonation of a nuclear weapon. The dogs will be studied for comparison of the excretion, tissue distribution, and biological effects of plutonium with results obtained in laboratory studies.

The distribution of inhaled PuO_2 in the sub-cellular fractions of lung tissue was determined in rats four months after exposure, Table 1.

TABLE 1

		Percent
1.	Saline washings of lungs	2
2.	Fibrous tissue	24
3.	Sub-cellular fractions	
	Nuclear and cell debris fraction	72
	Mitochondrical	1.4
	Microsomal	0.5
	Supernatant	0.4

It is thought that most of the Pu^{239} found in the nuclear and cell debris fraction was associated with the cell debris. However, this will be examined in further experiments. Knowledge of the cellular and sub-cellular distribution of Pu^{239} in the lung is expected to contribute towards developing methods for the removal of inhaled plutonium.

Two experiments were completed testing agents for removal of inhaled $Pu^{239}O_2$ in rats. In one experiment, DTPA and Pluronics by aerosol, $NaHSO_4$ + KI orally and intraperitoneally, and T.R. aerosol (a wetting agent produced by American Cyanamid) were ineffective. In another experiment, CO_2, Alevaire, Isuprel, and NH_4Cl were tested alone and in combinations. Only in Isuprel-treated rats was the lung burden significantly less ("Student t" test at 95% confidence limits) than the controls. Further tests are in progress.

Plasma levels of 17-hydroxycorticosteroids in several dogs exposed three years ago to PuO_2 did not differ significantly from the controls. It was expected that the lymphopenia occurring in animals after inhalation of plutonium dioxide might be associated with increased levels of the 17-hydroxycorticosteroids, thus possibly linking the adrenal gland to this apparently abscopal effect of radiation. These results suggest that a fresh approach to the problem is in order.

Milk Studies

Single tracer doses of K^{42}, Rb^{86}, Cs^{137} and Ba^{140} were given intravenously to three lactating Suffolk ewes in order to determine the comparative metabolic behavior of these elements with regard to plasma clearance and milk transfer. Blood and milk samples were collected at frequent intervals

1103572

for several weeks. Complete analytical results are not available at this time for all four isotopes. Preliminary results indicate the peak milk concentrations of both Rb^{86} and Cs^{137} occurred at about 7 hrs after administration of the nuclides. The milk-to-plasma ratio for both Rb^{86} and Cs^{137} at steady state was 10 to 15.

Gastro-intestinal Radiation Injury

Further studies on the protection of the irradiated GI tract by infusion of the bile salt sequestrant, cholestyramine, have not resulted in appreciably improved survival, even with supplementary antibiotic and fluid supportive therapy. Studies with C^{14}-labeled bile salts indicate that cholestyramine is only partially effective in removing bile salts from the intestinal lumen. In another approach to this problem, rats have been surgically modified in such a manner that the bile drains into the urinary bladder. This procedure is apparently effective in preventing the diarrhea which normally follows irradition. Insufficient data are available to evaluate its possible effect on survival.

Preliminary studies of histological preparations from rats pretreated with cysteine and exposed to cumulative intestinal radiation doses as high as 20,000 r seem to indicate that the intestine is able to undergo very extensive damage and subsequent repair of the mucosa without permanently impairing functional activity.

Secondary Disease Studies

Rat anti-mouse and mouse anti-rat antibodies from radiation chimeras have been separated by agar gel electrophoresis techniques. The mouse anti-rat antibody was associated with the albumin fraction, while the rat anti-mouse antibody moves more slowly with the gamma globulin fraction. Both antibodies were also detectable in the fraction which did not move from the origin. Mixtures of separated antibody fractions were re-separated electrophoretically without difficulty. There is some loss of antibody activity upon storage at freezer temperatures, the rat anti-mouse antibody appearing to be more sensitive to destruction during storage. These results support the thesis that the two antibody species are different and not simply a non-specific antibody capable of agglutinating both rat and mouse erythrocytes.

Studies with radiation chimeras derived from two different mouse strains (C31 and LAF) unlike the rat-mouse chimeras, have not shown serum hemagglutinins against either host or donor. There has also been no consistent improvement in survival of the radiation chimeras when the donor tissues are taken from animals pre-treated with host tissues.

A start has been made on a histochemical and cytochemical study of the development of secondary diseases in rat-mouse chimeras.

Microbiology

The phospholipid content of irradiated cells is higher, relative to the phosphorus content of the cell, than in unirradiated controls. The data are suggestive that leakage of phosphate as a consequence of irradiation is linked with phospholipids in the cell membrane.

Potassium uptake by strictly oxidative yeast was demonstrated to occur at about 1/20th the rate in oxidative-fermentative strains. The uptake is pH dependent with the maximum occurring between pH 7 and 8. Whether the low potassium uptake is related to reduced H^+ availability or to reduced permeability of the membrane to H^+ of K^+ is not yet clear.

Radiation Effects on Insects

Statistical analyses became available from a neutron radiation experiment performed last winter. Three-week-old virgin flour beetles, *Tribolium castaneum*, were given 980 rads of fast neutrons (average energy 4.6 Mev) from the $H^2(d,n)He^3$ reaction. The radiated beetles were mated in various combinations, such that the effects on the male or female reproductive ability could be distinguished. The number of adult F_1 progeny was the measure of the parental reproductive ability. The neutron exposure reduced the reproductive ability in all cases, the greatest being when both sexes were radiated. Females were more fit reproductively than males when only one sex was radiated. No differential radiation response of germ cells was observed. Cytological examination of female reproductive organs reflected the reduced reproductive ability due to the neutron exposure.

A more elaborate neutron effects study on *T. castaneum* was established at three temperatures with day-old adults. The same neutron exposure conditions were used to give approximately 850 rads. To determine the onset of reproduction a series of crosses using combinations of young and old males and females was conducted at one temperature. This initial test will be extended to other temperatures. The data are being analyzed.

Plant Nutrition

Bacterial infection of roots in nutrient culture prevented the carrying out of successful experiments on the mechanism of ion uptake. The source of the infection is not known.

The efficiency of iodide uptake from nutrient solution was increased about threefold by increasing the concentration of iodide in the substrate from carrier-free levels of I^{131} up to 1 µg I/ml of nutrient solution. Although distinct toxicity was noted at the 1.0 µg/ml level, no such toxicity was observed at the 0.1 or 0.01 µg/ml levels and all showed enhanced uptake.

Plant Ecology

Cheat grass shoots harvested in a greasewood community in late April and assayed for calcium, magnesium, sodium, and potassium showed the cheatgrass growing beneath the canopy of greasewood to contain twice as much sodium as those plants harvested on the adjacent areas between shrubs. The increased sodium content of cheatgrass growing beneath greasewood appears to result from the decay of greasewood leaves which are high in sodium and a subsequent uptake by the grass.

Rattlesnake Springs Limnology

Twenty mc of Rb^{86} were introduced as a slug into the upper end of Rattlesnake Springs. Water samples were collected at 2-minute intervals at six stations between the point of introduction and the upper dam. Preliminary results show that the activity peak decreased and flattened out at successive downstream stations. No pulse was found at the dam. Cursory examination of the counts from bottom deposit samples showed that the substrate adsorbed the isotope, although the magnitude of this is still being analyzed.

Fallout

Samples of water and particulate matter from the Tucannon River and Blue Lake were collected and are being analyzed for fallout isotopes. Preliminary reports show the presence of significant amounts of zirconium and niobium in the particulate matter.

Sampling and Analysis of Natural Populations

Study of "distance" sampling methods was continued, with a small-scale field test on herbaceous plants in the understory of sagebrush and greasewood plots at Rattlesnake Springs.

Two mule deer were collected on the Wooten Game Range for radioanalysis, and a series of sampling points established.

A study of the statistical problems associated with sampling for radionuclides was continued, with some analysis of Alaska data initiated, and study of computational methods.

HA Kornberg

Manager
BIOLOGY LABORATORY

HA Kornberg:es

TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORY

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposiums

Hanson, W. C. Radioecological studies of northern Alaska. Third Symposium on Radioactivity in Scandinavia, University of Lund, Lund, Sweden. May 21, 1963.

Hanson, W. C. Dietary sources of Cs^{137} in Alaskan Eskimos. Third Symposium on Radioactivity in Scandinavia, University of Lund, Lund, Sweden. May 21, 1963.

Park, J. F. Chronic toxicity of inhaled plutonium in dogs. American Industrial Hygiene Association and Society of Toxicology, Cincinnati, Ohio. May 6-10, 1963.

Ballou, J. E. Comparative toxicity of Pu^{239} and Pu^{238}. Radiation Research Society, Milwaukee, Wisconsin. May 27-29, 1963.

McClellan, R. O. Changes in calcium-strontium metabolism with age in young miniature swine. Radiation Research Society Meeting, Milwaukee, Wisconsin. May 27-29, 1963.

McClellan, R. O. Iodine-131 in the feed, thyroid and milk of dairy cows. Radiation Research Society Meeting, Milwaukee, Wisconsin. May 27-29, 1963.

Rhyneer, G. S. The effects of inhaled $Pu^{239}O_2$ in adrenalectomized rats. (Co-authors: H. W. Casey and W. J. Bair.). 11th Annual Radiation Research Society, Milwaukee, Wisconsin. May 27-29, 1963.

b. Seminars (Off-Site and Local)

Uyeki, E. M. Graft-host response in radiation chimeras. University of Washington, Seattle, Washington. May 2, 1963.

c. Seminars (Biology)

Watson, E. C., Radiation Protection Operation, Hanford Laboratories. Remarks on the Reactor Siting Criteria Sympsoium held in Bombay, India. May 2, 1963.

McClanahan, B. J. The use of element pairs in ion transport. May 7, 1963.

Wood, D. I^{131} studies in cattle. May 7, 1963.

Kimeldorf, D. J., Head, Physiology-Physcology Branch, USNRDL, San Francisco, California. Some prompt reactions to radiation exposure. May 17, 1963.

Ragan, H. A. Clinical studies in irradiated animals. May 21, 1963.

Hackett, P. L. Radiation and copper toxicities. May 21, 1963.

Entenman, C., Director, Institute for Lipid Research, Berkeley, California. Some studies on lipid metabolism in the X-irradiated animals. May 24, 1963.

d. Miscellaneous

Uyeki, E. M. Modification of radiation injury in mammals. Radiobiology Course, University of Washington Graduate Center, Richland, Washington. May 8, 1963.

Bair, W. J. Inhalation problems. University of Washington, Richland, Washington. May 22, 1963.

II. Articles Published

a. HW Documents - None

b. Open Literature

Bair, W. J. and D. H. Willard. 1963. Plutonium inhalation studies. III. Effect of particle size and total dose on deposition, retention, and translocation. Health Physics 9: 253-266.

Hanson, W. C., F. W. Whicker, and A. H. Dahl. 1963. Iodine-131 in the thyroids of North American deer and caribou: comparison after nuclear tests. Science 140: 801-2.

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.)

Tombropoulos, E. G., W. J. Bair, and J. F. Park. 1963. Effect of diethylenetriamine-pentaacetic acid and polypropylenoglycolethylene oxide polymer on excretion of inhaled $^{239}PuO_2$ in dogs. Nature 198: 703-704.

III. Visits and Visitors

a. Visits to Hanford

5/7-8/63	John Neff and Fred Beem, Acme Refrigeration Co. Check grwoth chamber (contacted J.F. Cline).
5/7	Dr. H. C. Monroe and G. C. Loud, Davison Chemical Company, Irwin, Tennessee. Visit facilities and discuss research with V. G. Horstman.
5/10	Dr. Charles Sachs, French AEC, Fontenay-aux-Roses. Discuss internal decontamination and Pu poisoning with Biology staff.

5/10/63 Northwest Association of Occupational Medicine. Tour Biology.

5/13/63 Governor Rosellini, R. E. Rose, etc. Olympia, Washington. Tour.

5/13/63 Columbia Basin College Zoology Students, Pasco. Tour.

5/17/63 Dr. D. Kimeldorf, USNRDL, San Francisco. Tour and present seminar.

5/20/63 Dr. R. Daubenmire, Washington State University, Pullman. Check field plots with W. Rickard.

5/24/63 Dr. Entenman, Institute of Lipid Research, San Francisco, Calif. Tour and present seminar.

5/24/63 Chemistry students from Zillah High School, Zillah, Wash.

b. Visits Off-Site

5/5-8/63 V. H. Smith attended the International Symposium on Physical Processes in Minneapolis, Minnesota.

5/5-8 and 5/16-17 C. E. Breckinridge escorted experimental dogs to and from the Nevada Test Site for Project Roller Coaster.

5/5-10 R. T. O'Brien attended the American Microbiological Society in Cleveland, Ohio.

5/6-10 J. F. Park presented a paper at the American Industrial Hygiene Conference and Society of Toxicology in Cincinnati, Ohio, and discussed cardiology studies at the College of Veterinary Medicine in Columbus, Ohio.

5/14-16 L. Haverfield visited Washington State University in Pullman to consult for taxonomic purposes with Dr. M. T. James.

5/14-15 P. A. Olson and C. O'Malley picked up silver salmon at the Eagle Creek Federal Fish Hatchery, Estacada, Oregon.

5/15-16 L. L. Eberhardt, W. H. Rickard, and C. E. Cushing collected specimens at the Wooten Game Range, Dayton, Washington.

5/17/63 D. D. Mahlum consulted on research with Drs. Wiese and LeTourneau at the University of Idaho, Moscow, Idaho.

5/17-6/8 W. C. Hanson presented two invited papers at the Third Symposium on Radioactivity in Scandinavia, University of Lund, Lund, Sweden on May 20-21; visited facilities in Stockholm, Oslo, Copenhagen, and Windscale Works in Seascale, England. Attended a special meeting in Washington, D.C. on June 7.

5/24/63 L. K. Bustad worked on the ionizing radiation study at the University of Washington with Professor Ruch.

b. Visits Off-Site (Continued)

5/21/63 M. E. Kerr visited Pendleton Grain Growers in Pendleton, Oregon to inspect ½" pellets.

5/26-31 R. O. McClellan presented a paper at the Radiation Research Society Meeting in Milwaukee, Wisconsin and discussed research with J. L. Felts at the University of Minnesota.

5/26-30 Glenda S. Vogt attended the Radiation Research Society Meeting in Milwaukee, Wisconsin.

5/26-29 R. F. Palmer and A. C. Case visited Lovelace Foundation in Albuquerque, New Mexico and Los Alamos Scientific Laboratory in Los Alamos, N.M. to confer on radiation measuring methods and facilities with Drs. R. G. Thomas and C. R. Richmond, respectively.

5/27-29 B. O. Stuart attended the Radiation Research Society Meeting in Milwaukee.

5/27-29 J. E. Ballou presented a paper at the Radiation Research Society Meeting in Milwaukee.

IV. Achievements

No degrees were earned, nor did any professional licensing or certification occur.

V. Honors and Recognitions

Dr. M. F. Sullivan was elected a Fellow in the American Association for the Advancement of Science.

VI. Professional Group or Organization Assignments

None

APPLIED MATHEMATICS OPERATION

MONTHLY REPORT - MAY, 1963

ORGANIZATION AND PERSONNEL

Linda S. Dean transferred into the group from Technical Information, effective May 20, 1963.

OPERATIONS RESEARCH ACTIVITIES

Reactor Simulation Studies are continuing, with work to develop data recording on a consistent and nonredundant basis. Recording forms have been revised to eliminate such problems.

The HAPO - Tri-City Model is being pushed for completion, in crude form, by the end of June. Considerable data difficulties cause delays. Ways are being sought to overcome data gaps by developing assumptions that can be, at least, tested for logical consistency and for consistency with the highly aggregated data which is available.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

N-Reactor Department

Parameter estimation techniques in connection with the weld crack problem associated with NPR primary piping have been successfully programmed. Estimation of all parameters is virtually complete, and a revised document setting forth the parameter estimates in tabular form is soon to be issued.

Irradiation Processing Department

A large number of metallographic variables from the second end-bonding experiment have been analyzed. The objective is to determine those canning conditions which will give the best fuel elements by understanding the effects that time, temperature, and pressure have on the important fuel element characteristics. A discussion was also held on a proposed system for controlling the quality of fuel elements manufactured by this process by controlling the quality of incoming material and the manufacturing process itself. Emphasis was laid on the need to correlate preirradiation manufacturing conditions and fuel element characteristics with postirradiation results.

Assistance was given in determining a plan to identify ingots of uranium as being normal or enriched on the basis of counting rates based on γ emission. A counting time and number were determined such that the two ingot types can be identified with specified and very high levels of assurance.

A discussion was held on the interpretation of the results of a test to compare Alcoa and Cliff canning components based on UT-4 measurements.

A discussion was held on the sample size necessary to determine if there are adverse side effects from a corrosion inhibitor applied to fuel elements.

Analyses were made on the data from a salary study to see whether there are different relative salary trends for the different salary levels.

A functional relationship was fitted to counting data in which one of the parameters should estimate the half life of thorium. However, the resulting estimates were poor, no doubt due to the restricted range of time wherein the counts were made.

Continued assistance is being given to determine the best conditions for reprocessing substandard rails for self-supported KVE and KVN fuel elements. In support of this objective, data were analyzed to determine the precisions and relative biases of several measuring instruments. Related work is being done here on determining how ellipticity of fuel elements affects the minimum annuli.

Relationships between the outlet water temperatures around the circumference of a tube and the thermocouple locations have been obtained for several tubes. This is a continuing program and is concerned with the determination of the temperature imbalance in the outlet water.

An estimate was obtained of the proportion of alpha particles emitted near the surface of a fuel element which reach the surface.

Analyses on the mathematical theory, use, error propagation, and EDPM program for both the optical and electronic process tube traverse mechanisms have been completed and recommendations for their improvement and use discussed with the customer.

<u>Chemical Processing Department</u>

The continued high plutonium content and relatively low process and measurement variation permitted demonstration of compliance with minimum purity specification for the first quarter of CY-1963. Due to the wide margin between demonstrated compliance and specification, the number of random samples selected per week for Pu assay was decreased from 40 to 10. The measurements of the minor component observed to data indicate a part by part (100 percent inspection) will be required.

[redacted] [redacted]

The estimation of U-235 content of UO_3 shipments will be made from duplicate measurements, using the 234-5 laboratories thermionic emission mass spectrograph, on each car shipped. At the same time, a UO_3 standard of known isotopic content will be analyzed to permit subsequent estimation of absolute reliability.

A limited examination of silicon content of feed material and number of casting rejects indicated a statistically significant relationship.

A comparision of the observed variation at five counting levels indicated the revised ASP counting instruments could be used to replace standard ASP instruments at all counting levels.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HANFORD LABORATORIES

2000 Program

The EDPM program which simulates the NPR stack gas composition was modified in such a manner so as to be applicable over a much wider range of input parameters. This revision makes it possible to incorporate immediately any new or revised reaction coefficients as fast as they are obtained from laboratory experiments. Moreover, in its new form, the program is now applicable to other reactors.

Analysis continues on the organic volume fraction profiles being developed from pulse column data. Data relating isotopic activity to location in the tube have been analyzed to estimate the total activity in the tube. The data are also being fitted with a harmonic-type function to see what general characteristics as regards their cosine, skewness, and peakedness effects exist.

3000 Program

A series of conferences has led to a firm plan to convert a Sheffield rotary contour gauge to numerical control. Appropriate mathematical relationships have been derived and a preliminary EDPM program is being written to prepare magnetic tape input for the system.

Work on the design of shear-spinning blanks continued following the initial successes at moderate shears. In particular, designs have been completed which will allow experimentation in the range of greatly-increased shear coefficients.

4000 Program

Routine calculations were performed to determine optimum particle size and proportionate mix factors for several cylindrical Vipac fuel elements. Considerable effort has also been devoted to the study of particle spiraling, a phenomenon suspected of playing a fundamental role in the vibrational compactification of annular-shaped fuel elements. Although the knowledge in this area is somewhat fragmented, a documentation of the state of the art is currently being compiled.

Limited progress has been made on the theoretical study of sonic transmission in elastic media. In particular, a method of interpreting the phenomenon of viscous damping (energy absorption) in the so called Voigt solid has been discovered.

A method of calculating with modified Bessel functions of large index was developed which has the advantage of exhibiting numerical stability on electronic computers. The problem arose in a study concerned with heat transfer from hot spots on fuel elements.

Modification of the machine language program which calculates void fractions and densities in uranium was accomplished, and routine analyses were processed.

5000 Program

During the month an improved method of using background spectra was incorporated in the calculational part of the IRA system. This modification gives significantly better results.

Further changes are being made to the IRA system to reduce computer times, and to handle modifications made in the storage and reporting of data.

Final arrangements were made to document the program for indexing X-ray diffraction powder patterns in the hexagonal case. A report will be issued early in June. A new counting routine for the orthorhombic program has been written and compiled but several errors remain. Work is continuing on this program. A new "master" program called INDEX was written, compiled, debugged, and successfully used. INDEX makes a subroutine out of the cubic program. Work is under way to add the hexagonal-tetragonal program to INDEX as a subroutine. Eventually an orthorhombic subroutine will be added so that INDEX will handle any of the four types of crystal.

6000 Program

Statistical analysis of data on adult progeny from neutron exposed *Tribolium castaneum* was completed, and a letter containing the results was prepared.

DECLASSIFIED

GENERAL

Special support was given in the design of extensive theory - experiment correlation program.

Design criteria included the ability to estimate simultaneously the theory - theory agreement between calculational methods employed in the program as well as the theory - experiment limitations of the methods.

Service is being provided to ROHO for its semimonthly employee survey, and to CPD for a special employee survey of its own.

In discussion with radiation protection personnel a basis was given for choosing at what film density value true radiation exposure is indicated.

Other

On Monday, May 27, the Chemical Research Seminar was given by J. B. Goebel, Applied Mathematics. The seminar was entitled "Indexing X-ray Diffraction Powder Patterns, Case II, Hexagonal-tetragonal Case".

Manager
Applied Mathematics

CA Bennett:dgl

[redacted]

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Comparison of Cm-244 Production Rates Using Th-232 and U-238 as Fertile Fuels

Two appealing methods of producing Cm-244 in power reactors are plutonium enrichment of either thorium or depleted uranium. Both systems exhibit low fuel cost even without a large Cm-244 credit. Fuel elements containing these materials should be less difficult to fabricate than complex composites which isolate plutonium of different ages.

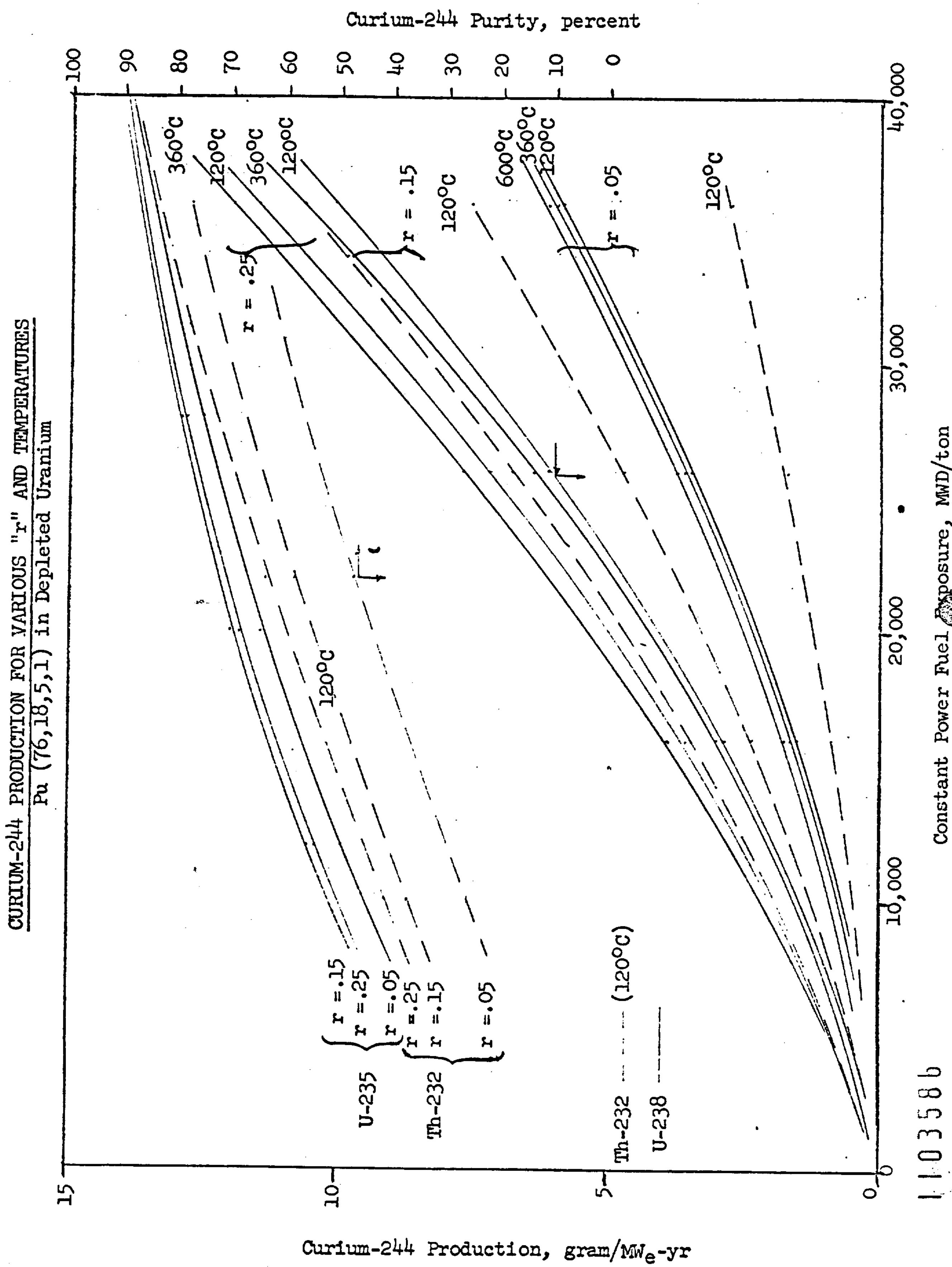
Curium-244 production rates using plutonium (76% Pu-239, 18% Pu-240, 5% Pu-241, 1% Pu-242) enrichment of thorium were discussed in the April 1963 Monthly Report. Now data are available to make a comparison of the production using the same plutonium composition enriching both U-238 and Th-232.

The major result appears to be that at a fuel specific power of 20 MW/T fertile, plutonium in U-238 will have a higher Cm-244 production rate and, also, a higher Cm-244 purity (Cm-242 is formed along with Cm-244 which detracts from the value of Cm-244) is obtained when U-238 is used as the fertile material. This advantage decreases as the neutron spectrum becomes harder. At different specific powers the relative advantages may change. This will be analyzed.

The better purity obtained using U-238 as a fertile fuel in the 20 MW/T example can be amplified further by physically separating the initial plutonium and U-238 in the fuel element to prevent new Pu-241 from forming more Cm-242 in the initial plutonium. The total production of Cm-244 would be unaffected, but the purity of that formed in the part of the fuel containing U-238 would be much lower.

Figure 1 is a plot of Cm-244 production versus fuel exposure for various spectral indexes and temperatures with plutonium enrichment of depleted uranium. The production, as shown in this plot, increases with both increased moderator temperature and spectral index, r. Higher values of r indicate harder neutron spectra. The purity of Cm-244 (Cm-244/Cm-244 + Cm-242) also increases withfuel exposure and spectral index. The production rate in thorium is designated by dashed lines for 120 C neutron temperature for comparison.

FIGURE 1



1103586

A comparison of these production rates in U-238 with those in thorium for the same plutonium composition (76, 18, 5, 1) shows the production rate of Cm-244 to be much higher in U-238, by about a factor of 2 at low r's. This is due to the U-238 in the fuel which supplies new plutonium atoms for eventual formation of more Cm-244. It is also noted that the purity has increased in the U-238 fuel. This is due to a higher flux in the U-238 fuel for enrichments which yield the same exposure. The absorption cross section for thorium is higher than for U-238 -- especially at low r's. Thus, the U-238 fuel needs less enrichment to attain the same exposure and, hence, a higher flux to achieve the same specific power. It is for this reason that the relative advantage of U-238 system may change with the specific power. The U-238 fast effect also helps to reduce the necessary fissile content. The higher flux means that the Pu-241 will have less time to decay (Pu-241 $\xrightarrow{\beta}$ Am-241 + n $\longrightarrow$ Cm-242) relative to the rate of Cm-244 production (Pu-242 + n $\longrightarrow$ Am-243 + n $\longrightarrow$ Cm-244).

General Fuel Cycle Analysis

Calculations of additional fission products from water moderated reactors using slightly enriched uranium are tabulated in Tables I and II. These calculations were made at 10 MW/T average fuel specific power.

It should be noted that Ru-103 production is less at higher exposures due to the saturation effect (i.e., the rate of decay is equal to the rate of production). This same effect reduces the production of Ru-106 at exposures above 10,000 MWD/T, but for lower exposures, the increase is due to the higher fission yield from plutonium than from uranium. The higher yields from plutonium also cause increases in stable ruthenium and palladium production. Rhodium production is low at very low exposures due to the accumulation of the Ru-103 parent inventory as well as the lower yield from uranium fission. At high exposures, rhodium tends to saturate due to its rather high cross section. Krypton production decreases at high exposure due to lower fission yield of plutonium as well as the saturation tendency of Kr-85.

Production of xenon and cesium has also been computed and typical data are summarized in Table III (trace quantities of Xe-129 and fractional percentages of Xe-130 have been neglected).

Effective Values of the Neutron Reproduction Factor

Values of $\overline{\eta}$, the effective neutron reproduction factor, for Pu-239 have been cited in the literature that compare unfavorably with those of U-235 and U-233. A consistent set of $\overline{\eta}$ values for these three isotopes as a function of their concentration in a water moderator was calculated therefore. The computations were made by combining the SPECTRUM code (for energies up to 2.38 ev) and the GAM code (for energies above 2.38 ev). The results for a homogeneous mixture of fuel and room temperature water are shown in Table IV.

TABLE I

ISOTOPES - GRAMS/MW_t YEAR AT DISCHARGE

Exposure MWD/T	Stable Ru	40 d Ru-103	1.0 yr Ru-106	Stable Rh-103	Stable Pd	Stable Kr	10.4 yr Kr-85
1,926	20.0	1.70	1.20	4.0	3.8	4.7	0.40
3,926	20.6	0.93	1.33	4.7	5.1	4.5	0.37
5,926	21.1	0.65	1.41	5.1	6.4	4.3	0.35
9,926	21.9	0.42	1.43	5.2	8.6	4.0	0.31
19,926	23.7	0.24	1.31	4.7	13.6	3.3	0.23
39,778	26.3	0.14	0.95	3.3	21.0	2.4	0.13

TABLE II

ISOTOPES - WATTS/MW_t YEAR 150 DAYS AFTER DISCHARGE

Exposure MWD/T	40 d Ru-103	1.0 yr Ru-106	10.4 yr Kr-85
1,926	8.2	28.0	0.28
3,926	4.5	31.0	0.26
5,726	3.1	33.0	0.25
9,926	2.0	33.0	0.22
19,926	1.2	30.0	0.16
39,778	0.7	23.0	0.09

TABLE III

ISOTOPE PRODUCTION IN GRAMS/MW-YR

Power MW/T	Exposure MWD/T	Xe-131 Stable	Xe-132 Stable	Xe-134 Stable	Xe-136 Stable	Cs-133 Stable	Cs-134 2.1 Yr	Cs-135 2x10^6 Yr	Cs-137 30 Yr
5	2,000	4.9	7.6	12.8	16.8	10.6	0.04	4.5	10.0
5	6,000	5.0	7.9	12.2	16.3	10.2	0.09	4.5	9.7
5	10,000	4.9	8.2	11.7	16.1	9.8	0.13	4.4	9.3
5	22,000	4.5	8.9	10.7	15.7	9.0	0.20	4.2	8.2
5	41,000	3.7	10.0	9.8	15.5	8.2	0.24	3.8	7.1
10	2,000	4.9	7.6	12.8	18.4	10.6	0.05	2.9	10.1
10	6,000	5.0	7.9	12.2	17.9	10.2	0.11	2.9	9.9
10	10,000	4.9	8.2	11.7	17.7	9.8	0.16	2.8	9.6
10	22,000	4.5	8.9	10.7	17.3	9.0	0.29	2.6	8.8
10	41,000	3.7	10.0	9.8	17.0	8.2	0.38	2.4	8.0
20	2,000	4.9	7.6	12.8	19.7	10.6	0.05	1.7	10.2
20	6,000	5.0	7.9	12.2	19.1	10.2	0.12	1.7	10.0
20	10,000	4.9	8.2	11.7	18.9	9.8	0.19	1.6	9.7
20	22,000	4.5	8.9	10.7	18.4	9.0	0.36	1.5	9.1
20	41,000	3.7	10.0	9.8	18.1	8.2	0.52	1.5	8.5

CESIUM HEAT GENERATION, WATTS/MW-YR (150 days after discharge)

Exposure MWD/T	5 MW/T		10 MW/T		20 MW/T	
	Cs-134	Cs-137	Cs-134	Cs-137	Cs-134	Cs-137
2,000	0.25	4.2	0.26	4.2	0.27	4.2
6,000	0.56	4.0	0.65	4.1	0.71	4.1
10,000	0.77	3.9	0.96	4.0	1.10	4.0
22,000	1.08	3.4	1.71	3.6	2.12	3.8
41,000	1.41	3.0	2.23	3.3	3.06	3.5

TABLE IV

VALUES OF $\bar{\eta}$ FOR U-233, U-235, AND Pu-239

IN A WATER MODERATOR AT 293°K

Fissile Atoms Per Atom Hydrogen	$\bar{\eta}$, Neutrons Produced Per Neutron Absorbed U-233	U-235	Pu-239
0.0	2.29	2.06	2.08
0.005	2.26	1.99	1.93
0.010	2.24	1.93	1.89
0.20	2.21	1.84	1.85

The method used to obtain the data of Table IV is somewhat unsatisfactory in that the effect of geometric self-shielding could not be evaluated directly. These results, therefore, are applicable only to a homogeneous mixture of the fuel, moderator, and cladding. Also, a somewhat arbitrary weighting between the GAM and SPECTRUM results must be introduced in the computation, and this factor may have a considerable bearing on the results of the study.

A comparison between the GAM-SPECTRUM results for 500°K moderator and some published(1) MUFT-SOFOCATE results are shown in Table V.

TABLE V

A COMPARISON OF $\bar{\eta}$ FOR Pu-239 IN A WATER MODERATOR AT 500°K CALCULATED BY TWO DIFFERENT COMPUTATIONAL METHODS

Fissile Atoms per Atom Hydrogen	$\bar{\eta}$, Neutrons Produced per Neutron Absorbed GAM-SPECTRUM	MUFT-SOFOCATE(1)
0.00	2.05	1.88
0.005	1.89	1.77
0.010	1.87	1.75

(1) These $\bar{\eta}$ values are based upon the alpha values presented by Fein and Noderer, CEND-146.

Comparison of $\bar{\eta}$ between systems involving U-238 and thorium are misleading unless ϵ, the fertile fast effect, is included (i.e., $\bar{\eta}\epsilon$). This is extremely significant because the ϵ value for thorium is essentially 1, while for U-238 systems it can be 1.1 (1.2 theoretical limit for uranium metal) and is 1.06 for some H_2O moderated UO_2 fueled machines. With an ϵ value of 1.06, the $\epsilon\ \eta$ products for U-235 and Pu-239 corresponding to the $\bar{\eta}$ values in Table IV range from 2.18 to 1.95 and 2.21 to 1.96, respectively. For U-233 in thorium, $\bar{\eta}$ and $\bar{\eta}\epsilon$ are essentially the same.

Fuel Conservation Calculations

An index of reactor performance that often appears in a discussion of the conservation of fissile material is the heat produced per gram of U-235 made unavailable. An atom of U-235 is considered to be "unavailable" when it either absorbs a neutron during the burnup or is diluted by U-238 in the discharged fuel elements or in the tailings of the diffusion cascade. A formula for this index can be written in terms of reactor and diffusion cascade parameters as

$$\psi = \frac{kE}{X}\left(\frac{1 - X_o/X_f}{1 - X_o/X}\right) \tag{1}$$

where

ψ = megawatt-day per gram U-235 made unavailable

E = reactor exposure, MWD/T

X = initial enrichment of the reactor, gram U-235/gram uranium

X_o = cascade tails composition, gram U-235/gram uranium

X_f = cascade feed composition, gram U-235/gram uranium

k = 1.102×10^{-6} ton/gram (constant)

The bracketed quantity in Equation (1) represents the grams of U-235 in the enriched product of the diffusion cascade per gram of U-235 in the feed (e.g., natural uranium). (Equation (1) is derived from mass balances.) Note that no account is taken of the plutonium bred during the burnup in this formulation.

It is interesting to compare the fuel conservation index, ψ, with the total fuel cost for the burnup. Table VI shows the reactor performance data obtained with the MELEAGER and QUICK codes for a water-moderated zirconium-clad graded-irradiation reactor simulation. The graded cycle used is theoretical with no control rod losses. (Burnable poison batch cycles are also being investigated.) Standard economic charges are assumed. Also shown is the net fuel conservation index, ψ net = $\psi/(1-c)$, where c is a production conversion ratio, grams fissile plutonium discharged per gram U-235 in natural uranium.*

The most important conclusion that can be drawn from the data of Table VI is that, while ψ increases continually with enrichment**, there is an intermediate enrichment for which the total fuel cost and the net fuel conservation index are extremes. Thus, the minimum fuel cost approach to fuel cycle calculations tends to maximize the heat produced per gram of the natural resource; i.e., the U-235 contained in natural uranium. This may not be true for plutonium prices other than the fuel value used here. Note that for comparison with the values shown in Tables VI and VII, approximately one megawatt-day of heat is produced by the fissioning of one gram of U-235.

The fact that the minimum fuel cost tends to give the maximum utilization of fissile material means that this approach may be synonymous with conservation of the natural resource. That is, if the present pricing scheme remains in effect, increased demand for supply of natural uranium will lead to an increase in the price of the feed to the diffusion cascade, which lowers the tails composition from the cascade***. This will, in turn, both decrease the initial enrichment for a given reactor and, therefore, increase the conservation index. This effect is shown in Table VII.

The data of Table VII show that, up to a point, increasing the price of natural uranium, increases the utilization of natural uranium. Depending on the fixed charges (i.e., FEFJ, separations, etc.), the conservation index will reach a maximum and will decrease with further increases in the cost of natural uranium. However, provided the interest charges on the fuel inventory are small, the fuel cost of a breeder reactor will be independent of the price of natural uranium. Thus, there is some point as the price of natural uranium is increased at which breeder reactors, however inefficient, will be cheaper than converter reactors. It is conjectured that this point will be at a uranium price much less than that for which the fuel utilization is a maximum. The maximization of ψ net occurs at high exposures. Thus, maximum ψ and minimum cost operating points tend to diverge as the jacketing costs are reduced. Table VIII shows this for the economic factors of Table VI, except that the jacketing cost is varied.

---

* This quality of natural uranium is that necessary as cascade feed to supply the initial enrichment level of the reactor.

** The increased exposure allows greater consumption of plutonium <u>in situ</u>.

*** The separative duty is increased with increased feed costs, which means that more electricity is invested. The conservation calculations will ultimately note this factor quantitatively.

TABLE VI

FUEL COSTS AND CONSERVATION INDICES FOR A WATER MODERATED REACTOR SIMULATION FOR VARIOUS FISSILE ENRICHMENTS

July 1, 1962, Uranium Price Schedule

4.75% Use Charge = $40 FEFJ
Plutonium Credit = $8 per gram plutonium
Graded Irradiation

Initial Enrichment %	Exposure, MWD/T	Grams U-235 in Natural Uranium Per Gram U-235 in Fuel	Total Fuel Cost, mills/kwh_e	Conservation Indices Ψ	Conservation Indices Ψ net
1.48	13,700	1.287	1.58	0.804	1.109
1.81	20,900	1.335	1.29	0.963	1.179
1.98	24,100	1.353	1.24	1.004	1.209
2.47	32,600	1.393	1.20	1.055	1.222
2.72	36,800	1.408	1.20	1.072	1.223
2.96	40.400	1.419	1.21	1.074	1.211

TABLE VII

FUEL COSTS AND CONSERVATION INDICES FOR THE OPTIMUM OPERATION OF A WATER MODERATED REACTOR SIMULATION FOR VARIOUS NATURAL URANIUM PRICES

4.75% Use Charge = $40 FEFJ
Plutonium Credit = 2/3 of the price of the U-235 in 90% enriched uranium(1)
Graded Irradiation

Cost of Natural Uranium as UF_6, $/kg	Optimum Tails, %(2)	For Minimum Fuel Cost: Initial Enrichment	For Minimum Fuel Cost: Exposure, MWD/T	For Minimum Fuel Cost: Mills/kwh_e	Conservation Indices Ψ	Conservation Indices Ψ net
10.00	0.348	2.74	37,100	0.92	0.884	0.983
23.50	0.253	2.51	33,300	1.19	1.060	1.225
30.00	0.227	2.48	32,800	1.31	1.106	1.288
50.00	0.175	2.31	29,900	1.64	1.177	1.409
100.00	0.116	2.05	25,400	2.34	1.226	1.528
500.00	0.034	1.78	20,300	7.10	1.234	1.620
1,000.00	0.019	1.70	18,700	12.68	1.205	1.606

(1) Corresponds to $8 per gram plutonium for a feed cost of $23.50/kg uranium.
(2) Based on separative duty cost of $30/kg uranium.

TABLE VIII

MINIMUM FUEL COSTS AND CORRESPONDING ENRICHMENT OPERATING POINT FOR VARIOUS FUEL JACKETING COSTS

(For a water moderated simulation graded cycle.)

FEFJ $/lb U	Minimum Fuel Cost (mills/kwh_e)	Enrichment wt % U-235 in U-238	ψ^* net
40	1.194	2.51	1.30
20	0.978	2.12	1.30
10	0.855	1.95	1.29
5	0.788	1.86	1.27

* Interpolated from Table VI.

Although other reactor types will give results of a different magnitude, the same general conclusions will likely prevail. Calculations for a D_2O-moderated zirconium-clad reactor have just been completed. It is planned to investigate a water-moderated stainless steel-clad type and a thermal breeder type. The magnitude of ψ should approach 1.6 for a successful natural uranium throw-away system.

Analytical Representation of the Westcott Non-1/v Factors

The cross section system used in the MELEAGER code is that developed by Westcott which has the form

$$\hat{\sigma} = \sigma_0 \left[g(T) + r\ s\ (T) \right] \qquad (2)$$

where

$\hat{\sigma}$ = the effective cross section for the reaction

σ_0 = the cross section for a reaction with neutrons having a velocity of 2200 m/sec

T = neutron temperature

r = spectral index (a measure of the proportion of neutrons having epithermal energies)

g = 'non-1/v' factor for thermal energies

s = 'non-1/v' factor for epithermal energies.

In this formulation, g and s represent the degree of difference between the integral of $\phi(v)\sigma(v)$ for the actual reaction cross section and the integral for a cross section that varies as 1/v. The neutron flux, ϕ, in this integration is assumed to be a Maxwellian in the thermal portion and to be inversely proportional to the energy in the epithermal portion. The boundary between these portions is proportional to the most probable neutron temperature (taken as the peak of the Maxwellian) so that both the g and s factor will depend strongly on the neutron temperature, T.

Because the neutron temperature changes throughout a burnup computation, the non-1/v factors must be available in MELEAGER as continuous functions of temperature. Therefore, a concise analytical expression must be obtained to represent these factors. Two functions have been investigated for this purpose -- a polynomial series and a half-range cosine series. In addition, since the non-1/v factors are related to the resonance integral, a functional representation of this parameter has also been investigated.

Experimental fits of the uranium and plutonium isotopic values have been made with both the polynomial and cosine series. (In most cases the temperature range was from 20 C to 1300 C.) The main conclusions are as follows:

1. With the exception of the s factor and the resonance integral for Pu-239 and Pu-241, an adequate representation can be obtained with a fourth order series (i.e., five arbitrary coefficients) of either function. The reason for these exceptions is that both isotopes have a large low energy resonance which is in the epithermal region for low neutron temperatures (moderator temperatures of 100 C to 200 C) and in the thermal region for high neutron temperatures (moderator temperature of 500 C to 800 C).

2. The cosine series (on the basis of the minimization of the sum of the squared errors) is usually better than the polynomial when the series is limited to the third order. However, when a greater number of terms are allowed, the polynomial usually provides a better fit.

3. A given order polynomial fit invariably gives a better representation of the resonance integral than it does of the s factor.

Table IX shows the order of the series that is required to give a specified accuracy for the parameters of Pu-239 and Pu-241. An accuracy of 0.5% was obtained with a fourth order series for all other parameters investigated. The allowable error that can be used for s is much larger than for g, as the r values used to multiply s in expression (2) for the effective cross section are usually less than 0.2 and are seriously questioned if above 0.3.

TABLE IX

ORDER OF THE SERIES REQUIRED FOR A GIVEN ACCURACY FOR Pu-239 AND Pu-241

Allowable Error .... Data	Polynomial 5.0%	Polynomial 1.0%	Polynomial 0.5%	Cosine 5.0%	Cosine 1.0%	Cosine 0.5%
Pu-239						
g factor - absorption	1	3	4	1	3	3
s factor - absorption	5	9	11	3	7	10
resonance integral - absorption	3	7	9	3	5	9
g factor - fission	1	3	4	1	3	3
s factor - fission	5	9	11	3	7	10
resonance integral - fission	3	7	9	2	6	9
Pu-241						
g factor - absorption	1	3	3	1	2	3
s factor - absorption	3	7	8	3	6	8
resonance integral - absorption	2	6	8	2	7	>10
g factor - fission	1	3	3	1	2	3
s factor - fission	3	7	8	3	6	8
resonance integral - fission	2	6	8	2	7	>10

Code Development

1. JASON

Work on the JASON-PLOTTER chain code progressed to the point where the code works satisfactorily through all codes up to and including the QUICK economics code for several modes of code operation. The modes tested during debugging have been those most usually used in economics or isotope production surveys. Some work remains to be done in coupling PLOTTER to the chain. Completion of the JASON-PLOTTER chain will provide a survey tool that permits ready evaluation of various fuel geometries with a reasonable degree of sophistication. The code accepts the fuel lattice as geometry and various isotope concentrations, converts to an axially symmetrical model, computes thermal utilization and resonance escape probabilities for each enrichment level (including plutonium), feeds these and other data to MELEAGER Burnup; and after burnup, the data are fed to fuel cost codes, which calculate the fuel cost; these data then go to MINIMIZER, the minimum cost operating points are determined; and all the pertinent data from the beginning to the end are then plotted.

It should be noted that for identical parameter cases, k (reactivity) calculated by MELEAGER does not have the same value as k calculated by JASON. However, the reactivity can be made to agree by increasing the value of the MELEAGER parameter SDPV, the volume weighted slowing down power. Increasing SDPV softens the spectrum and, in general, increases η and p.

By design, the MELEAGER definitions of η and p are not the same as the JASON definitions. Neither is the method of calculating k. JASON uses the classical four factor formula. The factors are calculated separately and multiplied to obtain k. In calculating the four factors, JASON takes into account the geometry of the fuel element and the neutron flux distribution that results from the particular geometry and cell composition.

By contrast, MELEAGER calculates ηpf (the neutron yield/neutrons absorbed) in a combined calculation which facilitates assessing the effects of burnup. Epsilon (ϵ), the fast fission factor, is entered as an externally calculated parameter and is multiplied by the internally calculated ηpf to obtain k. MELEAGER was specifically developed for plutonium burnup which involves resonance fission. As a consequence, the classical formulation of the discrete values of η, p, f are no longer applicable. Any effects of geometry in MELEAGER are included in the calculation of k by supplying appropriate values of SCA (the resonance shielding index) and SDPV (the volume averaged slowing down term). Both SCA and SDPV are used to modify the Westcott effective cross sections to give MELEAGER effective cross sections. These "geometry corrected" cross sections are then used in the ηpf calculation. With proper calibration, MELEAGER can be a very effective code.

2. MELEAGER

The subroutines, as altered by EDPO personnel, have been incorporated into the MELEAGER code and added to the Chain system. Survey type zone spectrum or "spectral shift" studies and "shielded burnable poison" studies can now be undertaken on the Chain system.

The fourth or dummy array has been modified so that the ten possible isotopes can be treated as belonging to a single interacting parent descendent group or as up to four separate parent descendent groups.

3. LEMON

The LEMON computer code which formerly computed a "least squares" polynomial to a set of empirical data points, now contains a Fourier series fitting routine. There is no restriction to the spacing of the data points -- a restriction of the existing codes. The LEMON code is now completely debugged with the polynomial - Fourier series option.

A set of instructions for preparing input to the Fuel Cycle Analysis Chained Program was completed. Copies of the instructions were distributed to frequent users of the chained programs. This brings together for the first time in a "do it yourself" form the pertinent information needed to operate JASON, MELEAGER, PROTEUS, QUICK, OPTIMIZER, PLOTTERS, and related CASE GENERATORS.

W. K. Woods/jm

Manager,
Programming

WK Woods:jm

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF MAY 1963

A. ORGANIZATION AND PERSONNEL

Charles N. Anderson resigned from the Company. Louise E. Nunn completed her temporary assignment. Mary S. Hostkoetter retired from the Company. Dennis C. Carl joined the Internal Dosimetry Operation. Myrtle S. Ghent and Lena R. Smith were reactivated from medical leaves of absence into External Dosimetry and Radiological Development and Calibrations, respectively.

B. ACTIVITIES

Occupational Exposure Experience

There was one new plutonium deposition confirmed during the month. The total number of deposition cases that have occurred at Hanford is 317 of which 229 are currently employed.

Eleven plutonium incidents required special bioassay sampling of personnel involved to determine if internal deposition occurred. Bioassay results are now pending.

A CPD operator tore a hood and surgeon's glove at the 234-5 Building on April 23, 1963, while cleaning surfaces in a plutonium machining hood. Although the surgeon's glove was contaminated, no contamination could be found on the employee's hand at the time of the incident. On May 2, 1963, the employee noticed a slight indication of contamination when he checked his hands on a four-fold counter. A survey using a poppy revealed 500 d/m contamination on the pad of the left index finger. Examination at the whole body counter on May 2, 1963, showed 5.7×10^{-4} μc plutonium. No DTPA treatment or excision was performed by Occupational Medical Operation.

Four CPD employees received plutonium nasal contamination ranging from 2,400 to 24,000 d/m at the 234-5 Building on May 23, 1963. The employees were working in a green house located in the duct level replacing filters in the 9-B hood exhaust system. They removed their respiratory protection upon completion of the work after a survey failed to show any contamination. Within a few minutes a poppy response indicated a rapid increase in the level of the plutonium contamination and the area was evacuated.

A CPD operator received a plutonium contaminated injury at the 234-5 Building on May 3, 1963, while working with plutonium nitrate solution in a hood. The employee was inserting a glass tube into a tygon hose when the tube broke causing a puncture on his right index finger. The initial survey of the wound showed greater than 40,000 d/m plutonium contamination on the skin and in the blood smears. Examination at the plutonium wound counter indicated the presence of 0.044 μc plutonium in the wound. After excision at Kadlec Hospital, no detectable contamination was left.

A CPD operator received fission product contamination up to 6,000 c/m on his face on May 14, 1963, while removing paper covering from the floor of a flat car. The flat car was used to move new equipment into the Redox tunnel. A survey upon completion of the work showed contamination only on the employee's face. Nasal smears counted approximately 200 c/m. Examination at the whole body counter on May 15, 1963, showed the presence of approximately 1% of the MPBB (20 μc - total body) of zirconium-niobium-95.

Six HL employees were exposed to airborne noble gases and tritium contamination at the PRTR on May 17, 1963, when a high pressure helium valve, located in the hot shop, broke. As a result of the loss of helium to the atmosphere, air contamination in the reactor containment vessel increased to a level $> 1 \times 10^{-6}$ μc/cc. Contamination on the six employees ranged from 500 to 30,000 c/m. Examination at the whole body counter showed the presence of xenon-135 ranging from a positive count to 0.4 μc. Analysis of urine for tritium showed an integrated dose for the next 28 days of 10 to 55 mrads.

Two IPD maintenance employees received nasal contamination ranging from 500 c/m to 2,000 c/m while repairing diversion valves at the 105-KE retention basin on May 28, 1963. Upon completion of the work, contamination ranging from 1,000 c/m to 8,000 c/m was found on the faces of the two employees. Examination at the whole body counter showed the presence of ruthenium-103, chromium-51, zinc-65, zirconium-niobium-95, and scandium-46 in the employees. The quantity of each isotope was less than 1% of the MPBB for each isotope.

Several GE employees were inadvertently exposed to a 27-curie iridium-192 source at N reactor on May 16, 1963. The source, licensed by the AEC for use by the Pittsburg Testing Laboratory, remained in an unshielded and unattended location in Cell 4 of the 109 Building for several hours. During the time the source was unattended, at least 14 GE employees were in the immediate vicinity. Evaluation of the film badge dosimeters worn by the GE employees showed a maximum whole body dose of 0.2 rem of gamma radiation. The exposures received by the construction personnel were investigated by the AEC Compliance Division.

A false critical radiation alarm occurred at the 309 Building on May 31, 1963, at 3:30 a.m. as the result of X-ray testing in the -11-foot level of C cell. The reactor engineer had just announced an X-ray shot over the loud speaker when the criticality false alarm sounded. Personnel evacuated the containment vessel. They did not evacuate the 309 Building since they assumed that the alarm was caused by the X-ray work. The alarm was bypassed for the remainder of the X-ray work. The reactor was not operating at the time of the alarm.

During the reporting period, there were two IPD incidents involving three employees with a potential for a significant overexposure.

An IPD pipefitter received a whole body dose of 0.7 r gamma radiation on May 14, 1963, while attempting to dislodge some irradiated balls that were hung up in the vacuum hose at the 105-KW reactor. The dose rate in the inner rod room to perform this work was measured at 1 r/hour. Upon completion of the task, off-scale readings were recorded for the employee's gamma pocket dosimeters. A survey of the work area after the incident revealed dose rates to 500 rads/hour six inches from the spot of the hose where the balls were dislodged.

Two IPD employees could have been exposed to high dose rates at the 100 C burial ground when a dump truck fell into a process tube burial pit on May 21, 1963. The bridge over the pit collapsed when the truck, loaded with sand, was backed onto it. A survey of the pit after the incident occurred showed a dose rate of 2 r/hour at the location of the employees. Evaluation of the film badge dosimeters worn by the two employees showed a dose of < 0.05 rem of gamma radiation.

Environmental Experience

General increases were noted in all air filter sample results during the week ending May 24, 1963. The average value at the off-site Pacific Northwest locations was 13 pc β/m^3, one of the highest weekly averages noted since the fall of 1961. This increase probably represents the expected spring influx of older fallout materials from past tests. Two of the air filters containing the highest concentrations of beta emitters were scanned in an attempt to estimate the age of the fission products collected. The results indicated an age of approximately one year.

A fuel element failure at 105-KE on May 12, 1963, released a small amount of fission products to the river. The incremental increase in activity was measured by the continuous river monitor at the 300 Area. Based on the results of routine river water analyses and the activity pulse indicated by the continuous river monitor, it was inferred that an infant with a 2-gram thyroid could have received a thyroid dose of 2 or 3 mrems from the radioiodine in the water. Other critical organ dose estimates were even smaller.

The following 252 biological, produce, and food samples were obtained for radiochemical analysis:

Milk	79 samples	217 gallons
Pasture grass	61 samples	
Beef thyroids	16 sets	
Oysters	3 samples	6 pounds
Ground round	7 samples	14 pounds
Fresh vegetables	6 samples	12 pounds
Fish	80 samples	

Studies and Improvements

The air flow rates of the stack exhaust sampling systems at Purex and Redox were calibrated this month. All air flows were within 10% of the figures previously used. Air sampling equipment in Yakima and Klamath Falls was recalibrated for correct air flow and checked for proper operation.

Data from dye tests between PRTR and Pasco, reported in HW-73672, were analyzed for an estimate of eddy diffusion coefficients in this part of the river. Although the experiment was not designed to provide this information, the calculated value for longitudinal diffusion of about 4×10^7 ft^2/day is comparable with values reported in the literature.

Plans were made to begin sampling the Richland water at more frequent intervals since the new water plant is scheduled to begin operation in June. The layout of the distribution system and the plans of operation indicate that it will be several months before the Columbia River water will be distributed throughout the city. During low water usage months, only Columbia River water will be supplied to the system.

Analysis of data from volunteers on an experiment involving a single intake of milk from cows fed I^{131} is complete for all but the last intake. The results indicate that fractional thyroid uptakes range from 0.1 to 0.3. The I^{131} is eliminated with an effective half-life of from 6.5 to 7.2 days. The last intake began May 22, 1963. The test milk is from a cow on a diet which includes 2 g/day of stable iodine.

Further studies were performed to provide precisely timed, short duration gamma or beta irradiations with the electron Van de Graaff accelerator. Relocation of the electron beam catching assembly was necessary to reduce scattered radiation dose when the beam is in the "off" position. A potential of about 3500 volts is now necessary to sweep the beam from the "off" position to the "on" position. The prototype equipment was tested at this potential and found to operate satisfactorily. With the present shielding arrangements, a background of about 5% of the "burst" dose will be present during short-duration, high intensity irradiations.

Attempts to provide neutron monitoring instrumentation with higher dose rate response by changing the gas composition and pressure within the currently used BF_3 detector tubes were not successful. The use of cadmium shielding about the BF_3 tubes has shown promise in reducing the sensitivity of these tubes. Cadmium shields from 0.0015" to 0.020" were studied. Factors of 10 and 100 in thermal neutron sensitivity reduction were attained using 0.0045" and 0.016" cadmium shields, respectively. Some further tests are necessary to determine if the performance of the BFQ double moderator instrument with cadmium-shielded BF_3 tubes is suitable for dose rate monitoring applications.

Neutron exposures using the positive ion accelerator continued with emphasis on studies of the neutron response of several types of silicon diodes. A total of four days was available on the accelerator during May. The energy response measured to date was constant within $\pm$ 20% although the entire neutron spectrum from 0.2 to 16 Mev was not yet studied.

The fast neutron spectrum was measured using the Li-6 neutron spectrometer at a point on top of a plutonium storage hood in the 234-5 Building. This preliminary measurement was a week end run and did not accumulate good statistics above about 1.0 Mev. The spectrum has a pronounced peak at approximately 0.8 Mev. Before the energy scale can be determined precisely, the spectrometer will be calibrated with the positive ion accelerator. Present plans are to accumulate data for 10 days and then calibrate the spectrometer system the next time the accelerator is available.

In an earlier experiment, it was found that the spectrometer could not measure the neutron spectrum from a PuBe neutron source. This is apparently due to silicon activation caused by neutrons with energies greater than 5.0 Mev. Approximately half of the PuBe spectrum is above 5.0 Mev.

All of the data taken so far were obtained with the spectrometer operating in the non-coincidence mode. A dual beam oscilloscope received this month was used to determine what the coincidence timing problems are and the necessary modifications are being made to the electronics system. The use of the coincidence counting will eliminate the background counting problems.

The particle generation equipment was modified for use in calibrating the Royco Particle Counter. Several attempts were made to calibrate the counter but they were not successful. A work order was issued for a complete maintenance and servicing by an instrument technician. The instrument seemed to be extremely noisy with spurious counts in the lower channels. The Goetz aerosol spectrometer was operated together with an aerosol generator built by Particulate and Gas Waste Research personnel in an attempt to determine the particle size distribution of a methylene blue and fluorescein mixture. Several runs were made and the filters are being analyzed by Chemical Research.

The mechanized system for reading ionization chamber dosimeters and recording the output as a typed report and on electronic data processing cards is being assembled. The unit is mechanized to the extent that insertion of the pencil dosimeter in the reader socket starts the appropriate timers and provides complete readout. The equipment is designed to provide one pencil reading each 12-second period. Modifications to provide readings at the rate of one each six seconds can be made when the appropriate programmer is obtained.

The two crystal array in the whole body counter was calibrated for Cs^{137}, Zn^{65} and potassium. The calibration of the mobile whole body was completed.

C. RELATIONS

Six suggestions were submitted by personnel of the Radiation Protection Operation during the month. One suggestion was rejected. Six suggestions are pending evaluation.

Safety meetings were held throughout the Section during the month. Topics included water safety, vacation safety, and a film entitled "Radiation in Perspective".

Radiation protection orientation lectures were presented to three Chemical Effluents Technology personnel, personnel at the Experimental Animal Farm, and forty Transportation and Maintenance Operation personnel. Two talks on the operation of alpha detection instruments were presented to Metallurgy Development employees.

A three-hour practice session and critique was conducted for members of the Emergency Radiological Staff. The exercise involved a simulated loss of water and subsequent melt-down of 105-KE reactor with the ensuing cloud of radioactive material passing directly over 100 N Area while construction personnel were still at work. The session was attended by 14 members of the Staff.

Three 4-hour sessions on Disaster Monitoring were presented in the 202-A Building Conference Room. These sessions were attended by forty Radiation Monitors and first-line supervision from CPD, IPD, and HL.

D. SIGNIFICANT REPORTS

HW-76525-4 - "Radiological Status of the Hanford Environs for April 1963" by R. F. Foster.

HW-77690 - - "Methods Used to Establish Permissible Concentrations of Radioisotopes in Fresh and Sea Water" by R. F. Foster.

HW-77775 - - "Monthly Report for May 1963 - Radiation Monitoring Operation" by A. J. Stevens.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

External Exposure Above Permissible Limits	May	1963
Whole Body Penetrating	0	1
Whole Body Skin	0	0
Extremity	0	0
Hanford Pocket Dosimeters		
Dosimeters Processed	6,312	32,612
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	9,733	47,233
Results - 100-300 mrads - β	202	844
- 300-500 mrads - β	20	91
- Over 500 mrads - β	1	17
Lost Results	20	113
Average Dose per Film Packet - mrad (ow)	9.2	6.6
- mr (s)	26.4	35.2
Hanford Neutron Film Badge Dosimeter		
Slow Neutron		
Film Processed	1,872	8,547
Results - 50-100 mrem	3	7
- 100-300 mrem	0	2
- Over 300 mrem	0	0
Lost Results	15	66
Fast Neutron		
Film Read	511	2,159
Results - 50-100 mrem	33	199
- 100-300 mrem	14	377
- Over 300 mrem	0	4
Lost Results	13	45
Hand Checks		
Checks Taken - Alpha	34,547	184,511
- Beta-Gamma	59,936	302,234
Skin Contamination		
Plutonium	29	125
Fission Products	52	203
Uranium	1	1
Tritium	0	0

Whole Body Counter

Subject	Number of Examinations 747-A WBC	1963	Mobile WBC	1963
GE Employees				
Regular	117	332	30	30
Incident Cases	32	93		
Terminations	6	29		
New Hires	47	88		
Special Studies	99	239		
Non-Employees				
Children	4	5		
Visitors	2	11		
Environmental Studies	5	9		
	312	806	30	30

Bioassay

Analysis	Current Reporting Limit	Results Above Reporting Limit May	1963	Samples Assayed May	1963
Plutonium	2.2×10^{-8} μc/sample	63	472	606	3,422
Fission Products	3.1×10^{-5} μc/sample	6	38	505	3,218
Strontium	3.1×10^{-5} μc/sample	0	0	0	0
Tritium	5.0 μc/l	195	782	280	1,305
Uranium	0.14 μgm/l	0	0	118	731
Special Studies		0	0	17	255

Calibrations

	Number of Units Calibrated May	1963
Portable Instruments		
CP Meter	1,115	5,330
Juno	277	1,360
GM	590	2,851
Other	160	959
Audits	103	524
	2,245	11,024
Personnel Meters		
Badge Film	648	3,788
Pencils	105	660
Other	288	1,454
	1,041	5,902

	Number of Units Calibrated May	1963
Miscellaneous Special Services	654	9,262
Total Number of Calibrations	3,940	26,188

Carl M. Unruh

for the
Manager
RADIATION PROTECTION

AR Keene:CMU:ljw

FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

Sixty-five 200-word summaries for the FY 1964 Science Information Exchange publication were prepared by Hanford Laboratories and submitted to RLOO-AEC at the request of the Division of Biology and Medicine.

The Hanford Laboratories' Control Budget was adjusted during the month to permit optimum utilization of program funds for FY 1963. A summary of the adjustments by research and development program follows:

(Dollars in thousands)	Previous Control	Adjustment Increase (Reduction)	New Control
04 Program			
Plutonium Recycle Program	$7 015	$(90)	$6 925
Irradiation Damage to Reactor Metals	1 315	100	1 415
Gas Cooled Reactor - Loop Project	114	(3)	111
Experimental Gas Cooled Reactor	85	(10)	75
Gas Cooled Reactor - Other	351	(22)	329
High Temperature Lattice Physics Studies	70	(5)	65
Neutron Flux Monitors	70	(5)	65
Waste Calcination Demonstration - Design	200	20	220
Low & Intermediate Waste Studies	190	(20)	170
05 Program			
Radiation Effects on Metals	126	16	142
Plutonium Physical Metallurgy Research	60	19	79
Statistical Eval. & Development	85	(12)	73
Isotopic Analyses	208	(7)	201
Radiochemical Analyses	589	(16)	573
06 Program			
Terrestrial and Fresh Water Ecology	772	(15)	757
Atmospheric Radiation	274	15	289

To reconcile Hanford Laboratories' control budgets with AEC financial planning, $200,000 was transferred from the 04 Program capital equipment budget to the research and development budget. The AEC Financial Plan, dated February 26, 1963, withdrew $235,000 from the Gas Cooled Reactor Program. Later, RLOO-AEC indicated intention to restore these funds. An underrun in equipment expenditures, now apparent, eliminates the need for such action.

Activities for which special accounting codes were established are described below:

.6X University of Washington - Summer Institute for 1963. Authorized funds are $300 to cover off-site travel and miscellaneous on-site expenses.

Organizational code 7210, Nuclear Health & Safety, was established on June 1, 1963 to accumulate the costs of this function, which is being consolidated in the Programming Operation. Concurrently, four employees engaged in this work were transferred from Reactor and Fuels Laboratory to Programming Operation.

General Accounting

Two letters seeking AEC concurrence in proposed actions were in process at month end:

AT-293 Payment of Page Costs for Papers Published in Scientific and Technical Journals (to AEC 5-22-63)

AT-295 Professional Research and Teaching Leave - K. R. Merckx

The following revised OPGs were issued during May:

OPG No.	Title
22.3.1 (pp. 3 & 4)	Approval Authorizations
2.3.7	Contract and Accounting Operation Manager Position Guide
1.13	Participation in Hazardous Business

During the month, $2,353,499 were transferred to Plant and Equipment accounts from Work In Progress accounts. Included are three projects transferred from the AEC:

CAH-866	Shielded Analytical Lab. - 325 Bldg.	$ 631 552
CAH-867	Fuel Element Rupture Test Facility (PRTR)	1 498 020
CAH-963	Geological and Hydrological Wells - FY 1962	71 524
		$2 201 096

Hanford Laboratories material investment at May 1, 1963 totaled $26.1 million as detailed below:

	(In Thousands)
SS Material	$ 24 711
Reactor and Other Special Materials	1 040
Spare Parts	324 -1)
	$ 26 075

(1- Includes a reserve of $78,513.

The value of nuclear material consumed in research by Hanford Laboratories this fiscal year to May 1, 1963 is $3.5 million comprised as follows:

2000 Program	$ 1 007
3000 Program	988
4000 Program	1 456
	$ 3 451

Hanford Laboratories Property Accounting discontinued the use of Parts 1 (tickler card) and 2(receipt card) of the Property Record Unit Control Card effective with May business. This should result in a saving of approximately 32 man-hours a month to C&AO Property Accounting and 10 to Hanford Laboratories. Property custodians will continue to receive Parts 3 and 4 of the Record Unit Card and will be responsible for verifying accuracy of the information thereon.

Nuclear Materials Operation announced the forthcoming Survey 20, Part 4, which will consist of a verification of HAPO inventories of plutonium (including WINE) and deuterium as of the end of June 1963.

Results reported on the physical inventories of movable catalogued equipment in the custody of Biology Laboratory and Test Reactor and Auxiliaries Operation personnel show:

	Biology Laboratory Quantity	Biology Laboratory Value	Test Reactor & Aux. Quantity	Test Reactor & Aux. Value
Balance Prior to Inventory	892	$728 597	1 436	$2 400 790
Reconciled Adjustment	13	20 948	(3)	(30 960)
Adjusted Book Balance	905	749 545	1 433	2 369 830
Physical Inventory	904	749 467	1 433	2 369 830
Missing Equipment	(1)	$ (78)	--	--

Biology Laboratory personnel are continuing to search for the one missing item - a pump.

Test Reactor and Auxiliaries Operation results reflect close control by custodial personnel. The addition of 32 items of unrecorded property valued at $9,898 to the record as a result of the inventory is unusually high but not unexpected, since this is the first physical inventory following (1) reassignment of FPD personnel and equipment to Hanford Laboratories, and (2) PRTR project unitization.

The following projects were unitized during the month and costs transferred to classified plant and equipment accounts:

CGH-858	High Level Utility Cell - 327 Bldg.	$ 383 686
CAH-866	Shielded Analytical Laboratory - 325 Bldg.	631 603
CAH-888	Biology Laboratory Improvements - 108-F Bldg.	417 578
CAH-901	Structural Materials Irradiation Test Equip.	123 313
CAH-963	Geological and Hydrological Wells - FY 1962	71 524
		$1 627 704

Savannah River Operations' calculations on a return shipment of heavy water made in April indicated receipt of 16,612 pounds which was nearly 56 pounds higher than the HAPO computed weight. This resulted in a credit to operating cost of $102. The estimated PRTR heavy water inventory at the end of May showed a loss of 1,871 pounds amounting to $25,971, while the PRCF loss amounted to 11 pounds valued at $150. Estimated heavy water scrap generated during the month amounted to 2,227 pounds resulting in a $3,073 charge to operating cost.

Hanford Laboratories Special Reactor Materials - Non-Fund Account (Account No. 0567) was reviewed and adjusted downward $219,228 to reflect the latest SROO heavy water non-fund price of $6.23 per pound. The offsetting charge was to Costs - Non-Product Depreciation from Off-Site which is closed to the AEC at fiscal year end. The non-fund price has steadily declined, which accounts for the large adjustment at this date. Responsibility for account 0567 was transferred to Hanford Laboratories from C&AO in April 1963, and this is the first time an adjustment has been made since the Laboratories began using heavy water.

At the request of Contract and Accounting, an effort was made to verify the existence of all returnable containers (349) assigned to Hanford Laboratories components as of February 1, 1963. Results submitted to Accounts Payable showed 126 had been returned to Central Stores and 12 are missing, most likely as a result of unrecorded destruction for safety reasons or return to Central Stores. Containers not located through a search at Central Stores will be purchased from the vendor(s).

Laboratory Storage Pool activity is summarized below:

	Current Month		FY to Date	
	Quantity	Value	Quantity	Value
Beginning Balance	1 277	$751 901	1 081	$ 562 200
Items Received	287	71 019	1 497	735 877
Items Reclaimed by Custodians	(27)	(34 085)	(172)	(217 212)
Equipment Transfers	(26)	(10 601)	(220)	(84 131)
Items Disposed of by PDR	(59)	(3 530)	(181)	(20 380)
Items Excessed	(88)	(33 147)	(641)	(193 532)
Adjustment	--	--	--	(41 265)
	1 364	$741 557	1 364	$ 741 557 -1)

(1- Includes 122 items valued at $101,843 on loan at May 31.

During the month, 36 items valued at $20,764 were loaned and/or transferred in lieu of purchases. A total of 343 items valued at $184,312 has been redirected to useful purposes this fiscal year. Operating cost for the same period was $14,263 indicating a net saving of $170,049 this fiscal year.

Total investment of equipment and materials in custody of the Laboratory Storage Pool at May 31, 1963 was $1.3 million including Reactor and Other Special Materials valued at $323,431 and other materials valued at $242,615.

Action occurred during the month as follows on projects indicated:

New Money Authorized to Hanford Laboratories

CAH-100	High Temperature Lattice Test Reactor	$ 4 000
CGH-999	Plutonium Recycle Critical Facility Conversion to Light Water	145 000

Construction Completion and Cost Closing Statement Issued

CAH-867 Fuel Element Rupture Test Facility

1103612

During the month the following contracts were processed:

CA-394 E. C. Lingafelter
SA-281 Background Music Systems
SA-282 State of Washington Department of Game

Personnel Accounting

Mary S. Hostkoetter will start normal retirement on June 1, 1963.

A. K. Postma received patent award No. HWIR 1190 covering the Separation of Sub-Sieve Particles Into Narrow Size Range Fractions.

Personal Share and Beneficiary of Record Statements were distributed to employees during the month. This action stimulated many employees to designate changes of beneficiary.

Personnel statistics follow:

Number of Hanford Laboratories Employees Changes During Month	Total	Exempt	Nonexempt
Employees on payroll at beginning of month	1 652	701	951
Additions and transfers in	27	4	23
Removals and transfers out	17	7	10
Employees on payroll at end of month	1 662	698	964

Overtime Payments During Month	May	April
Exempt	$ 7 161	$ 6 143
Nonexempt	27 009	26 389
Total	$ 34 170	$ 32 532

Gross Payroll Paid During Month	May	April
Exempt	$ 684 802	$ 692 299
Nonexempt	548 497	527 610
Total	$1 233 299	$1 219 909

Participation in Employee Benefit Plans at Month End	May		April	
	Number	Percent	Number	Percent
Pension	1 484	99.4	1 477	99.4
Insurance Plan - Personal	399		396	
- Dependent	1 256	99.9	1 247	99.8
U. S. Savings Bonds				
Stock Bonus Plan	156	42.3	158	43.0
Savings Plan	65	3.9	67	4.1
Savings and Security Plan	1 144	88.2	1 136	88.2
Good Neighbor Fund	1 194	71.6	1 183	71.5

Insurance Claims	May		April	
Employee Benefits	Number	Amount	Number	Amount
Life Insurance	0	$ -0-	0	-0-
Weekly Sickness and Accident	11	713	14	798
Comprehensive Medical	69	3 872	81	4 376
Dependent Benefits				
Comprehensive Medical	118	9 611	136	11 763
Total	198	$14 196	231	$16 937

TECHNICAL ADMINISTRATION

Employee Relations

Twenty-four nonexempt employment requisitions were filled during the month; 34 remain to be filled.

Suggestion plan activity included 48 submissions, 32 adoptions and 28 rejections. The number of suggestions (135) in process at month end was reduced from the previous month for the second month in a row.

Information and Presentations

News releases on the recycle of the first plutonium fuel element were completed and distributed:

Plant tour activity:

	Number	Total People
General Public Relations Tours	15	364
Special Tours	4	8

Visitors Center activity is summarized below:

May attendance	2 215
Average attendance per day open	82
Cumulative attendance since 6-13-62	44 062
Conducted groups	13 (totaling 341 people)

Beginning June 10, Visitors Center hours will be:

Monday through Friday:	1 to 9 p.m.
Saturday:	12 noon to 9 p.m.
Sunday:	1 to 6 p.m.

Document information flow during the month was comprised of 1,218 titles (10,948 copies) received at Hanford and 79 titles (8,074 copies) sent off-site.

A marked increase in costs of printing articles (page costs) in magazines and professional journals prompted a request to the AEC to increase the FY 1963 allocation for this purpose from $1,500 to $3,000.

Professional Placement

Advanced Degree - Seven Ph.D. applicants visited HAPO for employment interviews. Fifteen offers were extended; four acceptances and eight rejections were received. Twelve offers are currently open.

BS/MS (Direct Placement) - Twelve offers were extended. Ten acceptances and eight rejections were received. Nine offers are currently open.

BS/MS (Program) - Three offers were extended. Fifteen acceptances and 53 rejections were received. Current open offers total 34.

Technical Graduate Program - Seven Technical Graduates were placed on permanent assignment. One new member was added to the roll. Current Program members total 34.

FACILITIES ENGINEERING

Projects

At month's end Facilities Engineering Operation was responsible for 10 active projects having total authorized funds in the amount of $6,435,500. The total estimated cost of these projects is $10,680,000. Expenditures through April 30, 1963 were $594,000.

The following summarizes project activity in May:

Number of authorized projects at month end ----------------- 10

Number of new projects authorized -------------------------- 2
CAH-100 - High Temperature Lattice Test Reactor
CGH-999 - Pu Recycle Critical Facility Conversion to Light Water

Number of projects completed ------------------------------- 0

New projects submitted to the AEC -------------------------- 1
CGH-999 - Pu Recycle Critical Facility Conversion to Light Water

Projects awaiting AEC approval ------------------------------ 2
CAH-985 - Addition to the 222-U Building
CAH-986 - 300 Area Retention Waste System Expansion

Project proposals complete or nearing completion ------------ 2
PRTR Storage Basin & Experimental Facilities Modifications
Heat Transfer Apparatus for Model Studies

The current status of projects authorized or awaiting approval is:

CAH-916 - Fuels Recycle Pilot Plant - Additional funds were made available in reserve up to a total project cost of $5,900,000. The construction contractor was given a Notice of Award and a Notice to Proceed on May 15, 1963. Soil bearing tests have been completed and do not indicate a need to redesign the building foundations. The construction contractor commenced work on May 27, 1963.

CAH-922 - Burst Test Facility for Irradiated Zirconium Tubes - A sub-contract was awarded by J. A. Jones Construction Company for construction of the building shell. Two unsuccessful attempts were made to install the coffer dam in the existing 327 building basin. The dam was then removed, repaired and reinstalled. Total estimated value of equipment to be purchased by GE is $66,000. Orders totaling $51,000 have been placed.

CAH-958 - Plutonium Fuels Testing and Evaluation Laboratories, 308 Building - The Commission has not established a schedule for this project. Relocation of electrical cable trays was completed. Relocation of light fixtures to the crawl space above the new laboratory ceilings is progressing. Plastering of the walls and ceiling is expected to start about the end of the month.

CAH-962 - Low Level Radiochemistry Building - The Architect-Engineer submitted his preliminary design package to the Commission for review on May 7, 1963. The design was approved and review comments discussed with the Architect-Engineer on May 23.

CAH-977 - Facilities for Radioactive Inhalation Studies - Company and Commission representatives visited the Architect-Engineer's offices on May 15, 1963 to discuss design problems. Preliminary design drawings were reviewed by the Company and comments transmitted to the Commission on May 17, 1963.

CAH-982 - Addition to the Radionuclide Facilities - 141-C Building - The design criteria and scope drawings were transmitted to the Commission on April 24, 1963. No word of Commission action has been received.

CAH-985 - Addition to the 222-U Building - The project proposal was submitted to the AEC October 9, 1962 and no action has been taken.

CAH-986 - 300 Area Retention Waste System Expansion - AEC has taken no further action on this project. A meeting called by AEC was attended by A. R. Keene, E. R. Irish, J. L. Boyd, et al. to again review the need for the proposed work.

CGH-992 - Additional Fuel Loading Equipment - 308 Building - Construction is progressing satisfactorily.

CGH-995 - 309 Building Air Conditioning Modifications - Design was completed on May 24, 1963. Installation of duct work is continuing. J. A. Jones Company is preparing a Cost-to-Complete Estimate. It currently appears the water chiller may not be shipped by the July 15, 1963 scheduled delivery date.

CGH-999 - Pu Recycle Critical Facility Conversion to Light Water - The project proposal was transmitted to the Commission May 2, 1963; AEC Directive No. HW-550, dated May 29, 1963, authorizing total project funds has been received. The design criteria document was also transmitted to the Commission early this month for review and comment.

CAH-100 - High Temperature Lattice Test Reactor - A Directive authorizing $30,000 for Title I design services has been received, also a Work Authority authorizing $4,000 to the General Electric Company for technical guidance of reactor design work.

The design criteria document has been reviewed and approved by both the Commission and the Company. Commission approval was obtained May 28, 1963. The text and drawings are now being reproduced.

Engineering Services

Engineering service work and consultation provided to research and development personnel included trouble-shooting and direction of repairs to the 108-F source handling equipment, engineering assistance on installation of PRTR emergency alarm system, and engineering of instrumentation for 314 building.

Pressure Systems

Engineering reviews and surveillance of pressure systems continued during the month. Major items receiving attention included: (1) Low Level Laboratory - instructions to A-E, (2) waste calcination vessel fabrication, (3) 314 building high pressure autoclave, (4) review of autoclave design with viewing window, and (5) C-1 loop. The Third Party Inspector inspected the C-1 loop, witnessed the pressure test and approved the loop for operation. He also reviewed the PRTR gas loop heater and additional investigation is being made of the design and fabrication.

Plant Engineering

Consulting service was provided to operating and maintenance forces. Active jobs during the month included: (1) laboratory buildings waste line radiation monitoring equipment, (2) 325 and 326 transformer changes, (3) 325 normal power and ventilation scheduled outage, (4) 314, 305-B and 3717-B normal power scheduled outage, (5) 300 area power reliability study, (6) 300 area retention waste sampling, (7) 308 condensate pipe replacement, (8) 326 - 20 ton refrigeration unit, (9) 306 process sewer, (10) 340 off-gas filter and filter box replacement, (11) 325 basement mezzanine - air supply, (12) 325 decontamination facility and concreting operation, (13) sub-micron particulate material filter study, and (14) 325 space utilization review.

Major items of trouble-shooting involved 308 building backdraft damper adjustment and criticality alarm system operation.

Facilities Operation

Landlord costs for April were $208,634, which is 130 percent of the forecast for the month. The total cost to date for the first 10 months is $1,700,735, which is 99 percent of predicted. During this month improvement maintenance cost $51,466 as compared to the $15,000 forecast. Steam cost $31,570 against a prediction of $23,000. Engineering at $26,741 continued to exceed the forecast of $12,000. However, over-all expenditures are expected to stay within the landlord budget.

During this month the interim test procedure for criticality alarms was put into effect, and all of the systems in 300 Area Laboratories buildings were tested. Some horns did not actuate, and repairs were made.

The following tabulation summarizes waste disposal operations:

	March	April
Concrete barrels disposed	12	4
Loadluggers - dry waste	28	29
Crib waste, gallons	220 000	235 000

No basins exceeded radiation control limits, but two on different days were in the warning range. On May 11, 1963, Basin #2 activity level was 6.4×10^{-6} μc-β/ml, and on May 15, 1963, Basin #3 activity level was 6.7×10^{-6} μc-a/ml.

The new 450 gpm pump for transferring contaminated retention basin waste to the crib system was installed and tested.

The ventilating system in 325 was shut down on May 3, 4 and 5. The exhaust plenum, fans, stack base and adjacent ducts were decontaminated. They had been coated with a film of contaminated oil from the oil-seal vacuum pumps which were recently replaced by water-seal pumps. During this outage, the electrical switchgear, sheetmetal work and pneumatic control lines were inspected, adjusted, and modified. The system was run under different test conditions to determine its capabilities and flexibility.

Drafting

The equivalent of 130 drawings were completed during the month for an average of 24 man-hours per drawing.

Major jobs in progress are: (1) PRTR shim rod control, (2) PRTR cladding cutter assembly, (3) 108-F inhalation studies hood, (4) equipment for salt-cycle process - 325-A, (5) as-built drawings - PRP Critical Facility, (6) scope work - fast supercritical pressure power reactor concept, (7) scope work - PRTR D_2O decontamination system, and (8) high temperature gas loop.

Work performed by CE&UO and by Bovay Engineers during the month amounted to 16 man-hours and 189 man-hours respectively; work assigned was 80 man-hours and 193 man-hours.

Construction Supervision

Activity during the month on construction work (J. A. Jones Company) being performed for Hanford Laboratories components is given below:

	Unexpended Balance	Waste Calcination (Job 7005)
Orders outstanding beginning of month	$293 645	$223 740
Issued during the month (inc. suppl. and adj.)	197 053	
J. A. Jones expenditures during month (incl. C.O. costs)	178 623	58 529
Balance at month's end	312 075	165 211
Orders closed during month	105 159	

In addition, work on six maintenance work orders having a total face value of $7,852, issued to plant forces, was supervised.

Major active nonproject jobs in progress are: (1) 108-F plant growth room, (2) 108-F rooms 402 and 403, (3) 108-F room 405, (4) 141-C, install X-ray equipment, (5) 146-FR modifications, (6) 231-Z, fabricate and install glove box, (7) repair dams at Rattlesnake Springs, (8) 306, modify canning line lighting, (9) 306, 3705 and 3746, replace and repair roofs, (10) 306 automatic filter, (11) 308, replace steam and condensate lines, (12) 309, install gas loop heater, (13) 309, install shielding, (14) 309 concrete slab equipment, (15) 309 - piping and electrical work, (16) 309, electrical cable to Pole #16, (17) 309, back-up emergency pump controller, (18) 309, inlet and outlet face mock-up, (19) 309, construct rupture loop annex, (20) 314, mock-up equipment, (21) 314, repair and modifications to building, (22) 314, gas bottle facility, (23) 321, hatchway in canyon roof, (24) 321, control room, (25) 321 computer room, (26) 325, repair exhaust duct and filter boxes, (27) 325, exterior stairway to roof, (28) 325 rooms 407 and 407-A, (29) 325, electrical service to 325-A, (30) 326, replace Trane air conditioner, (31) 326, additional ductwork, (32) 327, install demineralizer, (33) 328 third floor offices, (34) 329 laboratory 11-B, (35) 340, install filter, (36) 300 area material storage yard, and (37) waste calcination prototype.

Eight purchase requisitions totaling $2,000 were issued during the month. Total value of equipment being processed is $108,000. New or revised M & E lists were issued for Projects CAH-922 and CGH-995.

W Sale
Manager
Finance and Administration

W Sale:whm

04 PROGRAM - REACTOR DEVELOPMENT

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output for May was 1010 MWD, for an experimental time efficiency of 58.2% and a plant efficiency of 50.3%. There were 13 operating periods during the month, five of which were terminated manually, seven were terminated by scrams (two of the scrams were manual scrams) and one operating period extended through month-end.

The core loading on May 1 consisted of 19 UO_2 elements, 14 Pu-Al elements and 52 mixed-oxide elements. At the end of the month, the core consisted of 15 UO_2 elements, 14 Pu-Al elements and 56 mixed-oxide elements.

Fuel exposure history at month-end was:

Maximum UO_2 exposure/element	3,406 MWD/TU
Average UO_2 exposure/element	2,412 MWD/TU
Maximum Pu-Al exposure/element	93.4 MWD
Average Pu-Al exposure/element	73.2 MWD
Maximum Moxtyl exposure/element	71.0 MWD (~1420 MWD/TU)
Average Moxtyl exposure/element	41.4 MWD (~ 828 MWD/TU)

The status of the various test elements on May 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 May 1, 1963, have been deleted from this table.

Test No.	Channel Location	FE Number	Description	Date Initial Charge	Date Discharged	Approximate Accumulated MWD
10	Basin	1067	UO_2-Vipac	11/3/61	3/26/63	88.3
14	1956	5097	Moxtyl-Swaged	4/2/62	--	37.5 Repad
14	1356	5098	Moxtyl-Vipac	5/8/62	--	71.0 Repad
14	1758	5099	Moxtyl-Vipac	5/8/62	--	61.1 Repad
37	1449	1096	UO_2-Physics	5/12/62	--	73.5
37	1649	1097	UO_2-Physics	5/12/62	--	71.5
37	1552	1098	UO_2-Physics	5/12/62	--	69.0
37	1548	1099	UO_2-Physics	5/12/62	--	70.7
37	1651	1100	UO_2-Physics	5/12/62	--	60.6
48	1243	5150	Moxtyl ($\frac{1}{2}$"x$\frac{1}{2}$" pads)	8/1/62	--	48.4
54	1948	5116	Moxtyl (Clip-on Pads)	5/8/62	--	60.5(45.0 w/clip)
54	1855	5118	Moxtyl (Clip-on Pads)	5/8/62	--	67.0(52.0 w/clip)
61	Basin	5176	Moxtyl-Physics	4/5/63	5/4/63	5.6
61	1150	5179	Moxtyl-Physics	5/28/63	--	<1
61	1443	5185	Moxtyl-Physics	5/28/63	--	<1
61	1756	5186	Moxtyl-Physics	5/28/63	--	<1
61	1847	5187	Moxtyl-Physics	5/28/63	--	<1

The reactor operated at about 60 MW until May 17 when the heat transfer flux limit was increased which permitted raising the power level to 70 MW.

Four new instrumented mixed-oxide elements for physics studies (PRTR Test 61) were charged on May 28.

D_2O and indicated helium losses for May were 1,871 pounds and 111,304 scf, respectively. During the month 1,438 pounds of reagent-contaminated D_2O were cleaned up through use of ion-exchange and filtration methods.

Equipment Experience

A total of 156 reactor outage hours were charged to repair work. Of this amount, 76 hours were used to complete steam export line containment bellows replacement early in the month, 24 hours were for heavy water leak repairs, 21 hours were for moderator pump repairs and cleaning of the strainer on moderator pump discharge line, and 9 hours for boiler level instrumentation repairs and calibration.

Log N #3 fission chamber was replaced. The guide rods were bent, restricting movement.

Preventive maintenance required 239 hours or 4.8% of the total maintenance effort.

Seven process tubes were examined during the month with no significant damage being reported.

Improvement Work Status (Significant Items)

Work Completed

Replace Rupture Disc Assembly on Emergency Depressurization Valve H-97
Shim Rod Rotameter Modification
Fusing Shim Rod Readout Power Supply
Thermal Barrier Positioning Device

Work Partially Completed

Inline Gas Sampling
Process Tubes Level Indicator
Primary Loop Drain and Flush Valve Modifications
Installation of Moderator Heater
Independent Criticality Alarm System of the PRTR Control Room
Recorder Ammeter Installation for Primary Pumps
Deionized Water System
Inlet Gas Seal Replacement
Backup Emergency Pump Low Speed Controllers
Effluent Activity Monitor Replacement

Design Work Completed

125 Volt Battery Disconnect Contactor
Control Wiring for 13.8 KV Motor Operated Switches

Design Work Partially Completed

Additional Fuel Storage and Examination
Vibration Snubbers for Earthquake Protection
PRCF Light Water Conversion
Control Room and Instrument Calibration Shop Air Conditioning
Decontamination Building and D_2O Cleanup
Instrument Power Transfer System
Revision of PRTR Exhaust Fan to Emergency Power
Flow Monitor Tubing Snubber Installation
Provide Separate Emergency Backup Water for FEEF
Autoclave Installation for ZR-2 Fretting Corrosion Studies

Process Engineering and Reactor Physics

A study was made to determine the enrichment needed to better simulate a power reactor fuel and to provide the reactivity needed for extended operating periods. The results of this study are as follow:

(1) The power equivalent of a uranium system enriched to 2% U-235 would be produced by a PuO_2-UO_2 system of approximately 1 w/o PuO_2 in UO_2, assuming the plutonium to be about 8% Pu-240.

(2) Fuel Elements of this composition can be charged into any location in the outer three or four rings of tubes without creating a power limiting situation.

(3) Extended operating periods of at least one month would be possible.

The Perkins and King fission product data was used in a computer program to develop an extensive set of heat decay curves. Also a short-cut method of calculating the shutdown heat generation rate from easily obtained fuel element data was developed.

A program change in the PRP was initiated this month to irradiate selected 19-rod clusters of LX Al-Pu fuel elements to higher than goal exposures in order to obtain increased Pu-240 content.

The final hydrostatic test of the primary system was conducted under PRTR Test No. 55 during May. The test was negative.

Procedures

New Operating Procedures issued	1
Revised Operating Procedures issued	3
Revised Operating Standards issued	8
Temporary Deviations to Operating Standards issued	2
Revised Process Specifications accepted for use	2
Maintenance Manuals and Procedures issued	3
Equipment Standards issued	1

Drawing As-Built Status	April	Total
Approved for As-Built	7	961
In Drafting		93
Voided		79
		1 133
Scheduled for review		327
		1 460

Personnel Training:	Manhours
Qualification subjects	461
Specifications, Standards, Procedures	57
Emergency Procedures	17
Maintenance Procedures	60
	595

Status of Qualified Personnel at Month-end:

Qualified Reactor Engineers	9
Qualified Lead Technicians	6
Qualified Technicians	17
Provisionally Qualified Technicians	1

Plutonium Recycle Critical Facility

Control rod calibrations and direct measurements of safety system time response were completed. The safety system was found to be completely effective within 0.5 sec. following either a period or a high level trip.

Minimum rod worth was achieved by relocating the rods to higher flux regions. Byproduct information of experimental importance was developed on measurement methods and on the magnitude of anti-shadowing.

Preparations for irradiated fuel experiments included (1) rotation of the core loading to move safety rods out-of-the-way for the thimble hoses; (2) first series of successful dry runs of the fuel handling operation using the cell crane; and (3) critical experiments (with cold fuel) to develop reactivity measuring techniques for use in a climate of massive photoneutron flux.

The document, HW-77607, "PRCF Light Water Dilution" was issued. This report analyzed the reactivity status of the PRCF if H_2O were added to the D_2O core when it was critical at low moderator levels.

One Revised Process Specification (A-3, Safety and Control Rods) was accepted.

One Special Condition (No. 6) was accepted regarding integrity of confinement.

Fuel Element Rupture Test Facility

Work Status

The following major elements of work are 90% complete:

Installation of the test section in B Cell
Rupture Loop Sampling System
B Cell Leakage Collection System

Operation

Sixty-four hours were devoted to training.

One additional technician completed the orientation and training period and passed the Rupture Loop qualification test.

GAS COOLED POWER REACTOR PROGRAM

Pressurized Gas-Cooled Loop Facility (Project CAH-822)

Project Status

Blower units two and three have been shipped from the vendor's plant and are scheduled for delivery in mid-June.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 28,170 hours. This includes 19,309 hours performed in Technical Shops, 6,286 hours assigned to J. A. Jones Company, 2,520 hours assigned to off-site vendors, and 55 hours to other project shops. Total shop backlog is 33,421 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 1,775 hours or 7.8% of the total available hours.

Distribution of time was as follows:

	Manhours	% of Total
N-Reactor Department	3 226	11.46
Irradiation Processing Department	6 986	24.79
Chemical Processing Department	353	1.25
Hanford Laboratories	17 605	62.50

Requests for emergency service increased requiring an overtime ratio of 7.8% versus 7.2% for the previous month.

LABORATORY MAINTENANCE OPERATION

Total productive time was 21,600 hours of 23,000 hours potentially available. Of the total productive time, 88.6% was expended in support of Hanford Laboratories components, with the remaining 11.4% used in providing service for other HAPO organizations. Overtime worked during the month was 5.1% of total available hours.

Manpower utilization for May was as follows:

A. Shop Work		4 100 hours
B. Maintenance		8 300 hours
1. Preventive Maintenance	1 900 hours	
2. Emergency or Unscheduled Maintenance	2 200 hours	
3. Normal Scheduled Maintenance	4 200 hours	
4. Overtime (included in above figures)	1 100 hours	
C. R&D Assistance		9 200 hours
		21 600 hours

Three new items of standards equipment were received and placed in service during May: (1) Silver Freeze Point, (2) Copper Freeze Point and (3) a new standard potentiometer accurate to 0.001% of observed reading.

WD Richmond
Manager
Test Reactor and Auxiliaries

WD Richmond

INVENTIONS OR DISCOVERIES

All persons engaged in work that might reasonably be expected to result in inventions or discoveries advise that, to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during the period covered by this report except as listed below. Such persons further advise that, for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

INVENTOR	TITLE OF INVENTION OR DISCOVERY
D. J. Foley, R. J. Hennig D. E. Rasmussen, D. H. White	"Fast Reactor Nuclear Safety and Control Devices"
M. F. Zeutschel	HW-77778, Multi-Beam Ultrasonic Crystal
R. F. Maness	Inhibition of 304-L Stainless Steel Corrosion in Zirflex Solutions by Anodic Passivation
C. A. Rohrmann	"Electrical Power Production from Boiling Fluids Outside of the Earth's Gravitational Field" (5-10-63)

H M Parker

Manager, Hanford Laboratories