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HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT
DECEMBER 1964

JANUARY 15, 1965

HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

GENERAL ELECTRIC

AEC-RLOO RICHLAND, WASH.

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HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT
DECEMBER 1964

90498

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By Authority of CG-PR-2
RM Iten 8-31-92
By J Tang 9-9-92
TL Phillips 9-10-92

Compiled by
Section Managers

January 15, 1965

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HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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PRELIMINARY REPORT

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Table I - Hanford Laboratories Force Report

Date: December 31, 1964

	At Beginning of Month		At Close of Month		
	Exempt	Salaried	Exempt	Salaried	Total
Chemical Laboratory	157	139	157	143	300
Reactor & Fuels Laboratory	209	204	210	204	414
Physics & Instruments Laboratory	134	80	136	81	217
Biology Laboratory	45	63	45	64	109
Applied Mathematics Operation	16	6	16	6	22
Radiation Protection Operation	51	81	50	85	135
Finance & Administration Operation	117	168	119	180	299
Programming Operation	5	2	5	2	7
Test Reactor & Auxiliaries Operation	62	305	69	352	421
General	2	5	2	4	6
TOTAL	798	1053	809	1121	1930

BUDGET AND COST SUMMARY

December operating costs totaled $3,485,000, an increase of $686,000 from the previous month. Fiscal year-to-date costs aggregate $17,329,000 or 48% of the current control budget for FY-1965.

Hanford Laboratories' research and development costs for December compared with the previous month and the current control budget are shown below:

(Dollars in thousands)	COST Current Month	Previous Month	To Date	Budget	% Spent
HL Programs					
02	$ 10	$ 33	$ 268	$ 542	49
04	1 795	1 195	7 826	16 126	49
05	143	120	747	1 694	44
06	358	260	1 817	3 570	51
07			7	7	100
08	47	33	246	500	49
	2 353	1 641	10 911	22 439	49
Sponsored By					
NRD	116	103	763	1 594	48
IPD	21	16	111	425	26
CPD	192	151	1 088	2 207	49
Total	$2 682	$1 911	$12 873	$26 665	48%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

Iron-aluminum additives to uranium fuel continue to appear superior to iron-silicon additives for inhibiting swelling both during irradiation and on postirradiation annealing. Uranium samples with other additives and combinations of additives such as Fe-Al-Si, UC-UP, in varying concentration, are being prepared and evaluated to develop improved compositions for swelling resistance.

[redacted]

$LiAlO_2$ and Li_2SiO_3 ceramic core targets irradiated in KER loops for 58 days released only a small fraction of the total gas generated when the capsules were opened after irradiation.

Metallic Th-U fuel elements operating in the ETR at a maximum temperature of 500 C and a maximum surface heat flux of 700,000 Btu/hr-ft^2 have reached 10,500 Mwd/ton exposure with a volume increase caused by solid state fission products of 1.5 wt%.

Experiments with an electrically heated test section representing the downstream half of an N-Reactor fuel column with interchannel mixing (as could occur at fuel element junctions) showed greater flow stability than tests without interchannel mixing. Higher tube powers may be permissible in N-Reactor than heretofore assumed.

A literature survey was made to determine methods used at other sites to establish boiling burnout safety factors used in the design and operation of nuclear reactors. It was concluded that a standard method does not exist and that burnout safety factors stated for various reactors in the United States are not generally comparable.

Additional data were obtained on critical flow of steam-water mixtures through pipe tees and elbows in the investigation of maximum discharge rates expected from reactor piping breaks. Efforts to predict the results of these tests by modifying correlations developed for flow from straight pipes have been partially successful, but additional work is needed to allow definition of the critical pressure for pipe fittings other than straight pipes.

Tests on AISI 304 SS tubing were completed that indicate residual pickling acids can cause intergranular attack. No attack was found in samples exposed to charred vapor phase inhibitor vapor.

The use of hydrazine (with in-reactor conversion to ammonia) for pH control in the N-Reactor graphite cooling system has reduced the O_2 concentration in the coolant from >15 ppm to <0.05 ppm.

Three nickel-plated Pu-Al fuel elements and five nickel-plated aluminum alloy dummies were irradiated for 30 days in neutral pH water at 260 C. Weight losses on fuel elements were 35 to 50% of the nickel deposit. Metallography showed considerable attack of the nickel terminating at the inner Ni-Al diffusion layer. On dummy elements, the plate was nearly intact.

Nickel-plated aluminum has been exposed to pH-10 water (NH_4OH) at 25 ft/sec and 330 C for 30 days and continues to show excellent corrosion resistance, much less than in pH-10 lithiated water. Weight changes after 30 days are +7 to 15 mg/dm^2; at neutral pH weight losses of 15 to 45 mg/dm^2 were incurred for 30 days at 300 C. Corrosion of the aluminum substrate in NH_4OH solution at 300 C is comparable to that in neutral water and one-tenth that in lithiated water.

Hydrogen pickup of Zircaloy-2 has been studied over a wide range of oxygen-to-water partial pressure ratios at 400 C. Except at very low oxygen concentrations in high pressure steam, hydrogen pickup is largely independent of oxygen to water vapor concentration ratios.

Comparisons of the zirconium alloy corrosion and hydrogen pickup behavior in oxygenated and hydrogenated water can be made on the basis of incomplete information on ETR Cycle 65 (low oxygen-hydrogen addition). Under fast neutron irradiation at flux intensities of about 10^{13} nv pretransition weight gains in hydrogenated water are lower by a factor of at least two and hydrogen pickup fractions greater by a factor of two than in water containing about 1 ppm oxygen (no hydrogen addition).

The second of a series of controlled pressure-temperature uranium swelling capsules continues to operate successfully in a reactor at 1000 psi and 575 C. The first capsule in this series operated at 1000 psi and 450 C.

Several nickel-base alloys are being studied to determine the effect of irradiation and environment upon their mechanical properties. Fourteen

[redacted]

tensile tests were completed at 650 C on Inconel 600, Inconel 625, Inconel 718, and Incoloy 800 specimens irradiated at exposures to 8.3 x 10^{20} nvt fast. These tests, as compared to room temperature tests, show a marked decrease in ductility of all alloys when tested at 650 C (1202 F).

A creep capsule containing an annealed AISI 304 SS specimen was charged into the reactor this month. Test conditions are 30,000 psi stress at 550 C.

Calibration tests to determine the spring constant as a function of crack length on the double cantilever beam specimen have been completed. These data are being used to determine the fracture toughness and ductile-to-brittle transition temperature of Zircaloy-2 and A-302B.

Examination of optical micrographs of stress-ruptured specimens of Inconel X-750 indicates that transverse specimen sectioning provides better detail than does longitudinal sectioning. All microcracks have been intergranular in nature, even with rupture times at 1350 F as low as 10 hr.

In support of the ATR Gas Loop Project, the high pressure helium purification system of the Hanford model gas loop was activated during the period of November 16 through December 3, 1964. The system produced helium with less than 0.1 ppm of the following: H_2, N_2, O_2, CH_4, CO, and CO_2.

The calcia-stabilized zirconium tube oxygen monitor for the model gas loop has been modified to use air as a reference gas. Voltages observed showed dependence on oxygen content as predicted by the theoretical equation. The revised probe is now being incorporated into a portable instrument.

Technical procurement specifications for the ATR gas loop chromatograph and total impurity analyzer have been prepared and have been sent to Ebasco Services, Inc., for comments. It is expected that procurement of these instruments will start about January 4, 1965.

[redacted]

Elevated temperature tensile tests are being conducted with high purity polycrystalline molybdenum to establish the temperature at which the flow stress becomes independent of strain rate. Tests to date have fixed this critical temperature between 900 and 1000 K.

A textured rod of unalloyed plutonium having a [020] axis was transformed into beta and measured and then transformed back to alpha and measured. Dimensional changes are extremely anisotropic.

Calculations indicate that differential contraction of the N-Reactor graphite bars under irradiation may fracture keys on the process tube blocks at about 45°. This would provide a ramp that with continued differential contractions in the parallel directions could result in a gross vertical expansion of the moderator stack. Recent out-of-reactor tests on full size blocks have confirmed both the mode of failure and magnitude of the stresses involved.

Intensity of the Co^{60} source in the gamma irradiation facility in the 3730 Building has been more than doubled. The source now contains approximately 87,000 curies, making possible studies in a radiation field of 7×10^6 r/hr.

The radiation-induced contraction of conventional nuclear graphites transverse to the extrusion direction of the bar at 1050 to 1150 C ranges from 0.3 to 0.8%, depending on the kind of graphite, after an exposure of 4×10^{21} neutrons/cm^2, E > 0.18 Mev. The contraction is much greater parallel to the extrusion direction, ranging from 2 to 4.8%. One experimental grade expanded slightly or contracted less than 0.5%.

Boronated graphite contracts at a high rate at 370 to 540 C when exposed to a predominantly fast neutron environment. When exposed to thermal neutrons, the material may expand at a rapid rate or contract, depending on the degree of dispersion of boron in the graphite.

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Out-of-reactor tests appear to confirm the postulate that PRTR shim rod sticking problems result from residual lubrication breakdown under radiation and/or washoff in hot water, leading to galling between the lead screw and ball nut.

HW-84449, "Transition Schedule - PRTR High Power Density Core," was issued.

A vibrationally compacted UO_2–2 wt% PuO_2 PRTR high power density core fuel element was successfully irradiated to a burnup of 1470 Mwd/ton_U with maximum fuel temperatures estimated to be above melting. Three additional high power density fuel elements were delivered to the PRTR.

A fuel element designed to simulate conditions of extreme fuel cracking and to allow possible impingement of molten UO_2 on the cladding has been irradiated for 4 days in the MTR.

Two modified laboratory test capsules were extensively hydrided at 310 and 320 C in the crevice region of a Vipac end cap. Essentially no hydride formation occurred in the swageable end cap region at the opposite end of the capsules.

A PRTR swage-compacted UO_2 element defected with a 6-1/8 in. longitudinal slit in one of the rods completed 16 days of operation in the rupture loop with only minor activity release.

In an out-of-reactor loop test, a defected Vipac-Nupac UO_2–1 wt% PuO_2 fuel rod, previously irradiated in PRTR to 1890 Mwd/ton, released a small amount of core material when thermal cycled four times from 300 to 40 C. Similar tests on an irradiated, incrementally loaded Vipac rod and an incrementally loaded swaged rod did not produce significant filter activity.

A heterogeneously enriched fuel element failed after an exposure of 6×10^{18} fissions/cm^3 in the MTR. The failed rod consisted of 0.465-in. diam pellets with UO_2–8.5 wt% PuO_2 packed powder peripheral enrichment.

A vibrationally compacted UO_2 fuel element with tube-in-tube geometry failed after an exposure of 3×10^{18} fissions/cm^3.

The underwater replacement of individual rods was successfully demonstrated in an irradiated simulated PRTR 19-rod cluster.

A new technique for variable enrichment of packed-particle fuels involves the use of a helical plutonium wire, which permits continuous axial variation of fissionable atom concentration by simply changing the helix pitch.

Repair of about 500 EBWR fuel rods will be accomplished by removing the cracked end and welding on a new end cap.

Maximum exposures of the 10 in-reactor (MTR) and 22 discharged EBWR specimens are 4.25×10^{20} fissions/cm^3 (16,000 Mwd/ton of fuel) and 3.08×10^{20} fissions/cm^3 (11,600 Mwd/ton of fuel), respectively.

Fabrication of fuel rods for the Saxton Reactor is proceeding on schedule.

No reaction was detected between PuO_2 and NiCr after heating a PuO_2-62 wt% NiCr pellet 12 hr in pure hydrogen at 1200 C. The PuO_2-NiCr cermet showed no swelling upon exposure to air at about 1000 C, in contrast with extensive swelling displayed by a UO_2-NiCr cermet.

Four capsules containing stainless steel-20 vol% PuO_2 cermet pellets were fabricated for irradiation in the ETR.

Extrusion of cermet fuel pins was continued using 93 to 96% dense spheroidized fuel particles. Metallographic examination showed that approximately 25% of the fuel particles fractured during extrusion, but no stringering was produced; that clad thickness control was escellent; and that as-extruded dimensions will allow final sizing of fuel pins by swaging with an approximate OD reduction of 0.010 to 0.015 in.

[redacted]

A process for metallurgically bonding metal cladding and end caps to cermets while simultaneously densifying the cermet has been demonstrated on a variety of materials during recent months.

Measurement of the dissociation pressure of AmO_2 can proceed, with the confirmation that both AmO_2 and PuO_2 are compatible with iridium sample holders at 1100 C, at which temperature both oxides earlier reacted with platinium.

PuN was prepared by carbon reduction of PuO_2 under 600 mm of nitrogen.

A Pu-31.6 at.% Zr alloy was hydrided and subsequently nitrided to form a single phase (Pu, Zr)N.

Modification of the MRC high temperature X-ray diffraction equipment is nearing completion to allow operation at temperatures up to 2200 C.

Electrical resistivity and Seebeck emf of UO_2 were measured, in-reactor, throughout the range between ambient temperature and 1800 C.

2. Physics and Instruments

The N-Reactor Phase III physics study of PuO_2 enriched UO_2 is essentially complete. Some two-dimensional analyses were made on the previously analyzed ThO_2 fuel verifying the validity of assumptions made in those analyses.

PCTR experiments in support of N-Reactor coproduct studies have been completed. Preparation of exponential experiments and N-Reactor irradiation experiments for determining conversion ratios of coproduct fuel are in progress.

Detector and circuit modifications were made to a Pu^{239} liquid sample counting system which improved sensitivity to 1 g/liter.

Enrichment of as-received N-Reactor fuel billets is being checked with a recently developed tester which has a sensitivity of 40,000 cpm per 1% change of enrichment.

A new test was successfully used to insure melting of the braze wire in a group of TIG brazed N-Reactor fuel elements being fabricated for reactor charging.

Criticality experiments on a PuO_2-plastic assembly with various composite reflectors, subcritical neutron multiplication measurements on an array of PuO_2 powder in cans, and exponential pile experiments with enriched uranium tubes in light water, were performed that provide data in support of plant criticality problems.

The effect on reflector savings of positioning neutron absorbing materials between a hydrogenous reflector and a PuO_2-plastic core was determined from critical experiments. Comparisons were made between the effect of boron, and gadolinium in thin stainless steel plates, and cadmium sheet. Data were also obtained for the relative reflector savings of lead as opposed to Lucite, and polyethylene containing boron.

The Class I HL-designed shipping container for fissile materials has now successfully passed drop, fire, and immersion tests in that order. (These containers have Bureau of Explosives approval for use in shipment.)

A new detector and detector shielding system is being prepared for the triple-axis neutron spectrometer to improve its operating characteristics.

It has not been possible to start slow-neutron inelastic scattering measurements with the new Time-of-Flight (TOF) spectrometer due to continued malfunction of the readout system of the TOF analyzer.

The new vacuum-lock sample changer for a mass spectrometer has been satisfactorily bench tested and is scheduled to be installed when the required high-voltage power supply is delivered.

Experimental regenerating neutron flux detectors undergoing long term test at KW-Reactor continue to perform satisfactorily.

A new DRD funded program was initiated to investigate microwave methods of detecting impurities in liquid and gas reactor coolants; later phases of the program will examine microwave temperature measurement techniques.

Ultrasonic boundary waves have been successfully used to detect the direction as well as the existence of fatigue stress in copper and aluminum samples.

Techniques for inspecting tubular geometries with eddy currents and displaying the data in representative cross sections have been demonstrated with samples which have bonafide and artificial discontinuities. Defect geometry, location, and surface eccentricity are shown in true cross-sectional perspective.

Three irradiated PRTR fuel elements are currently being destructively analyzed for burnup information. These are 1.0 wt% PuO_2 UO_2, 0.48 wt% PuO_2 UO_2, and natural UO_2 elements having exposures of 1000 Mwd/ton, 4900 Mwd/ton, and 4000 Mwd/ton, respectively.

Preliminary analysis of the first PuO_2 UO_2 lattice to be measured in the PCTR at a pitch of 8-3/8 in. yielded a k_∞ of 1.21.

Critical experiments with the 1.8 wt% PuAl fuel in H_2O continued in the PRCF. Three-zone loadings, each zone having slight differences in Pu^{240} content, were measured and the results indicated that the average nuclear properties of the highest Pu^{240} content fuel differed only slightly from the lowest. But subsequent measurements on individual fuel rods gave reactivity worths differing by as much as 25%. Similar measurements on four Hx PuAl rods yield differences a factor of 40 less. Additional measurements of this type are needed for characterizing the fuel being used.

Flux measurements with Lu_2O_3, Pd and Au foils have been made in the 1.8 wt% PuAl core. Additional reactivity measurements on various materials used in critical experiments such as aluminum, Lucite, etc., have been made. Noise measurements for determining β/ℓ are being analyzed.

Additional physics analyses, including temperature coefficients, were computed for the high power density core in the PRTR.

Two proposals for further development of the Phoenix fuel concept were developed: one to perform a burnup experiment in the MTR using high exposure plutonium, and the other to perform a series of beryllium moderated plutonium experiments at Hanford in support of studies on Phoenix fuel application in EBOR by General Atomic.

Studies to determine transplutonium buildup in a 100 MW Zr/H_2O Phoenix core indicated that the following quantities of isotopes in grams would exist in the core after 7174 Equivalent Full Power Hours (EFPH): Am^{241}-301, Am^{243}-1030, Cm^{243}-172, and Cm^{244}-244. These would have about -1.5% effect on reactivity.

Relative merits of PuN, $U^{233}O_2$, and $U^{235}O_2$ as fuels for compact fast reactors are being evaluated. These include critical sizes for various fuel compositions and void fractions, effects of coolant and thermal expansion on reactivity, and reflector control worth.

A parametric survey of critical size and required fuel inventories was performed for a series of PuO_2-stainless steel, fast reactors. These are exploratory analyses for the FFTR.

Considerable effort continued on the preparation of the Hanford Basic Library (cross sections) and its processing code BARNS-II. This system will soon be available for general off-site distribution. A limited first distribution was made to Lockheed Missile and Space Vehicles Company. Modifications to several other Hanford physics codes (HRG, ZODIAC, and HFN) were made to improve their utilization. The RBU code continues to be checked out through theory-experiment correlation studies.

A mockup assembly of the HTLTR has been operated at temperatures up to 1080 C. Based on information obtained from samples inserted in the assembly, the operation was apparently successful. The assembly is now being opened for examination. Tests in a furnace are continuing on materials to be used in the HTLTR. A test was completed in which a matrix of B_4C in graphite was maintained at 1100 C in nitrogen for 800 hr.

Calculations were completed regarding the relative value of bred fuels as enrichment in pressurized water reactors.

Studies of plutonium values as a function of cumulative exposure for different reactor types show a much lower value drop for reactors with low specific power than for those with higher specific power.

Improvements were made in the MELEAGER-ALTHAEA and VESTA computer codes.

Another field trip to Anaktuvuk Pass, Alaska, was completed. This latest trip completes a 1 yr study of the seasonal variation of Cs^{137} body burdens in the residents of this Alaskan village.

The previous calibration of the P^{32} counter was checked by counting a patient from the University of Oregon Medical School who had a known body burden of P^{32}.

In other radiological physics work development continued on the small calorimeter, plutonium counter, charged particle separator for the Van de Graaff, and an optical discrimination technique for application in neutron dosimetry.

A tentative calibration of the atmospheric physics airborne real time sampler was obtained this month. The zinc sulfide particle count to mass relationship was derived from paired RTS-bulk sample data collected in flight in an experiment performed early in the month. Agreement between the flight calibrations and one determined earlier in the year for a ground-based unit is good. Problems previously encountered in obtaining and recording samples with the aircraft were corrected.

3. Chemistry

Studies of uranium metal surface treatment for nickel plating have been concluded. Conditions were defined which result in adherent films.

The transport of polonium from bismuth oxide in flowing argon at 1000 C and a flow rate of 55 ml/min decreased rapidly after about 10% polonium was removed. Preliminary experiments in the removal of metals from bismuth by electromigration were not promising.

Thermal balance studies of the thermal decomposition of the plutonium(IV) oxalate indicate that carbon is produced by the catalytic decomposotion of carbon monoxide.

Little or no volatilization of technetium from 15M nitric acid solutions of ammonium pertechnetate was observed. The solubility of ammonium pertechnetate is not strongly dependent upon the ammonium ion concentration.

In Semiworks process support work (1) HEDTA was found more effective than DTPA in controlling iron, aluminum, and lead in the solvent extraction process, and (2) detrimental concentrations of manganese were discovered in the feed.

It has been found unnecessary to remove cerium prior to ion exchange purification of promethium if the promethium has aged longer than about one half-life (2.6 yr).

The cerium(IV) extraction capacity of degraded D2EHPA-TBP-diluent solvents was significantly improved by treatment with a nitric acid-potassium permanganate wash.

Approximately 12 lb of 2% PuO_2 in UO_2 have been prepared from irradiated fuels by the Salt Cycle process. This is the first part of a 30-lb lot to be remotely refabricated and recycled to the PRTR.

The detrimental effect of graphite on the deposition of plutonium in the Salt Cycle process has been verified in a laboratory experiment in which the presence of AUG grade graphite reduced the enrichment factor from 1.33 to 0.3.

The hot cell glass experiment was terminated near the end of the fifth hot run when catastrophic failure resulted in the loss of the furnace. Essentially all of the program objectives had been accomplished in preceding runs. Acquisition of laboratory data has been completed on the system: phosphate-simulated fission product-process oxide.

In engineering studies a powder mixture containing 70 wt% calcine from 1WW waste, 15% B_2O_3, and 15% P_2O_5 yielded a homogeneous melt which fused at 980 C. This melt contained a higher fraction of waste calcine than any other previously prepared in the current series of studies.

Design verification tests on the Pot Calciner Prototype continue to indicate very satisfactory operation with an induction heating system.

In preparation for tests of thermal stressing of irradiated N-Reactor target elements, changes were made in the experimental arrangement based on earlier tests and a hazards review.

A pilot test of reactor effluent disposal to ground was undertaken at 100-F Area and continued for 7 days.

No significant changes were observed in the gross beta activity levels in the ground water beneath the 200 Area disposal sites during the month.

Laboratory data indicated that use of Linde AW-500 zeolite in place of Decalso as a sorbent for cesium shipments will allow a threefold increase in loading.

The contract for the Containment Systems Experiment reactor simulator vessel was signed; delivery is expected in January, 1966. Erection of the containment vessel is proceeding on schedule. Aerosol development studies included studies comparing the behavior of simulated aerosols in steam-air and air atmospheres.

A critical review of the derivation of the currently accepted equation for the velocity of a particle in a thermal gradient revealed that unlike boundary conditions have been assumed for the particle moving under the influence of the thermal forces and the particle under the influence of the air drag force.

Using newly developed low level counting techniques, metastable Ag^{110} was observed in fallout samples and in Alaskan lichens, and Co^{60} was measured in caribou and in the flesh and bone of whale. Sr^{90} has been measured in fallout samples collected through the first half of 1964.

The final tissue samples from rats exposed in uranium ore inhalation experiments have been analyzed.

Experiments on the effect of chemical protective agents on the irradiation behavior of synthetic aqueous jellies have indicated the mechanism of radiation damage involves hydroxyl radical intermediates.

Shielding calculations have shown that transistors require greater shielding from Pu^{238} sources than from Pm^{147} sources. For personnel exposure the relative amount of shielding required by sources made up from these two isotopes is dependent upon the size of the source.

A Model 1220F Dynapak machine was installed in the 321-A Building and successfully completed acceptance tests.

4. Biology

Nine male sheep were fed 25 μCi Cs^{137}/day for 500 to 600 days. Cs^{137} burdens, as determined by whole-body monitoring, plateaued after approximately 100 days and thereafter varied seasonally with a summer minimum of 14 to 18 times the daily intake and winter maximum of 16 to 24 times the daily intake. This apparent correlation of Cs^{137} body burden

[redacted]

with environmental temperature may be related to underlying seasonal changes in metabolism or diet.

A collaborative study with R. J. M. Fry, of Argonne National Laboratory, has shown that the transit time of cells from the intestinal crypts to the villi tips of bile duct cannulated rats was increased over that shown by control rats. This may be interpreted as an effect of bile salts on cell removal and hence on transit times.

Pulmonary clearance of inhaled $Cr_2{}^{57}O_3$ was tested in dogs that had smoked 20 cigarettes a day, 5 days/week for 11 weeks. No alteration of pulmonary clearance was observed. Similar tests will be made with $Fe_2{}^{59}O_3$, which should provide a more sensitive indication of effect.

From aerial census estimates, about 94,000 ducks and geese were utilizing the Columbia River between the Yakima River mouth and Priest Rapids Dam. This is a substantial increase over the 18,000 estimated last month, but only one-half the number estimated for the same period last year.

The average Cs^{137} body burden in a control group of 24 Anaktuvuk Pass Eskimos was 760 nCi on 15-17 November. This is a decrease of 27% from September measurements.

A technique was developed for the detection of latent fingerprints. This technique involves the use of UO_2 as an alpha source, ZnS as a phosphor and high contrast film to detect the emitted light. Prints were developed from a variety of materials, including wood and paper.

TECHNICAL AND OTHER SERVICES

New exposure record folders and microfilm jackets were prepared for all employees transferring to Battelle-Northwest. Alphabetical and numerical cross index files were also established.

Preparations were completed for the transfer of bioassay and film dosimeter processing functions to the U. S. Testing Company effective January 1, 1965.

[redacted]

The construction of three criticality surveillance instruments was completed. These neutron-sensitive devices, to be installed in the 300 Area, will serve to verify the occurrence of a nuclear excursion if an alarm is received from the gamma-sensitive criticality alarm system. Final installation of these devices is pending receipt of recorders.

Progress was made on computer programming and utilization of existing programs in dose calculations from postulated accidents, in the calculation of reactor excursion transients, and in calculation of radiation level through various types of shielding. The latter program was initiated to examine the adequacy of casks for shipment of fuels and can include many other types of calculations of a similar nature. The postulated accident calculations include anticipated future operating conditions of the PRTR.

There were no new plutonium deposition cases among Hanford employees confirmed by evaluation of bioassay data during the month. The reevaluation of additional data resulted in one employee and two previously terminated employees being reclassified as nondeposition cases. The total number of individuals who have received internal depositions at Hanford is 316, of which 225 are employed at year's end.

A statistical quality control procedure was devised which allows incoming fuel ingots to be weight-checked in sets of three rather than individually.

A preliminary analysis was completed on experimental data from the calibration of the organic photometer to be used to monitor organic uranium stream concentration in the A-column test facility.

An analysis was made and a program written to describe the phenomenon of the spreading of a bounded ultrasonic beam as it travels through layered visco-elastic media.

A linear statistical model was developed for estimating the overall error associated with current determinations of the plutonium content in Pu-Al billets.

Power function calculations were started for the uniformly most powerful unbiased test of the hypothesis that a set of observed counts arose from a pure background uniform intensity source as opposed to the alternative hypothesis that they arose from a decaying source.

A statistical model was formulated in connection with the amounts of intervenously and orally administered Cs^{137} retained by sheep.

SUPPORTING FUNCTIONS

Reactor Plutonium Recycle Test Reactor output for December was 1105 Mwd for an experimental time efficiency of 68% and a plant efficiency of 51%. There were 14 operating periods, four of which were terminated manually, and ten were terminated by scrams (six of the scrams were because of high differential pressure on the rupture loop). A summary of the fuel irradiation program as of December 31, 1964, follows:

	Al-Pu		UO_2		PuO_2-UO_2		Other		Program Totals	
	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd
In-Core	0		6	1717.3	76	14 799.4			82	16 516.7
Maximum				375.5		413.2				
Average				286.2		194.7				
In Basin	7	572.5	27	3078.9	55	7 557.5			89	11 208.9
Buried							1	7.3	1	7.3
Chemical Processing	68	5465.8	35	1965.8	—	—	—	—	103	7 431.6
Program Totals	75	6038.3	68	6762.0	131	22 356.9	1	7.3	275	35 164.5

(Note: Mwd/Element x 20 $\simeq$ Mwd/ton_U for UO_2 and UO_2-PuO_2)

Heavy water loss and indicated helium loss for the month were 963 lb and 128, 564 scf, respectively.

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A total of 76 reactor outage hours were charged to repair. Main items were angle valve repacking, HPHC repairs, 5-L leak repair and rupture loop repairs.

Replacement in the Fuel Element Rupture Testing Facility of the Rupture Loop Temperature Control Valve trim has improved the temperature control significantly. The small differential pressure variation permitted across the test section by the Process Specifications, however, continues to cause problems.

The irradiation of swage compacted UO_2 fuel element No. 1030 with a 6-1/2 in. longitudinal slit was completed with the reactor shut down on December 11. The test element had endured 390 hr of reactor operation and had accumulated 7.8 Mwd or 127.8 Mwd/ton exposure for a total fuel element exposure of 170.8 Mwd. A thorough examination of the element is scheduled in January.

Total productive time in Technical Shops Operation for the period was 24,557 hr. Distribution of time was as follows:

	Manhours	% of Total
N-Reactor Department	1 845	7.5
Irradiation Processing Department	4 587	18.7
Chemical Processing Department	355	1.4
Hanford Laboratories	17 770	72.4

Total productive time in Laboratory Maintenance Operation was 15,900 hr of 18,000 potentially available. Of the total productive time, 95% was expended in support of Hanford Laboratories components, with the remaining 5% directed toward providing service for other HAPO organizations. Manpower utilization (in hours) for December was as follows:

A. Shop Work		2200
B. Maintenance		5200
1. Preventive Maintenance	1300	
2. Emergency or Unscheduled Maintenance	1800	
3. Normal Scheduled Maintenance	2100	
C. R&D Assistance		8500

[redacted]

PRTR heavy water inventory at the end of December 1964 showed a loss of 965 lb valued at $13, 337. Heavy water scrap generated during the month resulted in a $5311 charge to operating costs. Scrap weighing 17, 491 lb was shipped to Savannah River Operations Office during the month. Total scrap on hand at December 31, 1964 amounted to 2819 lb valued at $35, 045.

Cumulative data of Hanford visits:

	Number of Visitors	
	In December	Since 6-13-62
Visitors Center	860	83, 226
Plant Tours	76	---

Professional recruiting activity for December:

	Plant Visits	Offers Extended	Offers Accepted	Offers Rejected	Offers Open
Ph.D.	4	1	-	-	3
BS/MS (Direct Placement)	-	1	1	-	1
BS/MS (Program)	-	5	1	1	4

Authorized funds for 13 active projects total $10, 729, 000. Total estimated cost of these projects is $11, 917, 000. Expenditures through November 20, 1964 were $5, 585, 000.

R S Paul

for Manager, Hanford Laboratories

HM Parker:JEB:dh

REACTOR AND FUELS LABORATORY MONTHLY REPORT

DECEMBER 1964

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 02 PROGRAM

1. Metallic Fuel Development

Comparative Swelling of Uranium Alloy Fuels. Samples of irradiated fuels (KSE-5 fuel elements, 0.31 at% B.U.) containing 140 ppm Fe, 120 ppm Si ("N" composition) and 400 ppm Fe, 800 ppm Al have been post-irradiation annealed to obtain additional information on the influence of additives on swelling behavior. After 100 hours at 625 C, the Fe-Si-bearing fuel swelled an additional 1.1%, whereas the Fe-Al-bearing fuel swelled 0.7%. Optical and electron microscopy of the annealed samples have not been completed, but the volume changes indicate that the Fe-Al additives are more effective in inhibiting the agglomeration of fission gases than the lower level of Fe and Si present in N-fuel. After annealing, the total volume expansion for the Fe-Al and Fe-Si fuels was 1.9% and 3.3%, respectively.

Alternate Uranium Composition. Studies are in progress to determine the effects of altered fuel compositions upon fuel element fabrication, fuel material properties, and irradiation swelling resistance. The previous monthly report described tensile stress cycling tests at 400 C for uranium fuel material containing 150 Fe - 100 Si and 400 Fe - 800 Al. The density was determined on the gage sections of these samples and the change in density shown in Table I. Total strains were not sufficient in these tests to produce density changes large enough for the density measurement to differentiate between samples. Additional samples are being prepared for further testing. Initial voids were observed, primarily resulting from cracked UC inclusions. Examination was also made of the fracture region in samples tested to failure at 400 C.

Void formation was evident in both materials with the major difference being a smaller void size in the 400 Fe - 800 Al material.

TABLE I

Tension Cycling Tests of Uranium Fuel Materials - 400 C Test Temperature

Material	No. Cycles	Nominal Stress 1000 psi	Total Strain % in 1-in.	Density Change, %
U + 150 Fe,	300	25.0 ± 5.0	0.5	-0.35
+ 100 Si	300	27.5 ± 5.0	0.8	-0.26
	14*	30.0 ± 5.0	6.7	-0.36
U + 400 Fe,	300	25.0 ± 5.0	0.2	-0.51
+ 800 Al	300	27.5 ± 5.0	0.2	-0.46
	300	30.0 ± 5.0	0.2	-0.40

*Test terminated.

The thermal diffusivity of samples of these two materials was measured from room temperature to 465 C by D. E. Baker. Thermal conductivity was calculated by multiplying the thermal diffusivity by the density of specific heat. Selected values given in Table II show that the thermal conductivity of these materials is essentially the same.

TABLE II

Thermal Conductivity* of Uranium Fuel Materials

U + 150 Fe + 100 Si		U + 400 Fe + 800 Al	
Temp. °C	Thermal Conductivity cal/cm sec °C	Temp. °C	Thermal Conductivity cal/cm sec °C
27	0.050	27	0.056
272	0.062	268	0.066
465	0.073	460	0.075

A number of N-single tube elements (2.345 OD x 1.364 ID) containing the fuel materials: 150 Fe - 100 Si, 350 Fe - 800 Al, 250 Fe - 350 Si, and 250 Fe - 250 Si - 250 Al, were given both beta and gamma phase heat treatments and samples are being prepared for recrystallization studies and mechanical property testing.

Uranium shot containing both phosphorus and carbon (850-1280 ppm P and 400-550 ppm C) was received from National Lead Co. Metallographic

[redacted]

examination of material with approximately 450 C - 1130 P shows a structure of dendritic primary uranium in a eutectic matrix that occupies 1/2 to 2/3 the area of a field. Individual UC or UP particles in the eutectic are submicron in size. It is interesting to note that the phosphorus addition changes the type of dispersion from a crystallographically oriented pattern in shot containing only carbon to a eutectic distribution in the uranium plus U-UP-UC. The carbide was previously observed in the eutectic structure in slowly cooled phosphorus-containing material. Coextrusion components are being prepared for extrusion of this material to rod stock for additional testing and specimens of the shot have been heat treated to determine the stability of the structure.

Thermal Expansion of "N" Fuels. Measurements of the diametric thermal expansion of the uranium core of an NOE coextruded fuel tube have been made to provide data on the restraint imposed by the cladding. The coextruded fuel tube was beta heat treated and machined free of its inner and outer Zr-2 clad. In the temperature range 20-300 C, the diametric thermal expansion coefficient of the de-clad outer tube ranged from 18.0 to 19.5 x 10^{-6} in/in/°C for three different diameters. Previously determined diametric coefficients of clad fuel tube were 15.0 to 16.0 x 10^{-6} in/in/°C, which is approximately 25% less than for the unrestrained fuel.

Target Element Development. As part of the continuing program of support for the N-Reactor multiproduct target element development, several studies are being conducted on the target materials and target element components.

Two ceramic core targets of $LiAlO_2$ and two of Li_2SiO_3 that were irradiated in KER loops for approximately 58 days have been opened for gas sampling. Results show that only a small fraction of the total gas generated was released from the cores during irradiation. These results are in agreement with those from previous irradiation tests with ceramic cores. Analysis of gas released from post-irradiation annealing and melting tests is planned on sections from the cores.

A series of test samples of 0.739-inch OD x 0.030-inch thick Zr-2 tubes, containing aluminum rods to simulate co-producer target elements, are being tested in autoclaves under N-Reactor temperature and pressure conditions to determine the stability of the tubing with various radial gaps between the Zr-2 tube and the aluminum rod. Measurements after autoclaving for 64 hours showed that two test elements, one with 0.0065-inch and the other with 0.0075-inch radial gaps, had developed a warp of approximately 0.010-inch. No further changes were observed on the two test elements after an additional 60 hours in the autoclaves. Other test elements with radial gaps of 0.013-inch or greater have developed oval cross sections as a result of collapse of the tubing onto the aluminum core.

2. Corrosion and Water Quality Studies

Irradiation of Nickel-Plated Aluminum-Clad Fuel. Three nickel-plated aluminum alloy dummies were irradiated for 30 days in neutral pH water at 260 C. Weight losses on fuel elements were 35-50% of the nickel deposit. Metallography showed considerable attack of the nickel terminating at the inner nickel-aluminum diffusion layer. On dummy elements the plate was nearly intact.

Pressure Bonding Autoclave. Installation of the autoclave is essentially complete and checkout has been started. Remaining work to be done includes hooking up the thermocouple at the autoclave and some seal repair on the gas cooler on the bottom of the autoclave.

The compressor has been checked out and operated up to 10,000 psi and used to fill the accumulators to 2500 psi. Most instruments have been checked out, and only one difficulty has been noted to date. The autoclave has been checked for electrical operation, but it has not been heated or pressurized.

Corrosion of Nickel-Plated Aluminum in Ammoniated Water. Nickel-plated aluminum has been exposed to pH-10 water (NH_4OH) at 25 ft/sec and 330 C for 30 days and continues to show excellent corrosion resistance, much less than in pH-10 lithiated water. Weight changes after 30 days are +7 to 15 mg/dm^2; at neutral pH weight losses of 15-45 mg/dm^2 were incurred for 30 days at 300 C. Corrosion of the aluminum substrate in NH_4OH solution at 300 C is comparable to that in neutral water and one-tenth that in lithiated water.

Intergranular Corrosion of Type 304 Stainless Steel. A test to determine if residual pickling acids could cause intergranular attack in type 304 stainless steel has been concluded. Droplets of HNO_3-HF acid were placed on specimens, the acid was allowed to dry, and a sensitizing heat treatment was performed. The samples were placed in a moist environment. Within four days large amounts of corrosion product formed on the surfaces. Samples were metallographically examined after two to three months of exposure in the moist air. Intergranular attack up to 10 mils in depth was found.

A test to determine if charred vapor phase inhibitor vapors could cause intergranular attack was conducted in two large test assemblies, each with three sensitized, type 304 stainless steel tubing, U-bend attachments. Three grams of VPI and 15 ml H_2O were heated until the VPI decomposed in each assembly five different times over a period of two weeks. After five months of exposure, the tubes were tested for intergranular attack using an eddy current probe. No attack was found. The tubes were split lengthwise for visual examination and no attack was seen. Deposits of the decomposed VPI were present and some staining had occurred.

Reactor Decontamination and Related Corrosion Studies. A six-inch section of N-Reactor steam generator tubing (from steam generator 5B) was decontaminated using the following procedure: the tubing was first cut and split, brushed clean of all smearable contamination, and monitored. A half section of the tubing sample was then cleaned by immersion in alkaline permanganate (10,3) for one hour followed by immersion in Wyandotte 5273-0 (inhibited sodium bisulfate), 10% solution, at 70 C for one hour. Activity was reduced from 15,000 c/m to 450 c/m with less than 100 c/m smearable. Treatment in the acid solution only did not satisfactorily decontaminate a similar sample.

An N-Reactor carbon steel-Zircaloy rolled joint exposed to 16 decontamination cycles has been sectioned for examination. Cursory examination revealed no extensive attack between the CS-Zr-2 joint.

Coolant Decomposition, Graphite Cooling System. During initial operation of N-Reactor, large amounts of H_2 and O_2 resulting from radiolytic decomposition were observed in the relatively low-temperature graphite cooling system coolant. To eliminate gas generation and associated corrosion, it was proposed that the coolant pH be adjusted with ammonia rather than lithium hydroxide since previous loop tests indicated this would result in removal of the accumulated O_2. The change has been made on a test basis by adding sufficient hydrazine to the system to maintain a residual ammonia concentration of 10-14 ppm. In subsequent operation the O_2 concentration has decreased from >15 ppm to <0.05 ppm, confirming the predictions made based on the loop tests. Future plans call for direct addition of ammonium hydroxide instead of hydrazine.

Reactor Decontamination Studies. During recent decontamination activities at D- and N-Reactors, the observed decrease in the activity levels on the Inconel pigtails was very low. Samples of Inconel pigtail samples were obtained to evaluate the effectiveness of three single step decontamination solutions - Sulfam-3, Bisulf-16, and inhibited sulfuric-oxalic acid solutions. The tests were conducted in beakers, and the solutions were stirred vigorously.

The Sulfam-3 and Bisulf-16 solutions were tested at concentrations of 6 oz/gal and temperatures of 70 C. The DF's after 30 minutes were less than three, and the solutions were considered unacceptable. The inhibited sulfuric oxalic mixture was evaluated at 0.3 M and

0.1 M, respectively, and temperatures of 50, 60, 70, and 80 C. This mixture was unsatisfactory at temperatures below 65 C; however, at temperatures of 70 to 80 C, the mixture was quite acceptable, giving DF's of 5 to 11, respectively.

3. Gas-Atmosphere Studies

Carbon Dioxide - Graphite Reaction Studies. The rate of reaction of two samples of TSX graphite with 100% carbon dioxide was found to increase by about a factor of three for burnoffs in the range from 5 to 10%. Surface area changes with percent burnoff were also determined, and it was found the surface reaction rates, i.e., rate of weight loss per unit surface area, were independent of burnoff.

The effect of different pre-oxidation treatments on the activation energy was also studied. All samples were small flat plates, 1 by 0.5 by 0.25 inch. Activation energies were determined in 100% CO_2 in the temperature range from 750 to 900 C. The results may be summarized by the following three rate expressions:

Control sample: $1.48 \times 10^9\ e^{-67,000/RT}\ (hr^{-1})$.

Pre-oxidation in air 13%: $3.70 \times 10^9\ e^{-67,000/RT}\ (hr^{-1})$.

Pre-oxidation in air 9%, plus CO/CO_2, 3%: $2.28 \times 10^5\ e^{-44,700/RT}\ (hr^{-1})$.

Pre-oxidation in air increased the rate of burnoff. Pre-oxidation in air + CO/CO_2 (about 5% CO), not only increased the rate in the temperature range 750-900 C, but also lowered the activation energy.

It is suggested that oxidation with the inhibitor present gave a more uniform oxidation over the samples causing the exposure of more active sites at which subsequent CO_2 oxidation was accelerated.

Graphite Burnout Monitoring. Small graphite burnout monitors were discharged from Channel 1880 at KW-Reactor in December. The burnout profile showed no distinct peaks that would indicate the presence of O_2 or H_2O. The highest measured burnout rate was 1% per 1000 operating days (%/KOD). Monitors enclosed by a 3/8-inch thick wall of graphite exhibited maximum rates of $\sim$0.3%/KOD.

Burnout monitors were discharged from Channel 3461 at B-Reactor in November. The monitors were fabricated from KS graphite used in the B-Reactor stack, and from TSGBF used in the K-Reactor stacks. Burnout rates from KS graphite were higher by a factor of ~3 than the TSGBF rates. The highest measured rate was ~0.6%/KOD. The burnout profile showed only one dome-shaped peak which is attributed to the reaction between graphite and CO_2 and/or H_2O. There was insignificant difference between the rates of enclosed and open samples of TSGBF graphite, whereas the enclosed samples of KS graphite oxidized at approximately one-half the rate of the open samples. Apparently the depth of oxidation was much smaller in KS than in TSGBF.

4. Thermal Hydraulic Studies

Heat Transfer Experiments for N-Reactor. Laboratory heat transfer experiments were continued with a full-scale electrically-heated model of the downstream half of an N-Reactor fuel column and typical outlet piping and fittings. The test section used was identical to one used previously except that provision was made to allow interchannel mixing as would occur at the junction between each fuel element in the reactor. A primary purpose of these latest experiments is to determine the effects of interchannel flow on the thermal and hydraulic behavior of a fuel column.

The following experiments were completed with the second test section:

Zero power runs at 1200 psig; at water temperatures of 250, 370, 450, and 530 F; and flow rates ranging from 45 to 225 gpm.

Runs at 1200 psig and fuel column model powers of 1500, 2000, and 2500 kw (corresponding to powers of 3000, 4000, and 5000 kw in a full-length reactor fuel column). Flow rates in these experiments ranged from 235 to 75.5 gpm, providing outlet coolant conditions ranging from 75 F subcooling to steam qualities of 30%.

Runs at 1310 psig rear riser pressure at a fuel column model power of 2500 kw. Flow rates ranged from 197 to 128 gpm, with coolant outlet conditions ranging from 35 F subcooling to 15% steam quality.

A preliminary examination of the data indicates that the "inter-junction flow" may assist in preventing "burnout," or film boiling, resulting from subchannel instability. Calculations had been made of tube flow rates which would allow film boiling to occur in the model. During the experiments, flows were reduced approximately 20% below these calculated "burnout values" before any indications of film boiling were observed. For the three power levels investigated at 1200 psig, the start of film boiling was indicated at the lowest flows run by temperature increases of surface thermocouples.

Hydraulic Tests for the Once-Through Cooled Production Reactors. Consideration was given to the large oscillations of Panellit gauges that occurred at KW Reactor following the charging of thoria target elements into the central core. Several possibilities for the cause of the oscillation were considered:

> The oscillations were originating in the fuel portion of the tube and the pressure waves were reflected upstream to the pressure monitoring point.
>
> The 0.290 ID orifice in conjunction with the 0.419 venturi produced undesirable pressure fluctuations.
>
> The oscillations were magnified in the 3/16-inch copper sensing line between the Panellit tap and the Panellit gauge.
>
> The magnitude and frequency of the oscillations were such that when imposed on the Panellit gauge, natural frequency resonance occurred.
>
> Under-damping of the Panellit gauges caused inadequate suppression of the pressure signal.

Tests were performed in the Hydraulics Laboratory to investigate some of these possibilities. Tests using a full-length charge of thoria elements in an exact replica of the process tube and fittings used in KW Reactor did not show any abnormal pressure fluctuations which could be attributed to the thoria fuel. Pressure oscillations using the orifice in combination with the venturi were only 10 psi peak-to-peak and would not be of sufficient magnitude to cause a Panellit trip. And, in fact, the pressure oscillations with the venturi alone were greater which gave support to the conclusion that the use of the orifice with the venturi was not the major source of the oscillations at the reactor.

When 100 feet of 3/16-inch copper tubing was added between the pressure tap and the Panellit gauge, it was found that the amplitude of any oscillations was nearly doubled. Existence of air in these lines did not change the amplitude of the oscillations but did decrease the frequency of the oscillations from 60 cycles/sec to 10 cycles/sec.

It was concluded tentatively that pressure oscillations of various amplitudes and frequencies occur at the reactor due to the pumps and shape of piping which do not occur in the laboratory. A combination of these oscillations is seen by the Panellit gauge and causes oscillations of the dial which can be large enough to cause trips if the gauge is not damped. It is felt that when more precise measurements are made of the magnitude and frequency of the oscillations at the reactor, further laboratory tests can be performed to devise the best method of Panellit gauge damping.

In other hydraulic tests, the characteristics of OILIE I&E fuel elements with new arch bumpers were determined for B-D-F-type reactors. A comparison of these bumpers with the present elliptical type bumpered fuel elements indicates that the arch-type elements increase the pressure drop of the fuel charge by approximately 15 psi at operating flows.

5. Shielding Studies

NRD Shield Analysis. Results of the N-Reactor startup shield evaluation experiment have been applied in making shield calculations for full power operation. These calculations are nearing completion, although delayed by a bug in the MAC code. A strong absorbing outer shield region can cause MAC to stop calculation due to a check statement which is contained in the code. The situation has been remedied.

6. Graphite Studies

N-Reactor Graphite Irradiations. A report, "Monitoring Dimensional Changes of Graphite in Ball Channel 60 at N-Reactor," HW-84503, was issued. The report discusses the purpose of the experiments and describes the various sample geometries and measurements.

Samples for dimensional monitoring experiments are in preparation for Rod Channel 74 and will be charged early in 1965.

N-Reactor Distortion Analysis. It was recently suggested (DH Curtiss personal communication) that, if the stresses generated in the N-Reactor core graphite become large enough to cause fracture of the graphite keys, the corners of the keys might shear off at an angle to the moderator layer. Previously, it had been assumed that fracture would propagate parallel to the moderator layers. If the keys fail at some angle, the fracture surface could provide a ramp, and continued contraction of the bars could cause lifting of the upper moderator layers.

Calculations have been made which indicate that the small (1/4-inch high) keys on the process-tube blocks should shear at an angle of approximately 45° when the force reaches about 25% of that necessary to shear the key parallel to the bar. Recent tests run on full-size blocks have confirmed both the mode of failure and the approximate magnitude of the stresses involved. The calculations indicate that the larger (1-1/4 inches high) keys on the filler bars will probably fail parallel to the bars. Tests to check these results are scheduled for the near future.

N-Reactor Irradiations. The irradiation program for N-Reactor graphite continues to progress satisfactorily. The two fifth-generation capsules presently in the GETR, H-4-5 and H-5-5, have successfully completed 2-1/2 and 1/2 cycles of reactor operation, respectively. The maximum neutron exposure on the long term samples in H-5-5 is estimated to be approximately 7×10^{21} nvt, E > 0.18 Mev.

The final fourth-generation capsule, H-6-4, was removed from the GETR and disassembled without difficulty. The capsule operated successfully for 102.2 effective days. Samples and flux monitors are presently being measured.

Calculations are under way on effects of possible changes in reactor loadings on neutron spectra, effective cross sections for flux monitors, and comparisons of computer calculations and flux monitor results. (See Section C.6 on EGCR Graphite Irradiations for details.)

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 High Power Density Core. Two vibrationally-compacted UO_2 - 2 wt% PuO_2 high power density core type fuel elements are currently being irradiated in the PRTR. The number of elements of this type in the core will be increased to eight as quickly as feasible. One of the elements (FE-6000) was irradiated to a burnup of 1470 Mwd/ton_U with maximum fuel temperatures estimated to have been above melting. The element will be destructively examined in Radiometallurgy. It operated for 16 days at a tube power in excess of 1300 kw, with a maximum tube power of approximately 1350 kw for five days (1200 kw is estimated to cause maximum fuel temperatures above melting). Post-irradiation examination of the element in the PRTR basin indicated that its general appearance is good. Removal of two circumferential strip bands revealed that crevice corrosion is occurring at the contact points between the bands and the fuel rod surfaces. The severity of attack (not considered serious) appears comparable to that previously observed on other PRTR fuel elements; however, crevice corrosion could become a problem at the higher heat fluxes expected with the high power density core elements.

PRTR Fuel Elements. Two vipac and one swaged high power density 19-rod clusters having a 2% enrichment in PuO_2 were assembled and delivered to the PRTR. Another 2% PuO_2, 5-foot long cluster is being fabricated in which fuel rods have a modified end cap that will permit removal and replacement of selected fuel rods. Five 1% PuO_2, 8-foot long, 19-rod clusters are now ready for reactor use.

PRTR Fuel Fabrication. A marked decrease in the time required to vibrationally compact fuel into a PRTR tube has resulted when a low frequency impact is superimposed laterally along the tube axis during vertical excitation.

Cladding Procurement. Procurement of 500 PRTR vipac tubes for short core elements has been initiated. Delivery is scheduled in 8-10 weeks. The tubing is being ordered under the new hydride specifications; no more than 25% of the hydride platelets shall be aligned at angles greater than 40° to the tube radius in a transverse section.

Cladding Evaluation. Six PRTR short core rods were rejected during fabrication after ultrasonically examining the area held by the clamp during vibrational compaction. However, the indications of defects, 1-3 mils deep, were determined to be OD marks caused by excessive clamp pressure and not cracking due to excessive shaking.

PRTR Recycle Element. Fabrication of an experimental recycle element (2 wt% PuO_2) is under way. Ten and seven-tenths pounds of irradiated and electrodeposited ~ 2 wt% PuO_2-UO_2 material has been produced by the salt cycle process. Zircaloy-2 components have been fabricated and are ready for assembly. Fuel fabrication equipment has been adapted for hot cell use, and rod fabrication is scheduled for completion during January 1965. Charging of the assembled element in the PRTR rupture loop is scheduled for February 1965.

Slit Defect Fuel Element Performance. A PRTR swage compacted UO_2 element (FE-1030) defected with a 6-1/8" longitudinal slit in one of the rods successfully completed 16 days of operation in the rupture loop. Post-irradiation examination of the defected rod (pre-irradiated to 3270 Mwd/ton_U at a maximum tube power of 1157 kw) in the PRTR basin showed no evidence of fuel washout, waterlogging, or fuel rod swelling. The appearance of the slit was essentially the same as the pre-irradiated condition. The steady state activity release rate from this defect during irradiation was approximately three times greater than the release rate from the 3" long slit and five times greater than the 5/8" long slit in previous experiments. No activity bursts occurred during irradiation and a determination of fission product ratios in the coolant established the release mode to be a diffusion type, i.e., good communication between the fuel and coolant. The defected rod will now be examined in Radiometallurgy.

The next defect experiment to be performed in the rupture loop will be a 3" long slit in a pre-irradiated vibrationally-compacted UO_2 PRTR element.

Rod Replacement. The replacement of individual rods was successfully demonstrated for a simulated PRTR 19-rod cluster. Test element GEH-4-107 had been irradiated for 15 full power days in MTR after which the element was returned to Hanford. One rod of the four-rod cluster was removed and replaced in the water basin at the Radiometallurgy facility. The rods in the test element are identical to PRTR fuel rods except for length and end cap design.

Wire Enrichment. The enrichment of uranium dioxide fuel elements with plutonium wires is being developed. Wires may provide an economical enrichment method with wide versatility. Several feet of Pu-15 wt% Zr wire (0.040-inch and 0.090-inch in diameter) were extruded with the 280-ton extrusion press. The 0.090-inch wire was cold drawn to 0.040-inch diameter without annealing. The average reduction per pass was 10%. New tooling for the extrusion press is being fabricated. Several methods of cladding the wire with zirconium, aluminum, stainless steel, and ceramics are being investigated; clad wire increases the flexibility of fuel fabrication and is of interest in several new fuel concepts.

A new technique for variable enrichment of packed-particle fuels involves the use of a helical plutonium wire, which permits continuous axial variation of fissionable atom concentration by simply changing the helix pitch. The first "Flexi-Twist" plutonium enrichment element was fabricated by winding a 0.040-inch diameter Pu - 15 Zr wire on a 1/8-inch diameter mandrel. An invention report (HWIR-1788) describing the new technique has been prepared.

Heterogeneously Enriched Fuel Element. A fuel element designed to investigate the effect of heterogeneous enrichment failed in the MTR after an exposure of 6×10^{18} fissions/cm^3 (860 Mwd/ton). Examination in the MTR canal indicated that the failure occurred in a rod containing sintered UO_2 pellets around which a mixture of UO_2-PuO_2 had been vibrationally compacted. The appearance of the failure indicates a hot spot type defect. This could have been caused by short circuitry of water flow due to corrosion of the basket by pH 10 water circulated through the loop for a period of 24 hours.

Molten UO_2 Impingement Test. A fuel element designed to simulate conditions of extreme fuel cracking and to allow possible impingement of molten UO_2 on the cladding during reactor startup is being irradiated in the MTR GEH-4 facility. This element is designed to operate with a molten core while the reactor is at full power. A 0.254 cm (0.100 inch) diameter hole was drilled through the diameter of one of the UO_2 fuel pellets to simulate a crack which might be formed during a reactor shutdown. The hole should provide a direct path to the cladding for molten UO_2 during reactor startup. The element has presently operated for four days with the MTR at full power.

Examination of Test Element VBWR-II. Test element VBWR-II, which was a test of a nonfree-standing clad element, was recently returned to Hanford from Vallecitos for examination. The fuel element was comprised of nine rods of 2.43% enriched UO_2 vibrationally compacted to 87% of theoretical density in 304L cladding tubes of 0.008, 0.010, and 0.015 inch thicknesses. The element had been irradiated to an exposure of $\sim$ 1200 Mwd/ton_U at a surface heat flux of $\sim$ 250,000 Btu/hr-ft^2.

Visual examination of external surfaces of the fuel rods of the test element revealed no major cladding deformation or obvious corrosion. Three rods were selected for sectioning. The fuel showed no unusual changes, and no corrosion was visible on the internal surfaces of the cladding.

Hydriding of PRTR Fuel Rod Cladding. The laboratory hydride test capsule design was modified to include both a conventional vipac end cap and a swageable end cap. Two nonfueled Zircaloy capsules of this design were exposed to 310-320 C hydrogen. The first capsule failed in the crevice region of the vipac end cap after 40 hours of exposure at 320 C. Sectioning and microscopic examination revealed no significant hydride formation in the swageable end cap region. The second capsule did not fail during a 48-hour test at 310 C. However, microscopic examination of polished sections disclosed hydride formation at the crevice top in the vipac end cap region and essentially no hydride formation in the swageable end cap region.

Corrosion and Water Quality Studies

Zircaloy-2 Hydrogen Pickup. Hydrogen pickup of Zircaloy-2 has been studied over a wide range of oxygen to water partial pressure ratios at 400 C. Except at very low oxygen concentrations in high pressure steam, hydrogen pickup is largely independent of oxygen to water vapor concentration ratios.

Stress Analysis. A numerical method of analysis for thermal stresses in cylinders formed by a grid of triangular and rectangular rings has been formulated and programmed. The temperatures and material properties of each ring element can be independently prescribed. The method of analysis was initiated to calculate the stress conditions in bonded metallic cylindrical fuel elements with shaped end closures. Since this geometry is very general, the program has the flexibility to determine the thermal stresses in most proposed designs for transition joints between pipes formed from dissimilar materials. This method of analysis can be extended to consider stress relaxation.

Ceramic Washout Tests. A mixed oxide core element irradiated in PRTR has released a small amount of core material in ex-reactor core washout tests. The third in a series of tests to determine washout behavior (release of core material) was conducted on a 12-inch length of a Vipac-Nupac PRTR rod, Zr-2 clad with a core of UO_2 - 1% PuO_2, irradiated to 1890 Mwd/ton. The element was predefected with a slit 6" long by 1/16" wide and tested ex-reactor at 300 C, 1600 psi, and 13 fps. The test has been run for 1-1/2 weeks and cycled in temperature to 40 C three times.

In each instance an increase in loop filter activity of 100 mr/hr resulted. This activity increase is in contrast to tests run for 3-1/2 weeks with other elements which resulted in no washout. The first of these other rods was a Vipac-physical mixture irradiated to 5000 Mwd/ton, and a core of UO_2 - 1% PuO_2. The second rod was a swaged-physical mixture irradiated to 4100 Mwd/ton and a UO_2 - 1/2% PuO_2 core. Both rods were Zr-2 clad and defected with a slit 6" long and 1/16" wide.

Automated Continuous Analysis of Borate. Laboratory work on the H_3BO_3 analytical procedure in water was completed. This procedure uses a dilute aqueous solution of carminic acid as the indicator in a colorimetric reaction that occurs in basic solution. The procedure thus represents a significant improvement from the standpoint of safety over other common procedures using concentrated acid as the indicator solvent. The procedure is based on decolorization of the indicator in the presence of H_3BO_3 rather than formation of a stable colored reaction product--thus the procedure is sensitive to any mechanism that results in variation of the indicator concentration. Tests with the Technicon Autoanalyzer have demonstrated that the procedure is sensitive to boron concentrations as low as 0.1 ppm and is generally reproducible to within $\pm$ 5% of the indicated value. These values are satisfactory for our requirements on this project, and no further development work is planned.

The procedure was also tested using actual PRTR moderator samples spiked with H_3BO_3 and impurities to determine whether any interferences would be anticipated during testing at PRTR. Results indicated no apparent interference in the moderator samples as received, or spiked with 10 ppm NO_3^- or 10 ppm HCO_3^-. Since these are the only major impurities normally encountered in the moderator (result from the air in-leakage), no problems are anticipated at PRTR.

Boron Deposition Studies. H_3BO_3 deposition studies are being conducted under dynamic conditions in a small glass loop. These tests utilize a relatively large metal surface area-to-liquid volume ratio, and the boron deposition is monitored by observing changes in the H_3BO_3 concentration in the liquid. Results have been obtained for tests conducted with clean and filmed specimens of 6063 aluminum exposed to 100 ppm boron solutions at room temperature for one week. There was no significant effect due to surface condition since an equilibrium deposition value of about 3×10^{-6} gm boron/cm^2 aluminum was observed in each case. Indications are that 2-4 days are required for equilibrium deposits to form. This work will continue to check effects of variations in solution temperature and boron concentration as well as evaluating other materials. Desorption studies are also planned to evaluate factors affecting release of the deposited boron from the metal surfaces.

Prototype IX System for H_3BO_3 Tests. Anion resin capacity studies as a function of H_3BO_3 concentration in solution were completed. These tests were conducted under static conditions and verified the values used to design the prototype ion exchange system for the H_3BO_3 tests at PRTR.

Decontamination of PRTR. HW-84547, "Decontamination of the Plutonium Recycle Test Reactor," has been issued; the report specifies the procedure and chemicals required for decontamination of the PRTR primary system. Such items as chemical purity, gasket replacement, and backflushing requirements are also discussed. It is planned to decontaminate the IRP Loop (the ex-reactor loop used in plutonium fuel washout studies) using the procedure specified for the PRTR primary system.

Examination of PRTR Crud Films. As indicated in the report for November 1964, a section of PRTR pressure tube was decrudded with 6N HCl to determine the quantity of crud present. Analysis of decrudding solutions indicated that total crud deposition was 9.3 mg/dm^2. No measurements of the crud film thickness were made, but film thickness was calculated using the total deposition value and the following formula:

$$\text{Film thickness (cm)} = \frac{\text{Crud deposition (mg/dm}^2 \times 10^{-5})}{\text{Film density (g/cm}^3) \times \text{(solid fraction)}}$$

where the solid fraction is 1 minus the porosity. Assuming a porosity of 0.7 yields a calculated film thickness of 0.024 mil. A film thickness of this magnitude should not present any pressure drop problems. It should be noted the use of a value of 0.7 for porosity is probably quite conservative.

Reactor Engineering Development

Fretting Corrosion. The effect on fretting damage of radial clearance between a PRTR fuel element and pressure tube was demonstrated in an ex-reactor test (TF-7 Loop). A full-sized UO_2, 19-rod fuel element was subjected to externally excited vibration (3.5 to 4.5 mils in amplitude at 27 cps in the N-S plane) for 21 days at 123 gpm flow and 530 F. All end brackets (centering feet) were shortened radially by about 0.075 inch, increasing the radial clearance from ~0.050 inch to ~0.12 inch. The fuel element was installed so that the 1/16 inch wide (unsharpened) centering feet contacted the liner at S, NW, and NE locations. Examination after the run revealed liner marks six to seven mils deep opposite the S bottom end bracket centering foot. This mark is about three times deeper than those found prior to tests at similar conditions except for reduced clearance. Numerous marks, up to 9 mils deep, were found where wire wrap contacted the liner. In general, the deepest marks were found on the N and S sides of the liner (in the plane of the excitation) and were formed in an area 45 to 85 inches below the top of the vertical test section. The vibrator is installed 60 inches from the top. Fuel element wire wraps were severely worn at numerous contact locations.

The present fretting corrosion test with the prototypical high power density fuel element at 185 gpm was temporarily suspended after 304 hours due to excessive pump seal leakage.

Data from the test to determine the effect of PRTR HX-1 steam pressure on fretting have been completely reduced, and detailed analyses are being conducted. Analyses to date have shown only two predominant frequency components: the first is noted with a filter where the center frequency is 30.5 cps; the second occurs where the filter center frequency is 35.5 cps.

Data have been obtained to determine the effect of the injection pump on fretting in PRTR. Visual oscilloscope observations indicate no significant variations with an injection pump on or off.

PRTR Shim Rod Development. Installation of the second generation assembly in the reactor was completed. Initial operation of the assembly was satisfactory, but after 48 hours of reactor operation, difficulties with both "A" and "B" rods were encountered. "A" rod motor had two fuses blown, and "B" rod drive ran erratically in the up-direction. After the fuses were replaced in the "A" motor circuit, the "A" drive has operated satisfactorily.

During the first outage after installation, the driving head was removed and the lead screws were turned by hand. One screw in the "B" system exhibited significant "roughness." This screw was exercised by hand, but continued "roughness" was noted. No significant change in operation has been observed since. Samples have been taken of the moderator water to determine if particulate matter lodging in the ball-nut is the cause of the erratic operation.

A first generation Mark II assembly was used in additional investigation of the lead screw sticking problem. It had been postulated that residual lubrication washoff or breakdown (from radiation) might be the cause of rod sticking. The assembly was degreased in trichlorethylene and then installed in the environmental test facility. After 21 hours, the "A" rod jammed and would not operate either up or down. Visual examination of the components revealed galling of the lead screw and pickup of foreign material on the balls, probably aluminum and aluminum oxide.

The assembly, without the "A" ball screw reassembled, was recleaned and reinstalled in the facility. After 27 hours, the "B" rod stuck; however, in this case manipulation of the switch freed it.

PRTR-HPD Core. The final draft of HW-84449, "Transition Schedule--PRTR High Power Density Core," was completed. The schedule adopted is a 15-month transition to a batch load experiment. This transition has a target completion date of January 1966.

Some additional calculations were performed to check some of the assumed parameters in the fuel temperature study. The effect of changing the conductance of the fuel clad interface on the predicted size of the molten fuel zone was investigated. It was found that existing uncertainties in the fuel-clad interface conductance can change the size of the molten fuel zone, but not enough to cause concern at this time. The assumed coolant heat transfer coefficient is so close to calculated values that its effect on molten fuel zone dimensions is negligible.

Temperatures calculated using the fuel thermal conductivity used in Hanford Laboratories were compared with those calculated using other information published in the literature. There was very little difference in the calculated molten fuel zone dimensions.

Fuel Re-use. A revised draft of the Second Progress Report on the Fuel Re-use concept (HW-81282) was completed. Certain burnup calculations are being repeated to permit more exact and extensive (higher value) cost calculations. These will be reduced to graph form and final text changes will be made in early January.

Preliminary scoping calculations for a fuel re-use demonstration experiment indicated that, based on an assumed availability of space in a fast reactor, an average buildup of 1% Pu-239 (maximum buildup of 1.5% Pu-299) can be achieved in the depleted UO_2 "interchange" elements by December 1966 and 2% Pu-239 (average) might be achieved by February 1968--assuming irradiation in an inner blanket location.

Thermal Hydraulics Studies. A brief survey was made of the methods of determining the boiling burnout limits for nuclear reactors. The survey was made to provide a comparison with a method of calculating boiling burnout limits for the PRTR, particularly, with respect to the boiling burnout safety factor.

It is difficult to make a comparison between methods of establishing boiling burnout limits. This is due, in part, to the fact that the sources of information were usually hazards evaluation reports and the descriptions of the methods of determining boiling burnout limits were brief summaries. Also, the problem of determining boiling burnout potential of a fuel element is very complex, and the various people determining boiling burnout limits chose different methods of attack and different simplifying assumptions.

However, some general conclusions may be made from the survey. First, the hazards analyses often demonstrate that boiling burnout will not occur with operation at the design conditions. The implication of this is that possibility of boiling burnout is not a limit for these reactors.

Safety factors for reactors presently operating ranged from 1.4 to 2.0. In general, these safety factors are the ratio of the boiling burnout heat flux to the operating heat flux at the point in the reactor closest to boiling burnout. The analyses use various hot channel factors to determine that point in the reactor closest to boiling burnout. The boiling burnout heat fluxes are sometimes determined from data, sometimes from design correlations which are established at the lower edge of the experimental scatter, and sometimes of boiling burnout correlations which are established as the best fit of experimental data. The boiling burnout safety factors, then, are not only margins of safety but also are intended to account for uncertainties in the boiling burnout relation used.

Indicative of the future is the fact that two conceptual design studies used boiling burnout ratios of 1.5 and 1.25. These may reflect the belief that current boiling burnout analyses are unnecessarily conservative.

For comparison, the boiling burnout analysis being prepared for the PRTR defines the boiling burnout safety factor to be strictly a margin of safety. All uncertainties, including those of the boiling burnout relation, are to be accounted for separately. For the 19-rod bundle fuel element of the PRTR high power density core, a boiling burnout safety factor of 1.33 is proposed.

It was concluded from the survey that a standard method of treating the boiling burnout safety factor does not exist and that a comparison of burnout safety factors which are stated for various reactors in the United States is not completely meaningful.

2. Plutonium Ceramics Research

PuO_2 Stainless Steel Cermets. Four capsules containing stainless steel-20 vol% PuO_2 cermet pellets were fabricated for irradiation in the ETR. Cermet matrix material was prepared by mixing -325 mesh stainless steel powder with -60 +325 mesh PuO_2. High density pellets (97% TD) were made by pneumatically impacting the stainless steel PuO_2 powder. Low density pellets (83% TD) were prepared by cold pressing the stainless steel-PuO_2 mixture at 45,000 psi. These capsules are designed to operate at a power generation of 3.85 kw/inch and a maximum core temperature of 1300 C.

The Am-O System. Earlier attempts to measure dissociation pressures of AmO_2 failed because the americium alloyed with platinum sample holders. Recent experiments showed both AmO_2 and PuO_2 to be compatible with iridium at 1100 C in hydrogen. A set of 3/4" diameter, 0.005" thick iridium container pans was successfully hot spun.

Ninety percent of the two grams of AmO_2 which had previously reacted with platinum containers was recovered. The procedure involved dissolution in aqua regia and treatment with formic acid, NH_4Cl, NH_4OH, HNO_3, and $H_2C_2O_4$. The oxalate was converted to the dioxide by heating to 800 C in oxygen.

PuN Studies. PuN was prepared by carbon reduction of PuO_2 under 600 mm of nitrogen. This method of preparation is considerably simpler than the metal-hydride-nitride route.

PuN-ZrN Phase Studies. A Pu-31.6 at% Zr alloy was hydrided and subsequently nitrided to form a single-phase (Pu,Zr)N. This mixed nitride is expected to be stable to a very high temperature under one atmosphere of nitrogen since the components, PuN and ZrN, are stable to 2550 and 2950 C, respectively.

High Temperature X-Ray Diffraction. An improved vacuum system is being constructed to fit the existing hooded MRC high temperature x-ray diffraction attachment. The system should allow operation at 2000 to 2200 C by prolonging life of refractory metal heating elements. A special electrode system is being built to accommodate cermet wafer samples for high temperature analysis. Studies of phase transitions in the Pu-B system and of precipitation of metallic uranium from UO_2 will be undertaken on completion of the modifications.

3. Ceramics (Uranium) Fuel Research

High Temperature Oxygen-Uranium Phase Diagram. Study of the oxygen-uranium system at high temperatures in the composition range $UO_{1.75}$ to $UO_{2.0}$ revealed (a) a decrease of as much as 200 C in melting point with a decrease in O:U below 2.00, and (b) the existence of a single phase region above 2400 C at an O:U ratio of 1.75. A tentative high temperature phase diagram was established.

A decrease in melting point of as much as 200 C was observed in $UO_{2.x}$ as x was increased from zero to 0.25.

UO_2 Tube-in-Tube Element. A vibrationally compacted UO_2 fuel element with tube-in-tube geometry failed in the M-3 loop of the ETR after an exposure of 3×10^{18} fissions/cm^3. The element failed while the reactor was recovering from a spurious scram. Examination in the ETR canal did not reveal the type or location of the failure. This element will be destructively examined in the Radiometallurgy facility at Hanford.

Electrical Properties Irradiation Capsule. A capsule designed to allow measurement of the electrical resistivity and the Seebeck emf of UO_2 while in a neutron flux was irradiated. The UO_2 pellets were tungsten clad, and the capsule was vacuum insulated. The temperature of the vacuum insulated, tungsten cladding was monitored with a W-Re thermocouple. UO_2 resistance and Seebeck emf were measured between the cladding and a central tungsten electrode. The maximum clad temperature attained during the irradiation was approximately 1800 C.

The electrical resistivity decreased, initially, with increasing temperature with an activation energy for conduction of 0.40 ev. An unexplained reversal in resistivity and increase with increasing temperature was observed at high temperature. The Seebeck emf measurements indicated p-type conduction in UO_2 below approximately 1300 C and n-type conduction at higher temperatures.

<u>Spheroidizing UO_2</u>. Spherical UO_2, formed by agglomerating fines on seed particles and sintering in hydrogen, was incorporated into 25 wt% UO_2-stainless steel cermets. Pneumatic impaction was used to consolidate the cermets and extrusion was used to form the cermet into rods with an L/D ratio of 180. The performance of the 93-96% dense spheroidized particles was excellent, and no stringering was observed during the fabrication processes.

Plutonium dioxide prepared by rapid oxidation of plutonium metal has

(1) Excellent metallurgical bonding of end caps, cladding, and metal matrices,

(2) Variable ceramic and metal matrix densities, i.e., high matrix densities with low ceramic particle densities, vice versa, and various combinations of ceramic and metal densities,

(3) Reduced grain growth due to relatively low bonding temperatures and forging action of pneumatic impaction process,

(4) Flexibility in choice of components: refractory metals, high temperature metals and alloys, ceramics, radioactive materials, compounds, and additives,

(5) Flexibility in the physical shape and form of starting materials: loose powder, pre-pressed shapes, coatings such as by vapor deposition or plasma spraying, wrought metal, and vibrational compacted particles,

(6) Uniformly distributed ceramic particles in metal matrix,

(7) Little or no stringering or fracturing of ceramic particles during fabrication.

Cermets have been fabricated of SS-UO_2, Mo-UO_2, Nb-UO_2, and W-UO_2.

Materials and Information Exchange. Single crystal specimens of UO_2 and ThO_2 were prepared, characterized, and shipped to University of California at Los Angeles, and to General Atomic, respectively, for research studies. Approximately 3 kilograms of PuO_2 were received from ANL to be high energy-rate impacted, characterized, and returned for basic studies.

Status of 10^6 Ampere Welder. The contractor has completed fabrication of the welder and is now awaiting delivery of control components. Shop testing of the welder should begin during late January or early February, dependent on prompt delivery of controls.

The welder power supply project proposal is still awaiting AEC approval. During the month additional analysis of the possible effects of this supply on the 300 Area and adjacent BPA-served areas was furnished the AEC as a supplement to the project proposal.

4. Basic Swelling Studies

Irradiation Program. The second controlled pressure-temperature swelling capsule containing high purity uranium, U + Fe - Al and U + Fe - Si specimens continues to operate successfully at 1000 psi and 575 C. A third pressurized capsule is being assembled for operation in-reactor at 1000 psi and 625 C. The next series of capsules is scheduled for lower pressure operation.

Post-irradiation Examination. Recent experiments emphasize the need for precise pre- and post-irradiation density values of uranium specimens because of the small amount of swelling that is currently being observed. Consequently, a one-pan analytical substitution type balance, modified to weigh samples in air and in liquid beneath the balance case, is being used to determine density of unirradiated samples to an accuracy of $\pm$ 0.01 g/cm^3. This equipment, of course, will be available for use on materials other than uranium.

Electron microscope studies of replicas made from the longitudinal section of a 0.50-inch diameter rod of high purity uranium irradiated to 0.05 at% B.U. have been completed. During irradiation one corner of the sample operated in the alpha phase while the greater portion was in the beta temperature region. The microstructure of that portion of the sample operating in the alpha range revealed the original, small equiaxed grains with 0.2 micron diameter pores at their boundaries. The adjacent area now consists of large columnar alpha grains; a network of small pores exists which outlines the original alpha grains. The portion which operated very high in the beta phase and away from the corner of the sample has pores which are approximately 0.3 micron in diameter and are uniformly dispersed throughout the existing alpha grains. In some cases pores form segments of previous boundaries. In the core a prior set of boundaries is outlined by rows of pores which probably locate old gamma boundaries. The columnar grain structure ends abruptly in this region, and a large number of cracks believed to have formed when the sample cooled are present. Pores found in the core are larger in diameter, 0.75 micron, and exist in a much higher density than in any other section of the sample. Existing grain boundaries are free from pores. Denuding of pores adjacent to cracks was not observed. This suggests that gas mobility is still low despite the high operating temperature.

Supplemental Studies. A measurement of the breadths of the {111} and {112} x-ray diffraction peaks in irradiated uranium as a function of post-irradiation annealing temperature has been made. Anomalous behavior is observed in that, after initially decreasing, the peak breadth increases at approximately 500 C and then subsequently decreases again upon annealing up to 650 C. It is tentatively proposed that some lattice strain inducing phenomenon, possibly associated with gas pore formation, is occurring.

5. Irradiation Damage to Reactor Metals

Alloy Selection

Several nickel-base alloys are being studied to determine the effect of irradiation and environment upon their mechanical properties. Fourteen tensile tests were completed at 650 C on Inconel 600, Inconel 625, Inconel 718, and Incoloy 800 specimens irradiated at exposures to 8.3×10^{20} nvt fast. Results of these tests show a marked decrease in ductility of all alloys when tested at 640 C as compared to room temperature tests. Strengths of these alloys are considerably lower than the room temperature strengths of specimens irradiated to similar exposures. Control tests for these specimens are currently being conducted.

Density measurements for specimens of Inconel 600 irradiated to an exposure of 9×10^{19} nvt fast at 740 C have been made. Similar measurements were made on an ex-reactor control specimen given a similar thermal history. The irradiated and nonirradiated control specimens had densities of 8.41 and 8.51 grams/cc. After an annealing treatment of 1050 C for 24 hours, the density measurements were 8.13 and 8.55 grams/cc, respectively. This represents about a 3% decrease in density for the irradiated specimen. It is postulated that this decrease is the result of the agglomeration of helium atoms in the metal formed from the (n, alpha) reaction with boron. Further metallographic examination will continue in an attempt to verify these results.

Chromium Alloy Investigation

Efforts to alter the microstructure of the experimental chromium alloy (Cr, 0.7 Y, 0.7 Ti, 1.42 Zr) through various heat treatments have met with some degree of success. Carbide precipitates have been reduced in size and to some degree excluded from the grain boundaries.

Several wafers of the alloy have been heat treated at 1400, 1500, and 1600 C, and then water quenched or air cooled. Microstructure studies indicate that a fine carbide precipitate is held in the matrix and out of the grain boundaries by quenching from the treatment temperature.

In-Reactor Measurements of Mechanical Properties

An in-reactor creep test on annealed 304 stainless steel at 550 C and 30,000 psi has been started. Sufficient test time has not accumulated to determine an in-reactor creep rate. An ex-reactor test at these conditions exhibited a creep rate of 2.7 x 10^{-5}/hr after 300 hours.

A specimen recovered from a creep capsule after in-reactor creep testing was measured and found to have an elongation of 9 to 11% on breaking. The specimen had been creep tested at 600 C, 20,000 psi for 800 hours and subsequently used for activation energy studies during which time the specimen broke.

During the past month the design for a prototype high temperature in-reactor creep capsule has been completed, and construction has been started. Operation of the capsule at temperatures as high as 3500 F and stress as high as 45,000 psi is anticipated. Low voltage, high current molybdenum or tungsten heating elements will be used to obtain test temperatures. Capsule atmosphere may be vacuum or inert gas as desired.

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.

Two of four special quadrants containing Inconel 600 - Zircaloy-2 transition tensile specimens have been irradiated and discharged from the G-7 hot water loop after exposures between 1.0 and 1.2 x 10^{20} nvt (> 1 Mev). These specimens are to be examined and tested in support of the Advanced BONUS Core Development Program being conducted by Combustion Engineering, Inc. Two quadrants containing 12 specimens were charged into the out-of-reactor loop to receive exposures similar to those obtained in the G-7 loop. Both the irradiated and control specimens will be tested at room temperature and at 550 F. Special testing fixtures are being fabricated to test these tube section tensile specimens.

Fracture tests on annealed Zircaloy-2 and A-302B have been performed using the DCB (Double Cantilever Beam) specimen. Reduction of the data into meaningful information was made possible by performing calibration tests on specimens of identical geometry (but made of

7075-T6Al) to establish the relationship between crack length, applied force, and extension of the specimen arms. The ductile-to-brittle transition temperatures for the Zircaloy-2 and the A-302B have been found to be -130 C and -40 C, respectively. The fracture toughness values below the transition temperature agree with those obtained from much larger specimens, indicating the applicability of the DCB test irradiation effects studies. Hydrided specimens of Zircaloy-2 containing 100 ppm and 400 ppm have been received and are waiting to be tested to determine the extent of hydrogen embrittlement in Zr-2. A cryostat has been developed to facilitate remote testing of irradiated DCB specimens.

Irradiation Effects in Nickel-Base Alloys

Optical metallographic examinations of specimens of Inconel X-750 stressed to rupture at 1350 F show that transverse sectioning of the fracture area reveals microcracking to a larger degree than does longitudinal sectioning. These microcracks have been intergranular in nature even with rupture times at 1350 F as low as 10 hours.

Investigation of quality control techniques was initiated. During the initial testing and setting up of necessary equipment, two of eight rods of stock material of Inconel X-750 to be used for specimen fabrication were found to cause attenuation of the ultrasonic radiation to a greater extent than the others. Optical metallographic examination of a specimen from each batch revealed that the rods exhibiting ultrasonic attenuation had generally larger and nonuniform grains than the others (ASTM 5 vs 8). Such grain size differences could have large effects upon tensile and stress to rupture test results. Development of these quality control procedures will continue.

Environmental Effects

Analysis of zirconium alloy specimens exposed to 280 C low oxygen (hydrogen addition) water in G-7 ETR loop continues. Comparisons of corrosion and hydrogen pickup among Zircaloy-2, Zircaloy-2 low nickel, Zircaloy-4 and Zircaloy-3% Nb - 1% Sn in-reactor and out indicate the following: the ratio of weight gain in-reactor to weight gain out-of-reactor increased from a factor of 2 at the lowest flux level (6×10^{12} nv) to a factor of 5 at the highest flux (6×10^{13} nv fast) except for Zircaloy-2 Nb-1 Sn which showed no significantly accelerated corrosion at a flux less than 6×10^{13} nv (>1 Mev). At 6×10^{13} nv the maximum weight gain ratio for Zircaloy-3 Nb-1 Sn was 2.

Hydrogen pickup fractions were generally higher for all irradiated specimens. No consistent or significant effect of flux intensity on hydrogen pickup was observed in the case of Zircaloy-2 specimens, but the Zircaloy-4 and low-nickel Zircaloy-2 alloys showed marked increases in pickup fractions with increasing flux intensity. Zr-3 Nb-1 Sn proved to be the best alloy in this respect, also, with no significant change in the hydrogen pickup fraction which remained at the level of about 15%. In-reactor hydrogen pickup fractions for the other alloys exceed 34% in most cases.

Cautious comparisons can be made of the relative effects of dissolved O_2 versus dissolved hydrogen in the water environment at high fast neutron flux on weight gain and hydrogen pickup for Zircaloy-2. Under fast neutron irradiation at flux intensities on the order of 1×10^{13} nv and during the pretransition period, weight gain rates are higher by a factor of at least two and hydrogen pickup fractions are lower by a factor of about one-half for 1 ppm oxygen in the water environment as compared with less than 0.1 ppm oxygen and hydrogen in excess of 10 cc per kilogram.

ATR Gas Loop Studies

Helium Purification System. The model gas loop high pressure helium purification system was activated in December 1964, after a vendor representative from Engelhard, Inc., had worked two weeks on the system. In the acceptance run, helium with less than 0.1 ppm of H_2, N_2, O_2, CH_4, CO, and CO_2 was produced. A moisture content of 2 ppm was believed to be due to moisture on the walls of the sample lines.

A bypass purification system for the model loop, consisting of a simple liquid nitrogen freeze-out system powered by the pressure rise across the gas bearing blower, produced a loop atmosphere low in all impurities except 1.2 ppm H_2, 3.3 ppm N_2, and 12 ppm H_2O. An improved system nearly ready for installation consists of a charcoal trap followed by a 13X molecular sieve trap, both at liquid nitrogen temperature.

Model Gas Loop Heater Design. Design calculations on the second generation helium gas heater for the model gas loop are 95% complete, and detail drawings are 15% complete. The heater is being designed to incorporate and investigate salient design details of the compact two-stage heater presently considered for the ATR gas loop. Most major components have been submitted for suppliers' bids. Some material is on site, and fabrication will begin upon completion of the detail drawings.

Dynamic Materials Testing Apparatus. The dynamic materials testing apparatus is in generally good operating condition. Difficulties have been experienced in seating the alumina test section in the platinum alloy holder, resulting in a ΔT across the test section of about 100 F. A leak in the diaphragm of the compressor resulted in oil contamination of the gas stream.

After 189 hours of exposure in untreated Grade A helium at 2050 F and 350 psi, Haynes 25 Sample No. 2 showed an over-all weight loss of 5.4 mg/cm^2, even though the sample bore an oxide film. During the second run with this sample, the diaphragm leaked oil into the gas stream. The sample was thoroughly carburized and showed an over-all weight gain. The loop system was cleaned by circulating air-contaminated helium, while the test section (without sample) was maintained at high temperature; the carburized sample was then returned for a 406-hour exposure. The results have not yet been analyzed.

Oxygen Analyzer. The calcia stabilized zirconia tube oxygen monitor has been modified to use air as a reference gas. This new probe produced voltages as predicted by the theoretical equation:

$$EMF = \frac{RT}{4F} \ln \frac{2.1 \times 10^5}{P_{O_2}}$$

where P_{O_2} is the oxygen content of the gas in ppm. The revised probe is now being incorporated into a portable instrument.

Procurement Specifications. Technical procurement specifications for the ATR chromatograph and total impurity analyzer have been prepared and sent to Ebasco Services, Inc., for comments. It is expected that procurement of these instruments will start about January 4, 1965.

High Temperature Creep-Rupture Tests. Five additional tests have been run using the new high temperature creep-rupture apparatus, with the following results:

Material	Temp.	Stress	Minimum Creep Rate	% Elong.	Time to Rupture
Haynes 25	2100 F	1500 psi	7.0×10^{-3}/hr	11.7	12.0 hr
Haynes 25	"	1000 "	1.5×10^{-3}/hr	8.4	33.25 "
Hastelloy X	"	3000 "	4.0×10^{-1}/hr	21.9	0.46 "
Hastelloy X	"	2200 "	1.4×10^{-1}/hr	18.5	1.07 "
Hastelloy X	"	1800 "	1.02×10^{-1}/hr	21.3	1.85 "

The new Haynes 25 data combined with the previous data show a more pessimistic rupture life at low stresses than was predicted from earlier work using a dead weight loading system and induction heating. Additional tests at low stresses are planned to see if this observation is correct.

A review of the literature shows stress-rupture data for Hastelloy X only up to temperatures of 1800 F. Tests are planned to determine the creep-rupture properties of Hastelloy X in the range 1800 to 2200 F.

6. Nuclear Graphite Studies

Thermal Reaction of Graphite with Oxygen and with Water Vapor. An experimental system has been constructed to investigate the graphite-oxygen and graphite - water vapor reactions under a wide range of experimental conditions. The system includes a Cahn electrobalance to follow the sample weight as a function of time. With a two-gram sample (the maximum load for this balance) a sensitivity of about 10^{-6} appears possible, and hence extremely low rates of oxidation can be measured. The entire system including the balance can be readily evacuated to 10^{-5} torr, thereby allowing careful outgassing of the graphite sample before oxidation.

When oxygen is employed as the oxidant, it is supplied from a cylinder containing a mixture of helium plus 1% oxygen. The mixture is passed through a 5A molecular-sieve column, a rotameter, and then into the reaction chamber. Oxygen concentrations down to 200 vpm are obtained by diluting with purified helium. Purification is achieved by passage over hot copper turnings to remove traces of oxygen, hot cupric oxide to oxidize any hydrogen and carbon monoxide present to water and carbon dioxide, respectively, and, finally, 5A molecular sieve to remove any water and carbon dioxide formed.

When water vapor is the oxidant, it will be carried by the purified helium. Part of the purified gas is diverted through a trap containing water at 0°C. By adjusting the flow rate and the amount diverted, water-vapor concentrations in the range of about 50 to 6000 vpm can be obtained.

Initial measurements have been concerned with the graphite - oxygen reaction. A TSX graphite sample in the form of a parallelepiped 20 by 8 by 2 mm was employed. Its initial weight after a preliminary oxidation of about 1% was approximately 480 mg. Its rate of oxidation over the temperature range 497 to 630 C by the equation

$$\text{Rate (hr}^{-1}) = 1.79 \times 10^{9}\ e^{-49.9 \times 10^{3}/RT}.$$

The activation energy, 49.9 kcal/mole, agrees well with values published by other investigators. At 630 C, the reaction order with respect to the oxygen concentration was found to be 0.47 over the concentration range 0.2 to 1%.

Similar measurements are presently being performed using a sample of graphite containing approximately 5% boron by weight.

Gamma Irradiation Facility. An additional 57,000 curies of cobalt-60 was added to the gamma irradiation facility making a total of 87,000 curies. Eighteen cobalt pieces were placed in the water-filled tank of the facility in a single transfer from a 5-ton cask. The cobalt pieces were encapsulated in 1/2-inch diameter by 17-3/8-inch long stainless steel tubing at the Radiometallurgy Laboratory. A source holder was designed to position the cobalt around either a cluster of four 2-inch diameter irradiation tubes or around a single 2-inch diameter tube. Ceric sulfate dosimetry measurements have not been completed, but dose rates greater than 5×10^6 r/hr for the 4-tube configuration and 7×10^6 r/hr for the single tube are expected.

EGCR Graphite Irradiation. The ninth capsule, H-3-9, in the series of long-term irradiations of EGCR graphite continues to operate satisfactorily in the GETR. Maximum neutron exposure on the samples is estimated to be approximately 1.3×10^{22} nvt, E >0.18 Mev. Calculations of neutron spectra are being made, using the 2DXY computer code, on the effect of changes in reactor loading on the effective cross-sections for iron and nickel flux monitors, and for effective carbon damage. Calculations are being made for the H-3 capsule position, E7, as well as the N-Reactor graphite capsule positions, D7 and F7. Preliminary results indicate that replacement of a fuel element in the E6 position with a capsule experiment reduces the fast-flux intensity (E >0.18 Mev) 5 to 8% and increases the thermal flux 6 to 8% on the graphite samples. The decrease in the fast flux varies with energy as follows:

CHANGE IN FAST FLUX, %

Neutron Energy, Mev	Capsule Positions: D7 (N-Reactor Graphite)	E7 (EGCR Graphite)	F7 (N-Reactor Graphite)
2.2 to 10	-15 to -22	-0.5 to -6	-15 to -22
0.5 to 2.2	- 5 to -10	-4 to -6	- 5 to -10
0.18 to 0.5	- 2 to - 3	-4 to -5	- 4 to - 3

These spectral changes cause a decrease in gross number of carbon-atom displacements of 4% for E7 and 8% for D7 and F7.

Comparisons are also being made of the fluxes as calculated by the 2DXY code and of the PDQ code (used by the GETR physics group) with the results from flux monitors contained in all graphite irradiation capsules. Flux-monitor activations appear to agree more closely with 2DXY than PDQ. Further refinements to both calculations are under way.

New data on the Ni-58 (n,p) Co-58 cross sections are being used to revise the values of the differential cross sections used in the 2DXY calculations. A more detailed differential cross section for the Fe-54 (n,p) Mn-54 reaction has been obtained recently and is being used in the 2DXY calculations. Preliminary results indicate that the fluxes from the iron monitors may be 10% lower than those previously reported.

High Temperature Graphite Irradiations. Sample data are currently being analyzed from the high temperature graphite capsule, GEH-13-10, irradiated in the K7 position of the ETR. The capsule operated for 127 effective days at sample irradiation temperatures of 800 to 1150 C. Under the high gamma heat in the K7 position some of the capsule internal struts warped and allowed the sample holders to settle into the samples. This had two effects: (1) the thermocouples pulled out of the samples, and all but two failed before the termination of the irradiation; and, (2) the samples in the hottest section warped since there was no clearance for thermal expansion at operating temperature.

Neutron exposures are estimated to range from 2 to 5 x 10^{21} nvt, E >0.18 Mev. Preliminary length-change data indicate that the contractions ($\sim$ -0.8%) of transverse samples of CSF, NC8, and TSX graphites at 1000 C and 4 x 10^{21} nvt are approximately the same as at 500 C; however, parallel samples under the same conditions contracted $\sim$ -3.0%, which is more than the extrapolated amount for 400 C data. At 1100 C and 4 x 10^{21} nvt, the transverse data approximate the 400 C data, whereas parallel samples contracted approximately -4%, or more than any reasonable extrapolations of 300-400 C parallel data. For the above irradiation conditions, length changes for JOZ graphite were +0.6 to -0.2%, which are considerably smaller than the CSF, NC8, or TSX graphites.

Fabrication of a capsule, GEH-13-11, for continued irradiation of these samples is nearing completion. The capsule contains 25 samples from GEH-13-10, four samples from the H-3 (EGCR) capsule program, and five new samples including some raw-coke graphites. Five of the samples from GEH-13-10 were broken either during disassembly or in measuring and so could not be reirradiated. The GEH-13-11 capsule is to be installed in the K7 position of the ETR.

In-Reactor Strain to Fracture. In the next month six assemblies containing strain-to-fracture samples of transverse EGCR graphite will be charged for irradiation. All previous samples irradiated in this test have been parallel EGCR samples.

The maximum strains obtained to date in the high-strain-rate assemblies are 0.41% and 0.38% at 550 C temperature. This in-reactor strain is considerably in excess of the 0.15 to 0.20% maximum strain to fracture noted in laboratory tests on unirradiated graphite.

Hot-Test-Hole Irradiations. Length changes of CSF graphite disks and annular rings have been measured after the fourth irradiation at approximately 600 C to a total exposure of 1.5×10^{21} neutrons/cm^2, E >0.18 Mev. Two disks, 3-inch diameter x 1-inch thick, were cut with diameters perpendicular to the extrusion axis of a bar of CSF graphite. A 1-inch diameter hole was drilled through the center of one disk in the with-grain direction. Diametrical contraction (against the grain) during irradiation was essentially the same for both the disk and annulus. Thus, removing graphite from the center of the disk did not alter the contraction across the outside diameter.

Disks having the same dimensions but cut with their diameters in a plane parallel to two sides of the same graphite bar were also irradiated. The diameters with the grain, W, and across the grain, A, were identified for measurement. Axial holes were drilled in both disks, and a slip-fitting plug was inserted in one.

The diameter of the plug was rotated to coincide with the W diameter of the ring. Since the A thermal expansions are twice the W thermal expansions for the temperature of these irradiations, the plug exerted an outward force along the W diameter of the ring but no force on the A diameter of the ring. Differential radiation-induced contraction tends to increase the thermally-induced stresses created by inserting the plug.

The influence of the plug has become apparent during the last exposure period. The inconsistency of earlier data was attributed to the transition from growth to contraction in the A direction while the rate of W contraction slowly increased.

Contraction of the W and A grain diameters of the ring without a plug was 0.353 ± 0.001 and 0.230 ± 0.013%, respectively, while the same diameters of the ring with the plug contracted 0.330 ± 0.000 and 0.275 ± 0.000. Restraint of the W contraction and enhancement of the A contraction of the ring with the plug is apparent. The rates of contraction for exposures between 1.0 and 1.5 x 10^{21} neutrons/cm^2, E >0.18 Mev, are the same in A directions of the plugged and unplugged rings. However, the W contraction rate of the plugged ring is higher than in the unplugged ring. The apparent inconsistency between total contraction and rate of contraction may be related to a difference between thermal and radiation-induced stresses. However, further irradiation is required to make a quantitative evaluation of contraction rates.

Thermal Expansion of the Graphite Crystal Lattice. The lattice thermal expansions of a powder sample of CSF graphite were examined. The c axis expansion was less than values reported for single crystals, perhaps because the c axis expansion for CSF graphites is inhibited by the misorientation between crystallites. The a spacings were also different than those in the literature, reaching a maximum contraction of 0.053% near 550 C and finally reverting back to their original room temperature values at 1050 C. The a spacings probably, however, contain some error because of the small observed changes.

The bulk thermal expansions for CSF, CSGBF, and a series of graphites with different graphitization temperatures were calculated from the equation,

$$\frac{\Delta L}{L} = A \frac{\Delta c}{c_o} + (1 - A) \frac{\Delta a}{a_o}$$

where $\frac{\Delta L}{L}$ is the length change and $\frac{\Delta c}{c_o}, \frac{\Delta a}{a_o}$ are the lattice thermal expansions. The bulk length changes were more accurately predicted when the crystallite thermal expansions measured on the polycrystalline materials were used in place of the single crystal values previously used.

The A factor in the above equation was found to remain constant with temperature, but the bulk-to-crystallite volume thermal expansion ratio increases with temperature indicating that voids are filling up.

Temperature-Controlled Shielded Capsule. The temperature-controlled, thermal-neutron shielded capsule GEH-23-8 was discharged the latter part of November. Measurements on the samples are presently being made. However, preliminary data on the boronated-graphite samples show:

(1) Exposures attained over the length of the test capsule were 0.83 to 1.37 x 10^{20} nvt (thermal), and 6.9 to 11.8 x 10^{20} nvt (>0.18 Mev).

(2) Nominal sample temperatures over the length of the capsule varied along a modified cosine curve from 700 to 1000 F (370 to 540 C).

(3) The 20 samples that have been examined (there were 26 total) all show contraction. The range of percent contraction was:

7 wt% grey parallel, 0.73 to 0.80; transverse, 0.72 to 0.79;
7 wt% black parallel, 0.66 to 0.82; transverse, 0.14 to 0.20;
5 wt% grey parallel, 0.94 to 0.97; transverse, 0.70 to 0.74;
5 wt% black parallel, 0.56 to 0.86; transverse, 0.08 to 0.09.

(4) Weight changes of the samples showed slight losses and gains, ranging from -0.12 to +0.047%.

Two major differences exist in the results from these shielded samples and those of the unshielded samples (GEH-23-1-7). First, the shielded samples all contracted while the unshielded samples of 23-1 through -7 expanded, except for the 5 wt% parallel samples. Second, all samples having exposure greater than 10^{20} nvt (>0.18 Mev) gained considerable weight in the unshielded tests while no significant weight change occurred to the shielded samples.

Both these differences indicate the great sensitivity of radiation effects on boronated samples to the neutron spectra in which the irradiations are performed. Data at several significantly different neutron spectra will be required before the relative effects from fast and thermal exposure can be confidently predicted for the various types of boronated graphite.

GEH-23-13 was charged in ETR, M-6 position in November and is scheduled for discharge in January 1965. Installation problems resulted in loss of identification of thermocouple positions. The maximum reference temperature for 23-13 is 900 F (482 C). Thirty-five samples of the same material irradiated in 23-8 make up the 23-13 capsule. The average exposure of these samples will be about half of that for 23-8.

7. Irradiation of Thorium-Uranium Fuel Elements

Irradiation of three tubular Zircaloy-2 clad thorium-2.5 wt% uranium-1.0 wt% zirconium fuel elements continued successfully in the ETR-P7 loop. The fuel elements are currently operating at a maximum temperature of 500 C. The maximum surface heat flux is 52 cal/sec-cm^2 (7.0×10^5 Btu/hr-ft^2), and the specific power is 47 watts/gm (143 kw/ft). The integrated exposure is currently 3.6×10^{20} fissions/cm^3 (10,500 Mwd/ton). No swelling beyond that expected from the solid state fission products has been observed. The measured fuel volume increase is currently 1.5%.

8. Advanced Reactor Concept Studies

Nitride Fuel Cycle Study. The nitrides of uranium and plutonium appear to offer potential advantages as reactor fuels. A fuel cycle study has been initiated to investigate the technical and economic potential of nitride fuels in fast power reactors. Initial steps have begun in the selection of a reactor concept model for use in the study. The study will be based generally on a 1000 Mwe power plant incorporating a sodium-cooled fast reactor, with plant operating conditions basically similar to those previously used in AEC-directed 1000 Mwe fast reactor studies. The objective of this study is to generate enough information on nitride fuel characteristics and nitride fuel cycles that comparison with previous studies(1) will permit a valid evaluation of the relative technical and economic merits of nitride, oxide, and carbon fuels in fast power reactor fuel cycles. A thorough review of published and unpublished data on the properties of UN and PuN is currently under way.

Reactor Design Methods. A cooperative effort with Systems Research Operation has been undertaken to employ analog computer techniques for reactor design optimization studies. Initially, an example problem of moderately wide scope will be set up with functional relationships between parameters expressed empirically to a large extent. The problem tentatively selected is that of calculating optimized fuel costs for a fast power reactor.

(1) 1000 Mwe fast power reactor studies performed by GE-APED, Westinghouse, Allis-Chalmers,

9. <u>Critical Flow of Steam-Water Mixtures</u>

Previously, the general results of the experiments investigating the two-phase critical flow behavior of 1/2-inch elbow and tees were presented. In those experiments it was found that the "pseudo-critical pressure" (a pressure defined upstream of the fitting) of a 1/2-inch elbow and a 1/2-inch tee acting as an elbow bore an apparent fixed relationship for all corresponding enthalpies and flow rates. Several of these experiments were repeated during the current report period with more accurate and sensitive measurement instrumentation in an effort to reduce random variation in the results and to investigate the uniqueness of this relationship. These more exact measurements showed a slight dependency of the ratio of tee-to-elbow pseudo-critical pressure on enthalpy not observed in the earlier experiments. This ratio varied from 1.25 at an enthalpy of 500 Btu/lb to 1.17 at 1000 Btu/lb with a mean value of 1.21.

Effort is being applied to relate the observed phenomena to established theories and correlations. Preliminary comparisons of the measured "true" critical discharge pressure from wall taps in the test section to predictions of existing theories reveal excellent correspondence for the 1/2-inch elbow and less than 20% error for the case of the 1/2-inch tee. However, it is not possible to verify that the pressures measured by the wall taps in the vicinity of the fitting exit are indicative of true static pressure since there is no assurance that the direction of the turbulent two-phase flow is normal to the bore of the taps. In view of these difficulties, an alternate method of correlation of the results is being attempted. This method involves the use of a conventional separated flow critical flow model to predict the critical pressure, i.e., the pressure at the exit of the test section. A two-phase pressure drop correlation is being developed for the fittings to relate the pressure drop from the point of definition of the pseudo-critical pressure to the calculated exit critical pressure using the separated flow model. Preliminary attempts to relate the two-phase pressure drop of the fittings to the single-phase pressure drop of a straight length of pipe have proved to be moderately consistent. This method of pressure drop correlation is similar to that used successfully in HW-80970 REV 1 for the two-phase pressure drop through elbows and tees acting as elbows.

10. <u>Phoenix Fuel Program</u>

A revised proposal for the Phoenix-fueled MTR experiment was prepared in conjunction with other Laboratories components, based on the previously completed analysis and the review of the proposal with AEC Headquarters personnel in November. Design analysis cost was increased somewhat to allow for some of the uncertainties as to the extent of core modification, control system analysis, and possible design changes to improve compatibility with other testing if performed simultaneously.

11. Plutonium and U-233 Fueling of a Fast Compact Reactor

Core temperatures and coolant pressure ratio were calculated for the most recent 710 test reactor design to provide a comparison of methods with NMPO. The test case temperatures and NMPO temperatures correspond with 10 F; however, total pressure ratios are 10% less than those reported by NMPO because of a more conservative method of calculating pressure loss in the core. A 41-case parametric thermal-hydraulic study for U-235 and U-233 fueled reactors (sizes calculated by Engineering Physics Operation) was completed, but results have not been analyzed. In this study allowance was made for the spacing between hex elements (4%), outer clad thickness, and actual density for cermets (95%), based on an arbitrary fuel element outer dimension. For convenience, these allowances were all deducted from the void fraction used in the nuclear calculations, even though some of the material is tungsten rather than void, since the heat transfer code does not provide for a nonfueled solid region. These assumptions led to the following relation between gross void fraction used in the nuclear calculations and void fraction available for coolant and tube clad.

Gross Void Fraction	Coolant & Tube Clad Void Fraction
0.20	0.0662
0.30	0.1870
0.35	0.2465
0.40	0.3070
0.50	0.4270

Parameters varied were: gross void fraction (0.2-0.5); axial power profile (2 cases for U-235 fuel); and vol% tungsten in the cermet. Coolant bulk temperatures and inlet pressure, reactor power, and materials temperature limits were taken as equal to those of the 710 test reactor.

12. Nuclear Rocket Fuels Studies

Research and development in the field of nuclear fuels and alloy development in support of NASA programs continued. Details of these activities are reported separately via distribution directly to the sponsors.

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D. DIVISION OF RESEARCH - 05 PROGRAM

1. Radiation Effects on Metals

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

Measurements of lattice parameters of irradiated single crystal molybdenum rods have been continued. These rod samples, irradiated to 7×10^{18} nvt (E >1 Mev), were intended for simultaneous measurements of length change and lattice parameter. Length measurements, previously reported, show an increase of 0.023% to 0.024% upon irradiation. The measurement of lattice parameters has been complicated by what appears to be local differences in lattice spacings and by the refraction effects arising from the curved surfaces of the crystals. Measurements at 15 to 20 different points on each of two crystals yield an average value for each sample of a_o = 3.14765A ± 0.00006A. This is an increase of 0.016 to 0.020% over the unirradiated value. The length change is then substantially greater than the lattice parameter change. These results are compatible with a structure containing excess, isolated vacancies and excess, clustered interstitials.

Samples of molybdenum foils containing <10 ppm carbon and 400-500 ppm carbon have been annealed two hours at 1000 C after irradiation to 1×10^{20} nvt (E >1 Mev). X-ray lattice parameter and line breadth measurements were made for comparison with those of as-irradiated foils and foils annealed at 750 C for two hours. In the as-irradiated state the low carbon (<10 ppm) samples have a lattice parameter of 3.1478A which decreases to 3.1476A after a 750 C anneal and to 3.1474A after a 1000 C anneal. The high carbon (400-500 ppm) lattice parameter decreases from the as-irradiated value of 3.1476A to 3.1474A after the 750 C anneal. The 1000 C anneal produces no further change in lattice parameters. Line breadths for both samples decrease continuously with annealing temperature and attain values only a little greater than the pre-irradiation values after the 1000 C anneal. Additional transmission electron microscope results of molybdenum foils containing <10 ppm and 450 ppm carbon which were irradiated to 10^{20} nvt and annealed at 750 C for two hours have been obtained. Prior to annealing, both samples appear to be very similar, i.e., defect clusters and dislocation loops are present. After the annealing treatment, the samples are similar in that large dislocation loops are present and are interstitial in character.

However, the molbydenum containing carbon shows irregular dislocation networks which are not found in the low carbon samples. The irregular networks form as a result of loop growth and subsequent interaction during annealing. The presence of the networks is evidence that the damage sustained by the two types of samples is different though this difference cannot be detected by electron microscopy. This observation is in agreement with that made on similar samples irradiated to lower exposures (10^{19} nvt) which disclosed the presence of dislocation loops only in the molybdenum containing carbon.

Foils of molybdenum, similar to that described above, have been deformed in the "as-irradiated" condition. Fracture occurred after very little deformation. Examination of the foils failed to show any channels or dislocation networks. Apparently fracture occurred along grain boundaries and was highly localized.

Recent observations on replicas from severely bent irradiated molybdenum single crystals containing carbon revealed the presence of short slip line segments emanating from the region around carbides. It appears that the carbide particles are serving as dislocation sources as well as dislocation barriers as mentioned previously.

The equations for the "two beam dynamical theory" of electron diffraction through a foil with a general dislocation have been programmed. A document HWSA-3779, has been written which describes the program.

Tests on the temperature- and rate-dependence of deformation in high purity polycrystalline molybdenum are continuing. Attempts to experimentally determine the magnitude of τ^*, the "effective stress" on a dislocation, are in progress. In this series of tests, elevated temperature tensile tests are being conducted to determine the temperature above which there is no increase in flow stress with an instantaneous 10-fold increase in strain rate. Above this temperature, then, all the resistance to dislocation motion is derived from long-range internal stresses, and the short-range stresses are zero. By assuming that the internal stresses remain constant over an extended temperature range, then the difference in flow stresses between the critical temperature and some lower temperature is due to short range stresses. Tests have been conducted in vacuo at pressures of about 1×10^{-6} Torr, and the critical temperature has been determined to lie between 900 and 1000 K. Further tests will establish the temperature precisely.

2. Plutonium Physical Metallurgy

The objective of this program is the derivation of fundamental information relative to (1) the kinetics and mechanics of the phase transformations in plutonium, and (2) the mechanisms by which monoclinic plutonium deforms.

Experiments were performed to determine whether plastic deformation of the beta phase by tension would influence the $\alpha \to \beta$ transformation rate in a manner similar to plastic deformation of the beta phase by compression. It has been well established that deformation by compression decreases the $\beta \to \alpha$ transformation rate. The present experiments showed that the $\beta \to \alpha$ incubation time is increased and the transformation rate is decreased. The influence of deformation by tension is not as marked as deformation by compression.

The $\alpha \to \beta$ and $\beta \to \alpha$ transformations were shown to be very anisotropic. A highly textured rod specimen with the 010 plane parallel to the cross sectional area was cycled once through the $\alpha \to \beta$ and $\beta \to \alpha$ transformations. The length increased 7.9 and the diameter increased 0.8% during the $\beta \to \alpha$ transformation. This indicates that nearly all of the volume change during transformation occurred in a direction parallel to the length of the specimen. The length and the diameter decreased 1.2 and 4.3%, respectively, during the transformation. The sample after transformation cycling had increased in length by 6.5%. The diameter after transformation cycling had decreased by 3.4%. It is quite evident that both the $\alpha \to \beta$ and $\beta \to \alpha$ transformations result in anisotropic volume changes.

The plasticity of the beta phase formed from unalloyed alpha phase plutonium is much greater than the plasticity of the beta phase formed from gamma. The beta phase in beta stabilized alloys has a very low plasticity. It is formed from gamma during casting. The beta phase in beta stabilized alloys should be more ductile if the beta could be prepared by transformation from alpha rather than from gamma. Accordingly, the beta phase in a Pu - 3 at% Zr alloy was transformed first to alpha by isostatic pressing at 90,000 psi and then to beta by heat treating at high temperatures in the beta phase. The plasticity of the beta phase formed in this manner was indeed higher. For example, the creep rate at 265 C and 4000 psi was 10 to 12 times greater for beta formed from alpha than beta formed from gamma. This increase was not nearly as great as in unalloyed plutonium and may be due to the fact that only 70% of the beta transformed to alpha during pressing. It is to be noted that there was a significant difference in the intensities of the x-ray reflections obtained from the beta material prepared from (1) alpha and (2) gamma. The procedure used for making a stabilized beta alloy that can be more easily deformed shows considerable promise.

An attempt is being made to establish the effects of the various phase transformations on textures existing in alpha plutonium. This is done using the elevated temperature specimen holder on the x-ray diffractometer and following the intensity of various lines during the course of the alpha to beta, beta to gamma, gamma to beta, and beta to alpha transformation. It appears that a previously established texture is changed but not randomized by going from alpha to gamma and back. Interestingly enough, there is evidence that in thin randomly oriented specimens a texture is established by the various transformations.

Evidence has been found for a definite difference in the crystallographic texture of beta plutonium formed from gamma as compared with that from alpha. This may lead to an explanation for the marked difference in strength characteristics in the beta formed from gamma as compared with that formed from alpha.

Evidence has also been obtained for a marked difference in the mechanics of the beta to alpha transformation in the case of beta formed from gamma as compared with that formed from alpha. In the former case it appears that the transformation reverses itself several times during the process. There is evidence to indicate that the maximum transformation temperature during slow cooling is about 80 C for beta formed from alpha, and 95 C for beta formed from gamma.

There is a great deal of work to be done in this area, and experiments must necessarily be repeated many times to ensure that the observed effects are real and reproducible.

E. CUSTOMER WORK

1. Radiometallurgy Laboratory

Examinations

During the period November 20 to December 24, 1964, examination and testing included the following:

Photomicrography	30
Photomosaics	6
Autoradiography	28
Replication	15
Fission Gas	22
Density	9
Micro-drilling	74

Dissolution	30
Macro-hardness	124
Micro-hardness	16
Tensile Testing (Rm Temp.)	14
Tensile Testing (Hi Temp.)	23
X-ray	7

Addition and Equipment

Project CAH-136, Service Addition - 327 Bldg. Detailed drawings were revised by Vitro Engineering, and a second review was made by Radiometallurgy. Another list of comments was submitted for revision of the drawings.

Project CGH-857, Physical & Mechanical Properties Testing Cell. The remote impact tester was received from Testing Machines, Inc. Installation of the tester will be delayed until "G" cell work load decreases so the cell can be taken out of service without affecting the testing program.

"E" Cell Metallographic Facilities. Installation and alignment of the remote metallograph in the "E" cell blister was completed. A vacuum sample storage unit was fabricated and installed in "E" cell. Redesign of the fine grinding equipment was started after tests showed that the proposed method was too slow.

"B" Cell Modification. The "B" cell door was received from 100-H shop after completing installation of the large viewing window. Further modification of the north end of the cell was delayed by temporary installation of equipment for special work. A preliminary arrangement drawing was made for the equipment in the south end of the cell.

Remote Belt Sander. Final testing of the remote belt sander was completed. The unit will be installed in "D" cell with the fine grinding equipment.

Improved Cell Lighting. A study was started to improve in-cell lighting and reduce the viewing problem associated with radiation-induced darkening of the lead-glass viewing windows. Preliminary studies show that a 175-watt mercury-vapor lamp and a 500-watt quartz lamp both give the same amount of illumination through a 15-inch viewing window. This is twice the amount produced by a 300-watt incandescent lamp.

Zeolite Ion-Exchange Capsule Test Equipment. Fabrication of the vacuum leak-testing chamber and capsule-testing furnace was completed. A high-frequency start was ordered for the arc welder to facilitate in-cell fusion welding of the capsule.

2. Metallography Laboratories

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

It has been demonstrated that the recently installed metal vacuum cathodic etching apparatus is operable by selecting either of two high voltage polarity modes. One mode is represented by the anode at ground potential and the cathode at a maximum negative 3800 volts. The second mode places the cathode at ground potential and the anode at a maximum positive 3000 volts. Electrical conduction by the krypton etching gas results in arcing to the metal vacuum system at ground potential if the listed maximum voltage levels are exceeded. Overload switches protect the power supply, and, therefore, operation at maximum voltages is practical. Safe operation in the minus 3800-volt cathode mode is accomplished by utilizing the electrical resistance of water in 12 feet of plastic affluent and effluent tubing to the water-cooled, and otherwise electrically-insulated, cathode.

Experience has been gained with both modes and the negative cathode mode is preferred. In addition to the 800-volt greater potential, there are several other advantages to the negative cathode mode. Material removed by etching deposits on the removable glass chimney to a greater extent. The advantage of this is that difficult-to-clean portions of the apparatus require less maintenance than with the positive anode mode. A second advantage is the stability of the etching glow that occurs during operation. Etching conditions can fluctuate to greater extremes without a resulting loss of the glow which is frequently the case with the positive-anode mode.

During the report month, 511 samples were processed, a total of 686 micrographs and macrographs taken, 2011 negatives printed, and 6940 prints processed.

3. High Temperature Lattice Test Reactor (HTLTR)

Graphite core temperatures have been increased during the month with a maximum recorded core temperature of approximately 1000 C. The core was held at about 900 C for a period of one week during which time corrosion of graphite samples inserted into one of the core holes was studied. It was found that unexposed samples exhibited excessive weight losses during a short period following initial insertion of sample into the core. However, samples which had been previously exposed to the core or samples left in the core for an appreciable time period exhibited insignificant amounts of corrosion.

Gas analysis performed on the mass spectrometer have indicated respective carbon dioxide and hydrogen concentrations of up to 0.3 and 0.39 vol%. Water concentrations slightly greater than 1% were indicated by the in-line water monitor.

Small fluctuations in the heater element circuit current and voltage drop have been observed. It will not be possible to assess the cause until the experiment is terminated and the circuit components can be visually observed.

Fabrication of the vertical control rod element is 99% complete. The new shell has been installed over the test pit, and the operating mechanism has been checked in it.

Environmental Tests on HTLTR Materials. Approximately 5% B_4C dispersed in a graphite matrix is intended for use in the control mechanism for the HTLTR. For the most part this material will be enclosed within TD nickel structures, but there may also be occasion for the B_4C-graphite to contact the insulating brick at high temperature. A test was run to evaluate the compatibility of the B_4C-graphite compact with TD nickel and insulating brick under reactor operating conditions.

Capsules containing B_4C-graphite either clad in 0.010-inch TD nickel or in contact with K-23 insulating brick were exposed in a nitrogen-graphite environment at 1100 C for 800 hours. No gross reaction occurred either with the TD nickel or K-23 brick. Metallographic examination of the specimens is in progress.

Metallographic results obtained on TD nickel cladding from UO_2-graphite specimens exposed in a similar test described last month indicated that no significant reaction had occurred between the UO_2 and nickel. An environmental test was started in which pure UO_2 in TD nickel containers is being exposed in a simulated reactor environment. Also included in this test are Al_2O_3 and Al_2O_3 - Gd_2O_3 powder specimens in TD nickel containers.

4. EBWR Fuel Elements

Fuel Rod Repair. About 500 fuel rods were returned from Argonne after recently developed ultrasonic techniques revealed slight internal cracks at the end of the rod. Repairs will be made by cutting the tube just below the weld area, removing the end cap, decontaminating and welding a new end cap in place. By the end of January the rods are expected to be available for use at Argonne.

Studies are being made to determine the causes of the erratic cracking during vibrational compaction.

Thermal Cycling Test. In order to obtain preliminary information on the propagation of cracks found in the end cap region of EBWR fuel rods, thermal cycling tests were performed using the horizontal autoclave in the 308 Building. Eight EBWR rods were selected from a group, originally intended for shipment to ANL, which had been rejected during ultrasonic inspection. Defects found by the ultrasonic inspection were located in the weld region adjacent to the bottom end cap, and measured both as to magnitude and extent prior to thermal cycling. The rods were cycled by heating to 400 C and 1000 psig in one hour, holding at 400 C and 1000 psig for three hours, and cooling to 100 C and 0 psig over an 8-hour period. Ultrasonic examination was performed after 10, 36, and 52 cycles. The results are shown below:

Rod No.	Original Indication		Indication After Cycling* 10 Cycles	36 Cycles	52 Cycles
EV 32	5°	1.2 mil	1.2 mil	4.4 mil	5.2 mil
EU 96	10°	2.0 mil	1.2 mil	1.2 mil	2.8 mil
ET 04*	90°	5.0 mil	5.0 mil	13.6 mil	26.0 mil
EU31**	180°	7.2 mil	7.0 mil	12.8 mil(200°)	17.2 mil(200°)
EV07**	10°	1.4 mil	1.4 mil	1.4 mil	2.2 mil
ET 08	45°	2.6 mil	2.6 mil	4.4 mil	8.8 mil
EV 37	10°	2.0 mil	2.4 mil	10.4 mil(90°)	28.0 mil(90°)
EV 38	5°	1.6 mil	1.2 mil	1.5 mil	1.5 mil

*Angularity remained the same unless otherwise noted.
**Specimens double canned after 36th cycle to prevent autoclave contamination.

Propagation of the defects is occurring and is especially notable in fuel rods ET 04, EU 31, and EV 37. The test has been terminated, and metallographic examination of the defect areas will be performed.

Irradiation Testing of EBWR Prototype Fuel Rods. Capsule tests continue to indicate that EBWR rods will exhibit satisfactory performance under proposed EBWR conditions. The maximum exposure of the 10 capsules under irradiation in the MTR is 4.3×10^{20} fissions/cm^3 (16,000 Mwd/ton of fuel). Twenty-two capsules with a maximum exposure of 3.1×10^{20} fissions/cm^3 (11,600 Mwd/ton of fuel) have been discharged.

5. Saxton Fuel Elements

The fabrication of 150 Vipac UO_2-PuO_2 fuel rods for the Saxton Reactor is proceeding on schedule. These Zircaloy clad rods are three feet long and 3/8-inch diameter. Seventy-five kilograms of the UO_2-6.6 PuO_2 core material has been processed through pneumatic impaction, and the remaining one-half will be processed after the various quality control results have been checked. Zircaloy tubing has been received and is presently being tested ultrasonically. The first few tubes will be used in vibrational compaction studies.

6. Other Customer Work

PuO_2-NiCr Compatibility Studies. No reaction was detected between PuO_2 and NiCr after heating a PuO_2-62 wt% NiCr pellet 12 hours in pure hydrogen at 1200 C.

Small holes were drilled through NiCr-clad PuO_2-62 wt% NiCr and UO_2-62 wt% NiCr specimens prior to heating in air at 982 and 1093 C. Visual inspection of the PuO_2 specimens after 400 hours at temperature showed no change other than surface oxidation. The UO_2 specimens had swelled considerably.

Sixteen NiCr-clad PuO_2-62 wt% NiCr cermet test specimens prepared by pneumatic impaction, are being heated in air at four different temperatures for 400 hours. Compatibility between PuO_2 and NiCr will be determined by alpha counting and metallographic techniques.

FW Albaugh

Manager
Reactor and Fuels Laboratory

PHYSICS AND INSTRUMENTS LABORATORY

MONTHLY REPORT

DECEMBER 1964

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

NPR Utilization Studies

The NPR Phase III study of PuO_2 enriched UO_2 has essentially been completed. A few additional calculations are required to complete the examination of temperature induced reactivity changes.

Further 9-ANGIE, two-dimensional analysis of the NPR lattice has been carried out. The results indicate that the scheme of arithmetic averaging of cell reactivities to obtain reactor multiplication can be extended to 4-batch loadings. For a specific case using 4 batches of ThO_2 of initial enrichment of 4.0 w/o $U^{235}O_2$ ranging in exposure from "green" to $\sim$ 60,000 MWd/t, the error introduced in the average k was only 1.3%.

Some difficulty has been encountered with the 9-ANGIE code in getting the fluxes to converge in the moderator region even though the multiplication convergence of 0.1 mk has been met. This problem is presently under investigation.

NPR Control Rod Enhancement Studies

The fabrication of two test rods of B_4C and B^{10} has continued and is nearing completion. Small quantities of SmO_2, CdO_2, EuO_2, and Hf mixtures have been located in IPD and are available for additional experiments.

NPR Co-product Studies

Measurements of k_∞ and spline reactivity worth measurements on the NPR co-product lattice have been completed in the PCTR on schedule. The NPR exponential assembly and process tubes have been cleaned and are ready for the experiments. The start of these experiments awaits the receipt of fuel to be provided by N-Department.

Twenty-one tubes of co-product fuel are to be irradiated at full power in N-Reactor to provide experimental values of the conversion ratios. Design of this experiment has continued with the emphasis shifted from an axial block load to a peripheral block load. This does not jeopardize the physics but does enhance the ease of control of the block in the event that power peaking occurs. Continuing analyses of the block test will employ transport theory values of the thermal cross sections, as found from "THERMOS".

Analysis of NPR Startup Experiments

Reaction rates of various radioactivants (Cu, Au, Lu, Eu, Pu) in the annuli of the NPR fuel were determined in the startup tests on N-Reactor. These experimental data have been corrected for radioactive decay and other minor experimental variations and have been published.[1] Interpretation of these reduced data has begun with encouraging results. Transport calculations of the relative reaction rates of Cu-63 in the two fuel annuli agree quite well with observed values. Precise spatial distributions within each fuel annulus are not predicted well because of an inadequacy of the cell model used in THERMOS. This problem is under study at present.

Co-Product Experiments for NPR

Work is in progress to measure the infinite medium multiplication factor k_∞ and conversion ratios in both driver fuel tube and target rod of the co-product lattice. All experimental work has been completed in the PCTR to determine cadmium ratios, thermal utilizations, and k_∞ of the dry lattice. Another dry lattice without target rod was studied using perturbation and foil irradiation techniques to provide similar information. This completes the experimental phase.

All foil data for the wet cases have been processed by computer program APDAC and the results have been used to determine a first rough value of k_∞. The data for the dry cases are being prepared for computer processing.

Subcritical Experiments with Enriched N-Fuels for Nuclear Safety Guidance

A series of exponential and neutron multiplication experiments were completed with enriched N Fuels of 1.25 wt% and 0.95 wt% U in light water. The experiments provide data for nuclear safety guidance in fuel element handling and have included measurements with two fuel assemblies in each of four lattices. The fuel assemblies were the 1.25 wt% enriched outer tube (2.4 in. o.d., 1.8 in. i.d.), and the tube-in-tube assembly comprised of the outer tube and a 0.95 wt% enriched inner tube (1.25 in. o.d., 0.44 in. i.d.). Results of

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[1] L. C. Davenport, N. A. Hill, and T. B. Thornbury. Lattice Parameters and Spectral Index Measurements on the N-Reactor Lattice at Two Temperatures. Part I: Data, HW-84085. September 14, 1964.

the most recent measurements, which involved tight packed lattices with fuel elements in contact, are presented below:

Tube-in-Tube Assembly

(1.25 wt% outer tube; 0.95 wt% inner tube)

Lattice Spacing	Volume Ratio	Buckling	Computed Critical No. of 52 in. Assemblies from Buckling	Computed Critical No. of 26-in. Assemblies in Cylindrical Array
2.4 in.	0.66	1504 μB	416.6	798.7 (~36,756 lb. U)
		Outer Tube Only (1.25 wt%)		
2.4 in.	1.57	4436 μB	86.7	124.7 (~3,827 lb. U)

Note that for these assemblies, even in close contact in water, there is an appreciable degree of moderation, especially in the case of the 1.25 wt% enriched outer tube, for which the critical mass (~125 fuel elements of 26-in. length) in the close packed array is only about 40% larger than that for optimum moderation. A report detailing the results of these experiments is under preparation.

Instrumentation

A study initiated to improve performance of the gamma energy measurement portion of the N Reactor fuel element rupture detection system has recommended the following:

1. Raise the gamma energy signal channel from the 1.1-1.4 MeV band to 2.4-4.5 MeV band.
2. Temporarily disconnect the standardization alarm circuit to avoid repeated alarms.
3. Provide either a more suitable higher energy standardization (reference) gamma source or incorporate circuitry modifications to momentarily reduce system electronic gain during movement of of the detector past the present source.
4. Incorporate instrumentation on a "ready" status to record the approximate gamma spectrum of the corrosion products as encountered at the turrets.

5. Establish procedures and instrumentation for rapidly identifying corrosion products in case of failure of the present sample chamber backflushing technique.

6. Until sample flow difficulties are alleviated, incorporate the use of 80 second-old sample water.

Purchase orders have been placed for commercial instrumentation to be used in the prototype reactor irradiated fuel cooling age measuring system. Basic system design has been completed.

System Studies

A MIDAS program was written for the N Reactor three surge-tank simulation. The program used a simpler set of equations than the analog simulation developed earlier. The new program included only half of the analog simulation; surge tank controllers were eliminated.

An attempt was made to improve stability of the N Reactor pressure level control system by adding a derivative circuit to the Bailey Meter unit. Performance of the modified system has not as yet been evaluated.

Modifications are being made to the N Reactor controller simulator to expand the amount of controller equipment which it can simulate and to facilitate set up. Preparations were also started for simulation study of the power generation system of the N Reactor and WPPSS plants. Program objectives have been scoped with the customer and preliminary simulation data gathered.

SEPARATIONS

Experiments with PuO_2-Polystyrene Critical Assembly

A series of critical experiments were performed with the Remote Split-Table Machine and a PuO_2-plastic assembly comprised of a rectangular parallelopiped reflected with various composite reflectors on one end. The effect on the reflector savings of Lucite of positioning neutron absorbing materials at the reflector core interface was determined. Comparisons were made between the effect of boron, and gadolinium in thin stainless steel plates, and cadmium sheet. In addition, data were also obtained for the relative reflector savings of lead as opposed to Lucite and for reflectors of polyethylene containing 5 wt% boron and 10 wt% boron. These reflectors were 8 in. thick and effectively infinite from the viewpoint of criticality; i.e., a further increase in reflector thickness would not have reduced the critical mass or length. The critical assembly core was comprised of 2-in. fuel bearing cubes of PuO_2 in polystyrene spaced alternately with 2-in. cubes of Plexiglas. The average Pu concentration in the assembly was ~0.54 g/cc, with the H/Pu atomic ratio being ~35.

The experiments with neutron absorbing materials (stainless steel plates containing gadolinium and boron, etc.) positioned at the interface of the core and reflector were performed to provide data for assessing the relative effectiveness of neutron absorbers in the construction of stainless steel dissolver inserts for processing enriched U at Purex. It is interesting to note that for the above system Lucite was found to be a better reflector than lead, the difference in reflector savings being ~0.6 cm. The effectiveness of lead as a reflector is of concern in the design of shielded shipping containers for irradiated fuels. The material having the least reflector savings of any of the reflectors tested was the boron impregnated polyethylene (10 wt% boron). The various materials used in the reflectors and the measured reflector savings are given in the table that follows:

Reflector Savings of Composite Reflectors

Critical Length of 12-in. Square Column (cm)	Reflector Composition on One End of Column	Reflector Savings (cm)
34.254	None	
30.528	8 in. Plexiglas	3.73
30.269	8 in. Plexiglas (re-check)	3.99
31.220	8 in. Plexiglas with 0.25 in. Stainless Steel Plate	3.03
32.042	8 in. Plexiglas with 0.19-in. S.S. plate containing 0.3 wt% gadolinium	2.21
32.125	8 in. Plexiglas with 0.15-in. S.S. plate containing 0.7 wt% gadolinium	2.13
32.538	8 in. Plexiglas with 0.27-in. S.S. plate containing 1.95 wt% boron	1.72
32.368	8 in. Plexiglas with 0.18-in. S.S. plate containing 0.2 wt% boron-10	1.89

Critical Length of 12-in. Square Column (cm)	Reflector Composition on One End of Column	Reflector Savings (cm)
32.735	8 in. Plexiglas with 0.03-in. cadmium sheet	1.52
32.786	8 in. polyethylene impregnated with 5 wt% boron	1.47
32.905	8 in. polyethylene impreganted with 10 wt% boron	1.35
30.972	8 in. lead	3.28

Neutron Interaction Experiment with Cans of PuO_2 Powder in Low Density Cubic Array

Subcritical neutron multiplication experiments were performed with PuO_2-powder that provide data for nuclear safety guidance in storage and handling of PuO_2. The final subcritical assembly consisted of a fully reflected cubic array of 100 cans containing 208 kg of PuO_2. Each of the cans ($4\frac{1}{2}$ in. i.d., $5\frac{1}{2}$ in. high), which were partially filled, contained 2.08 kg of PuO_2. The dimensions of the cubic array formed by stacking the cans in close contact were 22 in. x 22 in. x 22.5 in. The average density of PuO_2 in the array was about 1.2 g/cc with the H/Pu atomic ratio being unity. The Pu contained 8.2% Pu^{240}. The concrete reflected cubic array was too far subcritical to obtain an accurate estimate for the critical number of cans, but the inverse neutron multiplication curves gave a value of about 150 cans ($\sim$ 300 kg PuO_2) for criticality, with an estimated uncertainty of ± 25 cans.

Pulsed neutron source measurements were also made on the assembly during its construction. From the observed variation in prompt neutron decay constant during the approach, the prompt critical number of cans was estimated to be 143, which is in qualitative agreement with the results obtained from the inverse neutron multiplication curves. Differences between delayed and prompt criticality are, in this case, within the uncertainty of the two measurements. One of the difficulties in performing the experiments was caused by excessive generation of heat from α decay ($\sim$400 watts). Temperatures within the central portion of the array rose to 95°C during the course of the experiments, and it was necessary to apply some air cooling to keep the temperature from exceeding this value.

Critical Mass Laboratory Instrumentation

Due to the fast systems encountered in the plutonium-plastic compact assemblies at the Critical Mass Laboratory, it is desirable to be able to measure neutron lifetimes in the 10^{-8} to 10^{-6} second range. As a solution to this problem, work was initiated on a time-to-pulse-height converter to be used in conjunction with the TMC 256 multichannel analyzer and pulse-height logic unit. The circuit designed will convert the time between two consecutive neutron detections into a pulse-height to be analyzed with the multichannel analyzer. The result is a prompt decay curve from which the neutron lifetime may be determined.

The existing output smoothing filters on the noise analysis system at the Critical Mass Laboratory, which are third order active low-pass networks, followed the signal too closely and caused excessive fluctuation in the analyzer output. To correct this condition, White Instrument Company sent four R-C low-pass output smoothing filters for evaluation. The low-pass filters received were for 3, 10, 30, and 100 cps. The 10 cps filter was chosen, as it followed the random signal with a minimum of fluctuation while being fast enough to provide an accurate display.

Consulting Services on Nuclear Safety-Criticality Hazards

1. Nuclear Safety in HL

   The following nuclear safety specifications were reviewed:

   a) Specification B-12, for Experiment Reactors Operation, on handling of 18-23 wt% Pu-Al alloy discs.

   b) Specification W-1, for Materials and Process Chemistry, on 25 wt% and 77 wt% U^{235} enriched uranyl nitrate solution.

   c) Temporary Specification No. 6, for Ceramic Research and Development, on 93 wt% enriched uranyl nitrate.

   d) Specification C-18, for Critical Mass Physics, on storage containers of PuO_2.

2. Nuclear Safety in CPD

   Assistance was provided on the completion of a hazards review for a new plutonium processing hood in the 234-5 Building. The results of this review are given in a letter to R. J. Sloat from D. E. Braden, C. L. Brown, and B. F. Judson, dated November 30, 1964.

3. Nuclear Safety in NRD

a) Limits for 2.5 wt% Enriched Uranium

A preliminary review of the nuclear safety of fabricating 2.5 wt% enriched N-Reactor fuel elements in 333 Building was completed for N-Reactor Fuels Engineering (reference: letter to W. G. Hudson from C. L. Brown and D. R. Oden, dated November 13, 1964). Safe processing limits were given for 2.5 wt% enriched uranium fuel elements, scrap metal, and solutions.

b) Revised Nuclear Safety Limits for N-Reactor Spike Fuel

As a result of the exponential measurements made on N-Reactor spike fuel (i.e., 0.95 wt% inner tubes and 1.25 wt% outer tubes), the nuclear safety limits for fabricating and processing this fuel in NRD are being revised. The new limits are given in the table below, together with the previous limits as comparison. For the 1.25 wt% outer tubes, the safe mass is increased 60%, and the safe slab thickness, 15%. For the tube-in-tube fuel assembly, the safe mass is about double the previous value, and the safe slab thickness is increased 36%.

Revised Nuclear Safety Limits for Enriched N-Reactor Fuel

	Safe Mass lb. U		Safe Volume, Gallons		Safe Slab Thickness, in.	
	Previous Value	New Value	Previous Value	New Value	Previous Value	New Value
Outer Tubes Only (1.25 wt%)	717	1157	31.4	47.0	8.9	10.2
Tube-in-Tube Assemblies (0.95 wt% Inner Tube, 1.25 wt% Outer Tube)	1067	2093	33.9	67.3	8.9	12.1

c) Nuclear Safety of Shipping Cask

A nuclear safety review of the Stanray Corporation cask for shipping irradiated N-Reactor fuel elements was made. (Reference: Letter to R. V. Poe from C. L. Brown dated December 21, 1964.) It was estimated that the cask would be safe for 1,000 linear feet of 0.95 wt% or 1.25 wt% enriched N-Reactor fuel elements or 616 linear feet of 1.95 wt% enriched fuel. To insure nuclear safety in the cases of the 1.25 wt% and 1.95 wt% fuels, boron steel plates are required for neutron poisoning.

4. Nuclear Safety in Training and Education

A course of lectures in nuclear safety for personnel in N-Fuels, NRD, was completed. Course emphasis was on the nuclear safety of fuel element fabrication operations in the 333 Building. Nine lectures in all were given, with 20 persons in attendance.

5. Test of HL-Designed Class I Shipping Container

The Class I HL-designed shipping container for fissile material that successfully passed drop and fire tests during October and November has now been tested for water in-leakage. The container was submerged in water to a depth of four feet for a period of four days. After the test, the container was found to be dry. The cotton, which was placed inside before the tests were begun, was found not only to be dry, but unscorched, indicating that there was no significant temperature increase inside the container during the fire test. The container has now passed all required tests.

Instrumentation and System Studies

Detector and solid state circuitry modifications were incorporated into the Pu-239 liquid sample counting system to improve stability and to provide a better signal to noise ratio. Laboratory tests of the modified system indicated 1 g/l sensitivity.

Development of an instrument to measure distribution of Am-241 in a process column has continued with partial completion of the electronic circuitry, and the design and partial fabrication of the detector probe holder.

A third model of the pot calciner was programmed and run on the EASE 1132 analog computer for Waste Solidification and Engineering Development Operation of Hanford Laboratories. A MIDAS check solution was written for the new model.

METALLURGY - Nondestructive Testing

N Fuels Testing

Work has been completed on a study to determine the feasibility of detecting clad thinning over the brazed (dogbone) regions of N fuels. A test employing boundary wave phenomena was found to be sensitive to clad thickness difference on the two fuel samples which were available for this experimental work.

Ultrasonic testing of the jacket-to-core bond on a pilot run of lithium-aluminum target fuels has been completed. Destructive correlation shows that the test is incapable of differentiating between the mechanical bonds that sometimes occur with this process and actual diffusion bonds which are not desired in the finished product. This limitation restricts the utility of this tester, but when used to statistically sample the product, the number of samples requiring destructive examination will be greatly reduced.

The X-ray fluorescence tester, which was designed to detect uranium contamination in N fuel end cap braze closure, has been modified to optimize the performance of this equipment. The optimization program included elimination of the external braze test and a reduction in the testing rate. Neither of these changes are compatible with production requirements. Preliminary data analysis indicates a factor of 3 to 4 improvement in tester reproducibility, and a slight improvement in the correlation between measured and actual contamination levels as a result of increasing the signal-to-noise ratio from 3 to 7.

Improved techniques of end closure autoradiography have produced test results which substantially exceed those possible with the present X-ray fluorescence equipment and current indications are that efforts on this project will be discontinued.

Fabrication of a fuel handling system for use with the surface contamination tester (AC-3) is about 95% complete. Fabrication of the outer surface detecting heads will be initiated upon receipt of a supplemental work order, with the final checkout and delivery of the system to follow immediately.

An enrichment tester (ERT-2), used for the measurement of as-received billet enrichment, is in the final stages of evaluation by N Fuels Process Control. A series of ten measurements were made over a two-week period to measure, on a statistical basis, the accuracy of the tester. The measurement error of this test was determined to be 4-1/2% at the 3-sigma level (± .06% in terms of enrichment). This figure is better than the previously obtained measurement accuracy which included some process variable error. The sensitivity of this test has been determined to be about 40,000 cpm per 1% change of enrichment when using a difference count method to compensate for daughter product buildup.

The instruction manual for this tester is in final stages of preparation.

The TIG braze closure test, which is used to determine whether or not the braze wire melted during the welding cycle, was successfully applied to the fuel elements that are ready for reactor charging. It appears unlikely that the definition of this tester will be as good as the existing UT-9 test on the flat capped fuel element.

The chevron end tester, used to test the bond quality of the beveled portion of the chevron end cap, has been used to inspect the initial production run of enriched driver tubes. In general, test results were not satisfactory. Major effort is required before March 1, 1965, to provide a suitable production line test.

Fabrication of the mechanical portions of the remotely operated, irradiated fuel test station (UT-10B) for use in 105-N Fuel Examination Basin, is approximately 98% complete. The pulser-amplifier chassis will be incorporated into the motor platform assembly to permit the usage of 15-ft. long cables on the bond test.

NIT fuel #1450-1, ultrasonically tested at 105-KE, was sectioned at Radiometallurgy Laboratory and found to have two large radial cracks. The ultrasonic map of this fuel had, in fact, indicated two cracks. A number of smaller cracks were also noted in the sectioned samples; it is surmised that some or all of these were caused by the cut-off saw.

An eddy current penetration tester has been renovated for use in the development of a stria detection test for irradiated N fuels. The sensitivity is low and modifications are required to obtain a workable signal in the low conductivity Zircaloy-2. A change in frequency from 20 kc to 120 kc is about 48 mils, whereas striae are expected to be found at depths ranging from 13 mils to 32 mils. An experiment with the "Phasorscope" showed prob-emotion to be in the same direction as Zr-U conductivity change. Seven NIT fuels with built-in striae have been extruded for standards. Unfortunately, the outer surface came out rough and they must now be machined to obtain a workable smoothness. A test head and rolls assembly were set up for performing laboratory tests.

N Reactor Testing

Procurement of a tester to check fuel element enrichment just prior to reactor loading is being considered by NRD. It was recommended that a short engineering study be conducted to provide data essential for the design of this equipment. A request for such a study is to be initiated shortly.

IPD Fuels Testing

Design of an updated model of the existing UE-1 tester in 313 Building is being considered by Quality Control, IPD Fuels.

Work has been initiated on the scoping of an improved fuels handling system. Scope prints will be issued for comment prior to starting on the detail design phase.

Development has been initiated on prototype dual-bond tester to monitor the internal bond quality of AlSi fuels from the outer fuel surface. Incentive for the new test is that it eliminates the internal probe which has been a persistent source of trouble during production line operation. Feasibility of the new technique has been demonstrated; efforts are now being made to produce a working prototype model. To date, very fast gating circuits have been developed and tested and a wide band nuvistor amplifier is nearing final evaluation.

NEUTRON CROSS SECTION PROGRAM - 02/04 PROGRAM

Triple-Axis Spectrometer

The design was completed and fabrication started on a new shield for the scattered-neutron detector of the spectrometer. This shield is to accommodate a 2-inch diameter end-window BF_3 detector. This detector should reduce the non-uniformity of the spatial sensitivity of the analyzing spectrometer. Several Scattering Law Measurements on 22°C H_2O were made, but the results of the measurements have been delayed in Data Processing. A neutron-sensitive phosphor and Polaroid film was used for imaging the monoenergetic neutron beam of the spectrometer.

Time-of-Flight Spectroscopy for Slow Neutrons

Measurements of the performance characteristics of the TOF spectrometer were continued. This work has been severely hampered by malfunctions of the readout of the TOF analyzer. A water sample of 10-mil thickness was prepared and X-ray tests showed bubble formation. Ultrasonic techniques are now being tried to eliminate bubbles and to check the integrity of the H_2O film. A FORTRAN program is being written to calculate differential inelastic scattering cross section values from the Scattering Law output of the LEAP and ADDELT programs.

Fast-Neutron Cross Sections

The results of several previous measurements of 3- to 15-MeV total cross sections were subjected to analysis in order to recover data which were in error due to various instrumental malfunctions. This analysis resulted in a good set of data for arsenic and some reliable results at lower neutron energies for eight other elements. Work continued on the new RF deflection and beam pickup system. The time-of-flight instrumentation suffered no direct damage in the flooding of the 3745-B Building. Work continued on required modifications to computer codes, including converting the Howerton cross section library tape to binary form. A number of errors in the listings on the Howerton tape were discovered. A request was received from LASL for the measurement of Li^6 and Li^7 cross sections.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

PRTR Burnup Experiments

Three irradiated PRTR fuel elements are currently being destructively analyzed. Two of the elements are mixed oxide fuel elements, one of which initially contained 1.0 w/o PuO_2 and one 0.48 w/o PuO_2. The 1.0 w/o PuO_2 mixed oxide fuel was irradiated to $\sim$1000 MWd/t and is the first 1.0 w/o element to be analyzed. The 0.48 w/o PuO_2 mixed oxide fuel has an accumulated exposure of $\sim$4900 MWd/t and is the second 0.48 w/o element to be analyzed. The third element being analyzed is a UO_2 element which was exposed to 4000 MWd/t. This is the fifth UO_2 element analyzed to date.

PuO_2-UO_2 Graphite Lattice Studies

Preliminary analysis of the experimental data for the 19 rod clusters on 8-3/8" pitch yield a k_∞ of 1.21. For this calculation the following (average thermal) values were assumed: $\alpha^{239} = 0.44$, $\alpha^{235} = 0.172$.

The results of chemical analyses of samples of the fuel material raised doubts about the homogeneity of the fuel. Hence the rods were autoradiographed and additional samples submitted for chemical analysis. The autoradiographs indicate that the core material is uniform. However, to remove any further doubts, relative reactivity measurements are being made on selected rods in the PCTR. Chemical analyses indicate that the percent PuO_2 in the fuel is close to the nominal value of 0.90 w/o. However, the ratio of $\frac{Pu}{PuO_2}$ is closer to 0.853 than to the stoichiometric value of 0.882. Thus the percentage of plutonium in the fuel is 0.766%, which is ~ 3% lower than the expected 0.79%.

Critical Experiments with PuAl Fuel

The PRCF was loaded to critical using 1.8 w/o PuAl fuel and H_2O moderator. In order to maximize the number of fuel rods needed for a critical loading, a three-zone loading was used in which the fuel rods with the highest Pu^{240} content were placed in regions of highest fuel element worth and those with the lowest Pu^{240} content were placed in regions of lowest fuel element worth. With this loading the PRCF is critical with 486.9 fuel rods. This number is only 0.6 larger than the number arrived at with the zones reversed.

An increment of a control rod penetration and an increment of moderator height were calibrated in the new core loading. An irradiation was performed to determine the radial flux distribution.

The reactivity worths of a number of 1.8 w/o PuAl Lx fuel rods were measured in the center of the PRCF and were found to be very non-uniform. Fuel elements which nominally have the same amount of plutonium and the same percentage of Pu^{240} differ by as much as 25%, or 200 micro-k, in reactivity worth.

By contrast, when four Hx rods (2.015 w/o PuAl) were measured, they were uniform to within 5 micro-k.

Cadmium covered Lu_2O_3 pins irradiated in the PuAl core have been counted. This completes the vertical and cell traverse measurement of the Lu cadmium ratio. The cadmium covered Lu data are not yet analyzed. The bare Lu data are about 50% analyzed.

Cadmium covered Pd and Au were irradiated in the PuAl core to determine the shape of the slowing down spectrum. Each of three pairs of 1/2" Pd and Au foils were placed in symmetric positions from the core center. The vertical positions are 9" below center and 3-7/8" and 16-3/4" above center along a 48" fuel length. A similar Pd-Au measurement was reported previously for the PuO_2-UO_2 core. The PuAl core will give a good comparative measurement of the slowing down curve.

Values of materials placed in the center of the 0.80 inch, 1.80 w/o PuAl lattice are listed in nk/gm; Lucite - 63.4, Teflon - 1,176, Aluminum - 1,913, Lexan - 527.3, Polyethylene + 510.5. These values are the worth when the material replaces water.

The value of aluminum in a void was measured to be -912.2 nk/gm.

These values indicate that Teflon is a reasonably good void when it is placed in a close packed water moderated core.

A recording of the reactor noise from the 1.8 w/o PuAl loading in the PRCF was analyzed to obtain the transfer function. An equation, which contains as one of its parameters β/ℓ, was fitted to the data by the method of least squares. A preliminary value of β/ℓ is 34 ± 4. The large error is believed to be caused by a malfunctioning recorder which was used to play back the recording. The recording will be reanalyzed with different equipment.

Flux data obtained after inserting negative reactivity into the PuO_2-UO_2 loading in the PRCF have been partially analyzed. A series of four exponentials has been fitted to the data, by the least squares method, for times between 0 and 2 seconds. Data for times between 5 and 300 seconds were recorded by a digital voltmeter. A series of three or four exponentials which describe the shorter transients is being fitted to these data. The parameters for these exponentials will then be incorporated into the fitting process for the data between 0 and 2 seconds in an attempt to obtain a consistent set of parameters which will describe all of the data.

In order to improve future kinetics experiments, the equipment which performs frequency analysis of the reactor noise recorded on magnetic tape has been improved. Analysis of white noise is under way as a final test. This system makes use of twin-t frequency filters and differential operational amplifiers instead of a band-pass filter and D. C. operational amplifiers previously used. Much of the circuitry developed in this apparatus will be used in the construction of a system capable of analyzing 10 frequency channels simultaneously.

PRTR - HPD Core

Temperature coefficients have been computed for the proposed high power density core in the PRTR and issued as Appendix B of HW-83483, A Proposed Physics Experiment in the PRTR. Moderator void coefficients were recomputed using HRG instead of GAM. The results showed only minor variation from those published in HW-83483 A.

Physics Code Developments

1. GRAFAN - Program GRAFAN has been coded, debugged, and released for customer use. GRAFAN in conjunction with program BARNS reads the anisotropic scattering parameters from the Hanford Basic Library tape, computes the cross section as a function of cosine of the scattering angle, and provides a plot using the Benson-Lehner plotter.

2. SCHWAC - Program SCHWAC has been coded, debugged, and released for customer use. SCHWAC homogenizes an immediately preceding HFN case and can add the results to an HFN data tape as a new material. Thus the various cells of a reactor can be homogenized using HFN followed immediately by a run with HFN for the reactor as a whole.

3. HFN - An error which gave incorrect power and flux normalization has been eliminated. The corrected version is currently in the SPL library.

4. TEMPEST - Program TEMPEST has been successfully converted to FORTRAN IV. The difficulty of reading binary cards on the input tape (A2) has been solved. To read in binary on A2 requires that a FORTRAN IV buffered "READ" has not at that time been executed by the calling program.

5. Hanford Basic Library - This was updated this month to include better anisotropic scattering coefficients for deuterium, carbon, nitrogen, oxygen, zirconium, and lead. Protactinium-233 was also updated to include the latest available information. The new decimal library now consists of 24,336 cards. An RBU Library Editor listing of the current version of the Hanford Basic Library is available for printing purposes.

6. BARNS-II - The first offsite distribution of the BARNS-II code to process cross sections from the Hanford Basic Library was made when a copy of the code was sent to M. Temme of Lockheed Missiles and Space Vehicles Company, Sunnyvale, California. Included in the package were subroutines to calculate elastic and inelastic downscattering transfer probabilities from stationary targets. Because debugging has not been completed, the code is being given quite limited distribution at this time.

7. TWENTY GRAND - The two dimensional neutron diffusion theory code TWENTY GRAND, developed at Oak Ridge National Laboratory, has been obtained through the Argonne Code Center and adapted to Hanford use. This code will handle from one to six groups in cylindrical or slab geometry and permits transfers between all groups. As many as 3000

mesh points may be used. Both direct and adjoint fluxes may be calculated.

8. HRG - A question by a puzzled customer has led to the discovery that the interpolation scheme used in HRG and GAM to calculate the surface term contribution to the resonance integral calculation is inadequate. Tests using an auxiliary code reveal that the currently used scheme, which assumes direct dependence on the pertinent variable, should be replaced by one assuming logarithmic dependence on the variable. This change is being made in HRG.

   Two offsite requests for HRG have been received and honored during the month.

9. ZODIAC Modifications - A method of operation has been devised for ZODIAC, whereby a superposition of two or more spectrum approximations may be used. If, for example, one wishes to study a water-cooled graphite moderated reactor, a choice must be made in TEMPEST between the Wilkins and Wigner-Wilkins approximations. The "true" spectrum can only be found by a space dependent multi-kernel code such as THERMOS. Once found, however, a good approximation can probably be obtained by mixing the less sophisticated spectrum models available in TEMPEST. The mixing operation in ZODIAC involves running separate TEMPEST cases with different spectrum or temperature options, and combining them in HFN. This technique will significantly broaden the scope of reactor configurations for which automated burnup may be carried out. Debugging of this scheme is currently under way.

Data-Theory Correlation - Mark-T, RBU Checkout

A test case was run to check the fission portion of the Monte Carlo collision routine. The number of neutrons from collision was correct as well as the number of neutrons expected from collision. The fission neutron spectrum will also be checked.

The properties of the plutonium nitrate solution filled sphere experiment, No. 2243 of the Critical Mass Laboratory, are being analyzed. The diffusion code multiplication constant was 0.99907 after one hour and stayed within 0.5% of this value for three hours. After four hours the value was 0.99953. Even with the small variation in the diffusion code multiplication constant, other parameters such as the average cross section have not converged everywhere to a constant value. Statistical analyses of the results are continuing.

Isotopic Analysis of PRTR Samples

Isotopic analyses were provided on 39 burnup samples of PRTR-irradiated, mixed-oxide fuel elements. Of these, 2 were uranium samples from fuel element number 5185, 8 were uranium and 5 were plutonium samples from fuel element number 5187, and 24 were uranium samples from fuel element number 5213.

Analyses were provided on 10 samples in cooperation with Analytical Chemistry's development of an isotopic dilution double-spike procedure. This procedure which uses U^{233} and Pu^{242} as spike materials will permit more accurate uranium-to-plutonium ratio measurements in support of PRTR-irradiated fuel studies.

Instrumentation

General circuitry design was partly completed on the reactor time-to-power meter instrumentation, which is being developed to aid in simplifying the startup of reactors.

Detailed calibration tests were planned for the fuel element underwater gamma scanning system. These will be conducted when the new collimator head has been completed in fabrication and installed in the system.

EBWR PROGRAM

Estimates of the effects which the Lexan templates and other materials have on the critical mass of the EBWR fuel in a 0.71-inch lattice have been completed. The worth of the Lexan was evaluated using a radial traverse of the worth of a styrofoam void and a vertical traverse of the worth of a tube void. The worth of a Saran guide tube for wire irradiations was determined to have a maximum worth of -0.2 fuel rods. The effects of the bottom aluminum support plate were negligible. The three Lexan templates were worth -11.3 fuel rods, while the three Lucite templates used in the approach to critical tank were worth -2 fuel rods.

Additional analysis on the power level calibration indicates that the original calculations were conservative by a factor of two.

PHOENIX FUEL PROGRAM

MTR-Hx-Pu Fueling Proposal

A proposal regarding the MTR Hx-Pu burnup experiment has been submitted to the AEC. The proposal summarized preliminary design calculations and cost estimates.

Reactivity Effects of Transplutonium Isotopes in Phoenix Cores

The reactivity effect associated with the buildup of transplutonium isotopes in "Phoenix" cores is presently being investigated. A calculation for a typical 100 MW Zr/H_2O Phoenix core using very high exposure Pu has been completed. The buildup of the transplutonium isotopes (Am and Cm) introduces an increasingly negative component into the system as follows:

EFPH at 100 MW	$\Delta\rho$
0	0
1838	- .0034
3652	- .0070
5435	- .0109
7174	- .0148

The calculated transplutonium isotope buildup is tabulated below:

EFPH	Am-241 (gms)	Am-243 (gms)	Cm-243 (gms)	Cm-244 (gms)
0	0	0	0	0
1838	127.6	192.7	15.0	11.1
3652	223.8	423.9	54.3	48.9
5435	283.0	700.8	109.1	122.2
7174	300.7	1029.7	171.6	243.5

Pu-Al Polyethylene Experiments in the PCTR

In preparation for the PuAl PCTR experiment, initial calculations of the amount of B-10 needed to poison the 20 w/o Pu in Al polyethylene system to a reactivity of unity have been carried out. The methods used were primarily those incorporated in ZODIAC. Both light and heavy moderator approximations were used in TEMPEST and were compared to a SPECTRUM 6B calculation which employed a polyethylene kernel. It appears that the polyethylene spectrum can be approximated by splicing Wilkins and Wigner-Wilkins spectra. With this in mind, the TEMPEST option of changing from the Wilkins to the Wigner-Wilkins approximation at some specified energy was utilized. For a changeover point of 0.1 eV, the k_∞ values for each method are given below.

Spectrum:	Wilkins	Wigner-Wilkins	SPECTRUM	0.1 eV Splice
k_∞	0.9127	0.8484	0.8762	0.8759

Pu-Be Studies

Survey calculations for some proposed Pu-Be critical experiments were completed. The critical mass for homogeneous mixtures of three plutonium compositions and metallic Be moderator in spherical geometry were estimated as a function of reflector thickness. The small difference in critical mass observed at 600 and 800 Be/Pu^{239} ratios for high exposure plutonium (20% Pu^{240}) is due to a calculated minimum in k_∞ which occurs in the 700 to 900 region. Resonance absorption in Pu^{240} was calculated using self-shielding factors which normalize the resonance integral to NRIA approximation values. A 20 kg inventory of up to 20% Pu^{240} plutonium should permit the construction of Pu-Be critical systems with the currently available beryllium supply.

Self-shielding factors for 20 w/o PuAl alloy plates were calculated as a function of plate thickness. Values in the 0.85 range were obtained for 20 mil plates using first flight transport theory which is justified at the plate separations appropriate to the fuel to moderator ratios of interest. The values are low enough to suggest that new critical mass estimates should be obtained using shielded cross sections. Further iterations are currently in progress.

Studies on the effect of fuel grain size are of particular interest in the Pu-Be system. Efforts to run the resonance absorption code ZUT, which can be used to calculate grain effects, have been successful only for homogeneous systems so far.

FAST REACTOR STUDIES

Compact Fast Reactor Studies

The study of the relative merits of PuN, $U^{23}O_2$, and $U^{25}O_2$ as fuels for compact tungsten fast reactors is continuing. To date, the following aspects of this study have been completed:

1) A survey of critical size for various fuel compositions and void fractions.

2) The effect of coolant on reactivity.

3) The effect of thermal expansion on reactivity.

An investigation of the reactivity worth of rotating drums in the reflector is in progress.

Physics Analysis of a Fast Flux Test Reactor (FFTR)

A parametric survey of critical size and required fuel inventories was performed for a series of PuO_2-stainless steel, fast reactors. The major emphasis in this survey was placed on finding the effect of variations in fuel composition and length to diameter (L/D) ratio. This work is presently being extended to include analyses of control statics characteristics.

Pu-Al Light Water Experiments in the PCTR

The vendor for the 10,000 discs of borated polyethylene is unable to produce satisfactory discs of 20 mil thickness. The order has therefore been changed to 5,000 discs of 40 mil thickness, and the tolerance has been relaxed from ± 2 mil to ± 4 mil. These discs will be tested for uniformity and absolute content of boron by reactivity measurements in the PCTR prior to beginning the Pu-Al experiment.

A chemical test has been planned to determine the accuracy and precision of the chemical method used to determine the Pu concentration in the Pu-Al discs. The results of this test will be used in a statistical analysis of the Pu concentration data to establish error limits for each disc.

HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

A mockup assembly of the HTLTR has been successfully operated at 1080°C--a temperature somewhat above the maximum design temperature, 1000°C, at which the reactor is to be run. The present loading of the mockup consists of a 2 x 2 x 10 foot block of graphite traversed by four graphite heater rods and surrounded by about 1000 cubic feet of insulating alumina brick. These materials and a nitrogen atmosphere are contained within a gas tight steel envelope. This initial run is thus a test of the HTLTR design concept for the moderator, heaters, and insulation and of the compatibility of these materials with themselves and with the nitrogen atmosphere. The environmental conditions during the run were similar to those to be expected in the reactor itself. The data obtained on gas contaminant evolution from the insulation and graphite will therefore be useful in predicting the amount of corrosion in the reactor. Temperatures at several locations were measured with a variety of types of thermocouples and at a few locations with an optical pyrometer. The thermocouples were sheathed in either Hastelloy-B or molybdenum. Only the Hastelloy-B sheathed couples survived the temperature excursion to 950°C. Although the system has not yet been opened for examination, some picture of what has happened inside is given by the graphite samples which could be inserted into and removed from the mockup periodically. Weight losses of these samples were insignificant up to about 700°C. At 850°C, the weight loss, if scaled up to the HTLTR and excluding graphite transport to the heat exchanger, would correspond to about 100 pounds of graphite over a ten-year period of normal operation.

A test was completed in which a matrix of B_4C in graphite was maintained at 1100°C in nitrogen for 800 hours. The sample was held in an open ended container of T.D. nickel. There was no visible reaction between these materials. A detailed micrographic analysis of the materials has not yet been made. Another test was started in which samples of UO_2, Al_2O_3, and Al_2O_3 mixed with Gd_2O_3, contained in sealed T.D. nickel capsules, are being heated to 1100°C in a nitrogen atmosphere. These materials will be used in the HTLTR fuel elements and in dummy elements. The purpose of the latter is to control the temperature coefficient of reactivity of the reactor.

Manufacturer's data and information from users is being used to write a specification for the neutron generator on HTLTR. The generator will be used for a startup source and for some experiments. Neutrons will be produced by bombarding a tritium target with deuterons. Whereas most applications require good focusing of the deuteron beam our application does not, but it does require high neutron production and constant source strength. Target regeneration systems, both continuous and intermittent, are being considered.

A mathematical model of HTLTR that uses rectangular boundaries instead of the cylindrical boundaries is being developed. The goal is to have a flux map across the reactor with discrete fuel elements and control rods in the reactor. However, it is necessary to go by steps to such a detailed picture and at the present stage the fuel still has to be smeared into a homogeneous blanket around the core. When finished, the method of analysis should apply to the PCTR with little modification because the geometries of the PCTR and the HTLTR are very similar. Application to PCTR where results can be compared to measurements will serve as a check on the method.

Development and acquisition of the HTLTR instrumentation system is moving rapidly forward. A Commission hosted bidders' conference for the computer system was held on November 30, and extensive effort has been made to answer numerous bidder technical questions which arose subsequent to the conference. Bids were received December 21 and are being reviewed. A revised instrument design criteria was written and issued for comment.

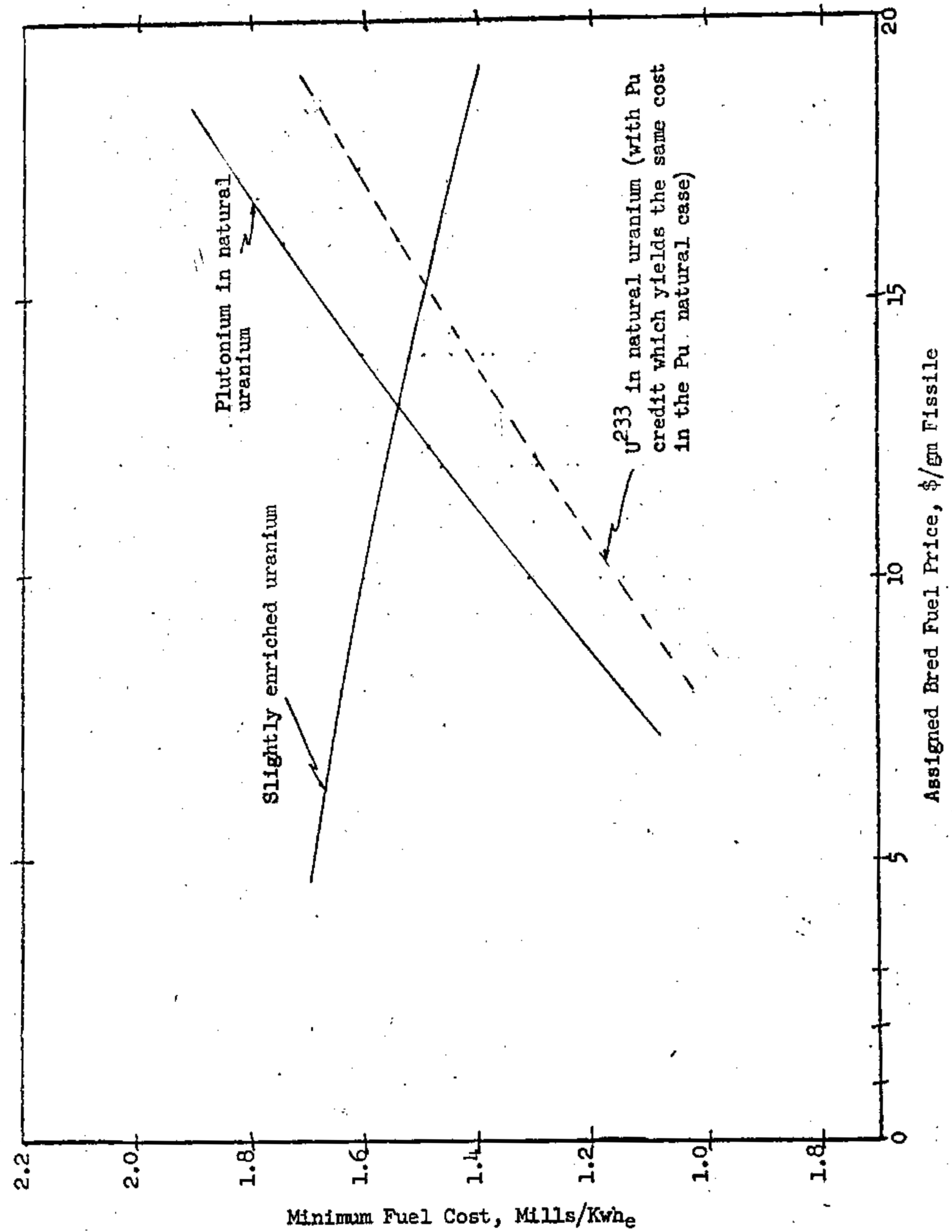
Development of a flux digitizer for the HTLTR instrumentation system proceeded with work performed on a vibrating capacitor modulated amplifier. Tests with the experimental instrument indicated that linearity is obtained over an input current range from 10^{-9} to 10^{-11} amperes with usable outputs down to 10^{-12} amperes. Higher amplifier gain will be required for adequate system stability.

FUEL CYCLE ANALYSIS PROGRAM

Relative Value of Bred Fuels as Enrichment in Pressurized Water Reactors

Plutonium and U^{233} bred fuel values are calculated by comparing their performances with that of U^{235} in the form of slightly enriched uranium in the case of plutonium values and fully enriched uranium mixed with thorium in the case of U^{233} values. Most published values of bred fuels are made by assuming that the reactor fuel geometry is fixed and that the bred fuel is substituted in the lattice directly for U^{235}. This technique is proper when bred fuel makes up a small portion of the enrichment, or if it is very expensive to modify the fuel lattice in any way. This may be the situation when bred fuel is first used in power reactors, but as the use of bred fuel becomes extensive reactors will be especially designed to maximize the performance of bred fuels.

The effect of fuel cell optimization on fuel values and the relative bred fuel value have been analyzed in a pressurized water reactor by varying the lattice spacing. Thus, for each type of fuel considered, the initial enrichment and lattice spacing were optimized to yield the lowest possible fuel cost for an assigned fuel price. Then, by comparing the minimum fuel cost of each fuel, the indifference values were determined. Simple indifference values are found by the intersection of the bred fuel cost curve and the slightly enriched uranium cost curve plotted against the bred fuel price. Figure 1 shows these curves for slightly enriched uranium, plutonium enrichment of natural uranium where the initial plutonium composition is 76% Pu^{239}, 18% Pu^{240}, 5% Pu^{241}, and 1% Pu^{242}; and U^{233} enrichment of natural uranium where the initial composition is 76% U^{233}, 18% U^{234}, 5% U^{235} and 1% U^{236}. This system was idealized by assuming that the U^{233} in U^{238} was chemically separable after irradiation. Hence, this study is indicative of the incentive to fabricate fuels such that U^{233} could be isolated from U^{238} after irradiation. One intersection shows that plutonium would have a value of \$13.10/gram fissile and, from the intersection fuel cost of 1.54 mills/kwh$_e$, U^{233} will have a value of \$16/gram, where the plutonium credit is \$13.10/gram. This means U^{233} is 1.22 times more valuable than plutonium as a substitute fuel for U^{235} as enrichment for natural uranium in pressurized water reactors with optimum neutron spectrum. The relative values can be compared at any fuel cost shown on Figure 1 and the ratio of value will be nearly the same. This indicates that the value ratio (U^{233} value/plutonium value) is constant over the range of prices and neutron spectrums examined for the pressurized water reactor. The ratio



BRED FUEL INDIFFERENCE VALUES FOR PRESSURIZED WATER REACTORS

FIGURE 1

is different for other reactor types and other fertile fuels. These effects are being investigated at this time.

Value of Plutonium Fuels Successively Recycled for Ten Years

Computed plutonium value curves have been presented (November 1964 monthly report) where value is plotted against cumulative exposure or Pu^{242} concentration. These curves show values change at different rate for different reactor types. A number which is often of interest is the value over a given time span. Table I shows the first cycle discharge value (initial) and the value after 10 years of successive recycle in that particular reactor type using the present cost of uranium. Reactors with low specific power (MW/M.T.) show much less value drop than those with higher specific power. Table II shows this same data for a lower natural uranium price which may exist in the future. The specific power of each reactor type including the effect of an 80% load factor is tabulated below:

Reactor Type	Specific Power (MW/M.T.)
PWR	20.57
BWR	11.16
D_2O	53.60
SGR	25.47

Fuel Cost Parameter Study

At telephone request by AEC-Washington a special study was completed regarding the impact on minimized fuel costs of private ownership, uranium costs, uranium cascade costs, fuel jacketing costs, and chemical separation plant costs. The data were automatically plotted after processing through the MELEAGER (Physics) and QUICK (Cost) codes. Annual interest rates on the fuel varied from 6% to 12%. Uranium ore was priced at $13.10, $17.30, $22.50, $32.00, and $40.00 per kilogram as UF_6 for four costs of separative duty: $15, $30, $45, and $60 per kilogram uranium. Fuel jacketing costs varied from $30 to $120 per kilogram uranium. The costs of chemical separations were set at $10, $20, $40, and $60 per kilogram uranium. Minimized fuel costs were determined and plotted for all combinations of the foregoing cost parameters for three reactor simulations, the PWR, BWR, and HWR.

Self-Shielding Factor Calibration for the MELEAGER Code

The effective cross sections used by the MELEAGER burnup code are determined using Westcott's formulation

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TABLE I

PLUTONIUM VALUES FOR A TEN-YEAR PERIOD WITH U_3O_8 AT $8/lb

		6		8		10		12		ΔFEFJ	
		Initial	10 yrs	Initial	10 yrs	Initial	10 yrs	Initial	10 yrs	Initial	10 yrs
Self-Produced	BWR	10.75	10.25	10.80	10.38	11.19	10.50	10.97	10.27	10.80	10.02
(Pu^{239} α = 0.57)											
Self-Produced	PWR	11.70	11.15	10.65	9.30	10.30	7.95	X	X	9.90	7.70
Self-Produced	SGR	12.71	10.05	12.85	10.20	13.05	10.55	14.65	12.00	12.37	10.05
Self-Produced	HWR	6.45	4.00	9.60	7.30	9.55	7.80	X	X	8.55	6.00
50% Self-Produced	BWR	-	-	-	-	12.55	12.55	-	-	-	-
50% Self-produced	PWR	-	-	-	-	10.75	9.30	-	-	-	-
50% Self-Produced	SGR	-	-	-	-	12.19	10.65	-	-	-	-
50% Self-Produced	HWR	-	-	-	-	X	X	-	-	-	-
Unlimited	BWR	-	-	-	-	-	-	-	-	-	-
Unlimited	PWR	-	-	-	-	-	-	-	-	-	-
Unlimited	SGR	-	-	-	-	-	-	-	-	-	-
Unlimited	HWR	-	-	-	-	X	X	-	-	-	-
Self-Produced											
(Pu^{239}, α = 0.51)	PWR	-	-	-	-	15.80	12.20	-	-	-	-
(Pu^{239}, α = 0.46)	PWR	-	-	-	-	19.40	17.25	-	-	-	-

ΔFEFJ

- Case will not be run
X Case not completed, is to be run

TABLE II

PLUTONIUM VALUES FOR A TEN-YEAR PERIOD WITH U_3O_8 AT \$6/lb

		6		8		10		12	
		Initial	10 yrs	Initial	10 yrs	Initial	10 yrs	Initial	10 yrs
Self-Produced	BWR	10.50	10.00	9.73	9.10	X	X	9.75	9.10
	PWR	X	X	8.95	7.30	9.45	7.85	8.75	6.70
	SGR	11.20	9.00	11.35	9.20	11.45	9.40	11.65	9.40
	HWR	X	X	5.09	3.50	6.95	5.40	8.09	7.10
50% Self-Produced	BWR	-	-	-	-	X	X	-	-
	PWR	-	-	-	-	X	X	-	-
	SGR	-	-	-	-	10.65	9.40	-	-
	HWR	-	-	-	-	X	X	-	-
Unlimited	BWR	-	-	-	-	-	-	-	-
	PWR	-	-	-	-	-	-	-	-
	SGR	-	-	-	-	-	-	-	-
	HWR	-	-	-	-	X	X	-	-

- Case will not be run
X Case not completed

$$\sigma = \sigma_0(g + rs)$$

where

σ_0 = 2200 m/sec cross section
g = ratio of the effective cross section to a 1/v cross section
s = excess resonance integral

Both g and s are functions of the neutron temperature.

The MELEAGER code calculates an effective value for s by the formula

$$S = \left(S_\infty + bg_\infty \quad 1 + \frac{YZ}{SCA}\right)^{-\frac{1}{2}} - bg$$

where

$S_\infty + bg_\infty$ = resonance integral for isotope at infinite dilution
Y = concentration of isotope, nuclei/b-cm
Z = height of low-lying resonance, barns (self-shielding factor)
SCA = scattering term (See HW-68100)
bg = effective resonance integral corresponding to the effective thermal cross section

I_{eff} and the dilute resonance integral, I_∞, can be substituted for $S + bg$ and $S_\infty + bg_\infty$. Then, for a given lattice cell where I_{eff}, I_∞, Y, and SCA are known, a value for Z can be calculated.

In order to obtain an I_{eff} for a given isotope in a particular cell, the GAM and THERMOS codes are being used. The GAM code provides information from which I_{eff} from 10 MeV to 0.683 eV can be calculated, and the THERMOS code provides the additional information necessary to calculate I_{eff} from 0.683 eV down to a particular cut-off energy, μkT, for the MELEAGER code. The value for I_∞ is taken from the RBU basic library by the GANDS code (HW-81117). Calculations are in progress to determine whether this value for I_∞ is the same as the one obtained from the GAM-THERMOS combination for a dilute system.

At the present time, results of critical experiments in the EBWR are being used to obtain shielding factors for the plutonium isotopes. These values will then be applied in the MELEAGER code to determine k_∞ for the cell which will be compared to the critical experiment results.

MELEAGER-ALTHAEA Code Development

THERMOS and RBU calculations the past few months have indicated lower reaction rates than MELEAGER calculations for isotopes which have a large resonance at the low energy end of the epithermal range (such as Pu^{239}). The difference is due to the deviation of the epithermal flux from the theoretical 1/E flux assumed in the derivation of MELEAGER cross sections. A first stage correction for the epithermal flux shape has been derived and incorporated into MELEAGER and is now being tested. The resonance integrals of all fuel isotopes were divided into two portions using 0.68256 eV as the dividing line. The low energy portion is dependent upon the Maxwellian temperature (MELEAGER uses a low energy cutoff at 3.681 kT) so the s or "excess" resonance integral is represented by a fourth order polynomial in T. The upper resonance integral is not dependent upon neutron temperature so a single value for each isotope is adequate for the input data. Reaction rates in the lower resonance integral are now reduced in proportion to the fractional loss of neutrons due to the upper resonance integral. In addition, the isotopes defined by input as fertile isotopes are treated in a special way. They must be arranged in the order of the decreasing energy of the principal resonance; for example, in the present list, isotopes 5 through 8 are defined as fertile and are placed in the following order:

Isotope	Sequence Number	Energy of Lowest Major Resonance
Th^{232}	5	22
U^{238}	6	7
U^{234}	7	5
Pu^{240}	8	1

The resonance integral of the first fertile isotope is not subject to any further correction, but resonance integrals of the remaining fertile isotopes are reduced by multiplying by the resonance escape probability from the preceding fertile isotopes. During this recompilation of MELEAGER-ALTHAEA memory allocation was made for subdivision of the upper resonance integral into four subintegrals in the event refinement of this scheme is warranted. Table III presents typical dilute resonance integrals for a neutron temperature of 300°C and the percent of this integral above the 0.68256 eV energy. For comparison the percent of each integral above 4 and 20 eV is also included.

Note that only Am^{241}, Pu^{241}, and Pu^{239} have most of their resonance integrals below 0.68256 eV and that they will be the isotopes most affected by the revision. Also note that most of the isotopes have over half of the

TABLE III

DILUTE RESONANCE INTEGRALS OF MELEAGER ISOTOPES

Isotope	Resonance Integral at 300°C (0.182 eV)	Percent of Integral Above Indicated Energy		
		0.68256 eV	4 eV	20 eV
Th^{232}	85	96.7	94.7	93.7
U^{238}	282	99.6	99.3	53.6
U^{234}	368	90.4	86.7	18.0
Pu^{240}	8580	97.1	1.7	1.7
U^{236}	324	99.3	98.5	26.8
Pu^{242}	1047	99.2	5.0	4.8
Am^{243}	1163	99.9	11.6	1.1
Pa^{233}	667	97.4		
Np^{237}	1384	78.7	58.7	44.9
U^{233}	1033	75.8	27.7	15.5
U^{235}	618	65.5	52.2	31.6
Pu^{238}	326	53.5	24.4	13.0
Am^{241}	2945	38.2	12.4	4.6
Pu^{241}	1671	29.8	26.7	8.0
Pu^{239}	2930	16.5	14.6	8.0

resonance integral below 4 eV, while only Th^{232} and U^{238} have most of their resonance integrals above 20 eV. Calibration calculations are in process to compare MELEAGER data with GAM-THERMOS computed data and with Yankee experimental data. It may prove desirable to further subdivide the upper resonance integral near 4 or 20 eV, or both, to obtain reaction rates indicated by experiment.

VESTA-Fuel Utilization Code

The case changer system for the BESTA code has been debugged and is working properly. The case changer significantly reduces the number and complexity of the VESTA computer data cards, and is now an integral part of the VESTA code. The VESTA code, which calculates the uranium requirements for hypothesized future nuclear power growth projections, is also being modified to include calculation improved plutonium recycle techniques. The following reactor types are now, or soon will be, simulated for a wide range of recycle rates and uranium price schedules:

A. Thermal Reactors

1. Boiling water reactor
2. Sodium-graphite reactor
3. High temperature gas-cooled reactor
4. Heavy water reactor
5. Pressurized water reactor (PWR)
6. Modified PWR with high specific power and low alpha (thermal neutron capture-to-fission ratio) as parameters.

B. Fast Breeder Reactors

1. Plutonium oxide fueled - 20-year doubling time
2. Plutonium carbide fueled - 7-year doubling time
3. Plutonium carbide fueled - 3.5-year doubling time
4. U^{235} enriched UO_2 fueled

Some preliminary studies indicate that, in the middle of the next century when breeder reactors become dominant and the rate of growth of electricity requirements has dropped, the supply of tails material from cascade operation will be used. Then U^{238}, which will be required for breeder reactor inventories, will become the valuable constituent of uranium ore. This situation will occur at a much later time if the thorium cycle is used to supply a significant portion of the total electricity requirements.

NEUTRON FLUX MONITOR PROGRAM

Continued correct performance was achieved at the KW Reactor facility with the two uranium regenerating in-core neutron flux monitors and continuous data are being recorded. The original flux-measuring wire has been discharged and replaced by a second wire and arrangements were made to determine the activity of the detectors. It was noted during a second reactor startup that the dynamic range of the monitors appeared to be an order of magnitude less than during the initial startup; this may be due to either chamber activation or simply to fission product buildup in the reactor.

Analysis of the data obtained during KW Reactor tests of the B-11 neutron flux detector and cables indicates a strong dependence of the noise signal generated on the cable dimensions. This dependence, however, is not a simple relation and additional factors appear to similarly influence the signal. A least-squares-fit series of calculations were made using the various cable cross sections and the readout data, and an empirical formula was developed as a function of insulation area cross section and area of the center conductor of the cables. The equation, developed for Al_2O_3 insulation and a stainless steel center conductor, is independent of neutron flux, and the results indicate that a cable with a ratio of insulation area to center wire area of about 60 should not cause a steady-state current in the KW Reactor environment. Further testing is planned to verify the conclusions. In addition, Co-60 gamma pit measurements (to 2×10^6 R/hr) were made with various cables; resulting currents ranged from 3 to 8×10^{-11} amperes.

Improved readout instrumentation is being prepared for use in the next in-reactor tests. Experiments with a bi-polarity logarithmic pico-ammeter appeared successful. In addition, design was completed on an improved B-11 detector without voids and with improved mechanical stability.

Microwave in-core neutron flux monitoring activities established plans to conduct two more in-reactor experiments and assembled waveguides for the first test. First attempts will be to measure plasma generation in He-3 and in He-4 (5400 barns and essentially zero barns cross-sections respectively). Both capsules for the gases have been fabricated and filled. The planned tests will provide direct comparisons of plasma generation in the two gases. In further data analysis work, it was established that the obtained frequency shift follows the in-reactor neutron flux density although the electron densities within the cavity were considerably less than anticipated. It has not been established that the shift is entirely due to neutron flux density changes and the observed differences may well be due in part to both temperature and gamma effects. The planned tests with He-3 and He-4 should provide data relating to effects from both neutrons and gammas, and the cavities to be used were designed in a manner to assure correct mode of operation.

MICROWAVE DETECTION OF REACTOR COOLANT IMPURITIES AND MEASUREMENT OF IN-CORE TEMPERATURE PROGRAM

Work began on a new program recently authorized to investigate advanced microwave techniques of monitoring reactor data. Initial two-fold objectives are to establish feasibility of detecting the presence of light water contaminants in heavy water coolant systems, and detection of moisture in gas coolants. Both investigations consider the effect of impurities on the propagation characteristics of microwaves through the liquid or gas coolants. Later phases of the program will investigate the possibility of measuring high gas temperatures by two techniques: One similar in principle to that used in detecting impurities, the other based upon the emission of microwave energy by hot objects.

Preliminary efforts have been made to assemble needed experimental apparatus. A suitable air supply with varying humidity has been obtained and a microwave cavity was designed to provide measurement of the dielectric constant as a function of moisture content. Studies indicate that detection can be obtained for air moisture changes of 10% or less; plans for obtaining greater sensitivity are being developed.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

The recently developed tubing cross section display unit has been successfully demonstrated on a section of stainless steel tubing having a 2-3/4 inch o.d. and 1/8 inch wall. Artificial flaws consisting of 1/8 inch diameter holes with depths from 0.003 to 0.062 inches were fabricated on the inside and outside surface of the tube. These flaws were detected and displayed on the tubing cross section display to show both the location and relative flaw size.

This test makes use of the multiparameter test capabilities to provide an independent readout flaw location while discarding unwanted probe wobble signals. For this demonstration, the equipment was operated at both 250 kc and 70 kc to give four parameter capabilities.

Two nomographs which will be useful in designing eddy current tests have been drawn. One nomograph solves the skin depth equation, and the other is for use in testing bar stock with encircling coils. This latter chart is so constructed that the approximate test coil impedance normalized to the impedance of the test coil in air, can be obtained from the chart for a wide range of bar and coil diameter. These charts will be used here at Hanford to determine test specifications and will be presented in the Nondestructive Testing Handbook for use throughout the industry.

Fundamental Ultrasonic Studies

Previous analysis of the wave propagation model did not provide a simple expression for the beam spread correction factor which is needed in taking attenuation and diffraction data. To facilitate calculation of this correction factor, a computer program has been written to calculate the incident angle for a given material thickness and ultrasound velocity which will yield the same beam spread as the standard. Using ultrasound incident at these computed angles, the pulse shape through known path lengths of water, fused quartz and stainless steel were obtained. These are being used to compare with analogous pulse shape changes predicted by the attenuation wave propagation model currently being developed.

A study is also in progress to determine the magnitude of beam spread errors in earlier measurements of shear wave attenuation coefficients in stainless steel made using narrow band pulses.

Fatigue Detection Studies

The feasibility of detecting fatigue by critical angle ultrasonic and ultra-stable, high frequency eddy current methods looks encouraging on the basis of recent experimental data. Ultrasonic boundary wave signal changes were observed as early as 20% of fatigue life in aluminum sample and at 40% of fatigue life in the copper sample. These signals varied more than a factor of five from initial fatigue cycling to failure. Less definitive results were observed on the stainless steel samples with the ultrasonic test. This in part may be due to the fact that the sample loading was incorrectly chosen and the fatigue failure point was never reached.

A significant observation from the ultrasonic test was that it was sensitive to fatigue orientation. This property appears to make possible the development of a test to determine the absolute degree of fatigue. This may be done by comparing reflected boundary wave signals from energy propagated in perpendicular directions at a given point.

The ultra-stable eddy current test equipment was more sensitive to fatigue in the stainless steel samples than either the aluminum or copper samples. Eddy current signal changes were observed after 600,000 cycles of fatigue in a stainless sample which withstood more than 10 million cycles. Calibration showed that a decrease in conductivity of 0.16 micro-ohm centimeter or a conductivity change of about 0.25% resulted from the 600,000 cycles of fatigue. Although small, this change was readily detected by the specially designed eddy current equipment, and it is believed that the ultimate sensitivity of the equipment may be a factor of two better.

It has not been fully determined whether the signal changes observed in the stainless steel samples are due only to electrical conductivity changes. Efforts to establish the cause of these signal changes are under way and the results may influence future equipment improvements.

Heat Transfer Testing

Experimental equipment is being designed to evaluate the capabilities of the sinusoidal steady state thermal test. Simplified experimental equipment for initial evaluation of the new testing concepts is being fabricated, including a sinusoidal thermal wave drive which will operate at frequencies from 0 to 2 cycles per minute and a 3-inch square sinusoidal thermal wave transducer.

Once confirmed the new concepts are expected to serve as a guide in the development of practical thermal test equipment. Further evaluations were made of the general analytic solution for the surface impedance of a plane layer of finite thickness in resistive contact with a semi-infinite second layer. These showed that the thermal surface impedance followed a curved locus as contact resistance was varied; thus, compensation for variations in contact resistance can be accomplished by a center of curvature method.

Ultrasonic Imaging Research

As a follow-up on the theoretical treatment of the phenomena of liquid surface distortion by ultrasonic energy, an attempt is being made to measure the amplitude of these surface displacements. The method employed consists of a Fizeau interferometer in which the liquid forms one of the reflecting surfaces. The other surface of the interferometer is an optical flat which is positioned above the liquid surface. The flat is adjusted to show a single interference fringe with no surface incident ultrasound. With the ultrasound on, the fringe pattern distorts and the amplitude of the displacement can be determined by counting the fringes. The experiment is being conducted at the Rattlesnake Mountain site in order to obtain a relatively vibration-free environment.

Quantum Physics Devices

Color centers have been produced in Li_2SO_4-H_2O by irradiation with 2 MeV electrons. Initial measurements indicate radiation induced optical absorption bands at 3900 and 5850 Å. Efforts will continue to correlate these absorption bands with the particular type of defect center producing them. Pure and copper doped single crystals of this material are being grown from aqueous solution for these studies.

Quartz delay lines and holders for evaporative layer transducers have been constructed. The delay line is a quarter inch diameter quartz rod one inch long having optically flat and parallel faces. The quartz is vacuum plated with stainless steel, silver and then the piezoelectric layer. Cadmium sulfide is being used as the piezoelectric material for these initial experiments. Tests are being conducted to determine the proper evaporating temperature and substrate temperature.

A diffusion furnace to be used for semiconductor and hypersonic device research has been received from Hevi-duty Electric Company. This is a tube type furnace having an upper temperature limit of 1300°C, controlled to ± 1°C.

Ultrasonic Transducer Development

Circuitry and high temperature mechanical specimen holders are being developed for use with the electrostatic transducer for inspection of materials at high temperature. A high repetition rate pulser is being developed to permit the use of the pulse super-position method for determining the velocity of the ultrasound in the specimens. A repetition rate of 65 kilocycles will permit velocity measurements to be made on most metals while utilizing the electrostatic transducer. Variation in repetition rate is less than 0.1 percent which will permit accurate velocity measurements to be made.

A mechanical holder is being developed to permit measurements to be made up to 700°C. A second holder is also being developed which will withstand temperatures up to 1300°C. The 700°C holder will hold the test specimen in a vertical position and utilizes the weight of an electrode and the test specimen to provide the necessary pressure on the dielectrics. Since the test specimen and electrodes are not rigidly held axially, it will be possible to obtain constant pressure and alignment even at the higher temperatures.

Nondestructive Testing of Isotope Heat Sources

Development of nondestructive tests for the isotope heat source cells continued. Design of the mechanical portion was completed, and the prints sent to the shops for fabrication. This unit will be capable of providing the necessary scans to inspect both spheres and cylinders. Only those brackets necessary to hold the wall thickness measuring transducers have been incorporated; however, there is sufficient space to add more transducers and brackets as additional tests are developed.

In the prototype cells inspected to date, the central portion, or highly worked area of the cylinders, along with the end welds, appear to be the weakest points. Considerable difficulty was encountered when ultrasonically measuring the wall thickness where the metal is highly worked. However, by highly damping a low frequency transducer, this inspection can be performed. If information describing degree of metal working is required, it could be obtained by measuring the amplitude of the wall thickness test performed over the weld area to detect cracks or voids oriented perpendicular to the cylinder axis. A second test, utilizing 45° shear waves, would detect cracks or voids oriented at 45° to the cylinder axis. A third test, using a transducer to inspect the weld area from the side of the cylinder, would detect cracks or voids oriented parallel with the cylinder axis.

Several simulated isotope heat sources were produced by slumping glass cylinders into 1/16" thick 310 stainless steel cans. Gaps were produced at the core to cladding interface by inserting a 1/16 inch thick iron shim prior to slumping, and then etching with nitric acid to remove it afterwards. These gaps are approximately 1/2 inch and 1 inch wide, and have about the same thermal resistance as a thin void crack of that width. Although one crack would not be too detrimental, a number of such cracks encircling the core near the outer edge would cause a molten core. This would reduce impact resistance and increase chances of widespread contamination upon impact.

A simple transient thermal test has been developed for initial tests on the simulated heat sources. The test was applied by heating the cylinders to 450°C, then externally cooling them with a cold air blast during rotation of the cylinder in a lathe. Initial test results indicated that the 1/2 inch wide defect could be detected, but more recent tests revealed possible doubts due to thermal gradients in the furnace in which the cylinders were heated. The tests will be repeated using a heavy aluminum container now being fabricated to act as an isothermal enclosure for the simulated sources during heating.

Equipment for making sinusoidal steady state tests on various cladding and core material combinations in the isotope heat source program is nearly complete. This equipment will be used on simple blocks of material to test the theory thus far developed. Once confirmed, the theory will be useful as a guide for development of equipment for application to less idealized samples.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Substantial progress was made this month in the development of the system for detecting a zinc sulfide tracer with airborne samplers. The detection and recording problems that were experienced in earlier tests were remedied. All components within the aircraft performed satisfactorily in conditions that will be met in a sustained field program. A tentative calibration curve for the Real Time sampler was derived from paired particle counts of the airborne RTS and bulk samplers. This curve agreed well with results obtained from paired counts of ground-stationed units. The calibration range of concentration extends only over one order of magnitude. Extension of the curve to provide the calibration required for large scale experiments is planned early next year when the weather becomes favorable.

Analog computer analysis of turbulence data collected last summer jointly with the University of Washington progressed considerably. Using filtering techniques, power spectral estimates for all three components of the wind at frequencies of 0.2 cps and higher were obtained. In addition, the variance of the wind components and cross-correlations were derived. These direct measurements of atmospheric turbulence are essential for a thorough statistical description of all diffusion processes that occur in the air. The data provided by this analysis will provide insight into the nature of these fundamental mechanisms.

The persistent winter snows that fell during the month provided the opportunity to determine the scavenging efficiency for snow. Two tests were successfully completed in which snow samples had fallen through stack off-gases were obtained. After assay of these samples, expected to be completed early in January, washout coefficients for snow can be calculated. At the present time, there are no known experimental or theoretical studies pertaining to the scavenging properties of snow.

Radiological Physics

Another field trip to Anaktuvuk Pass was completed. This latest trip completes a one year study of the seasonal variation of Cs^{137} body burdens of the residents in this Alaskan village. Results of the counting show that the average body burdens are still decreasing since the July peak. The average adult body burden was 710 nCi compared to 1280 and 1030 nCi obtained in July and September, respectively. The maximum burden measured was 1660 nCi. The body burdens should continue to decrease until about January and then start increasing due to the availability of caribou that have been feeding on lichens.

A patient of Dr. Osgood at the University of Oregon Medical School was counted with the P-32 counter after being injected with 1.2 mCi of P-32. The patient permitted us to calibrate our counter with him; the data agreed with our previous calibration.

The plutonium counter was again used to assist the biology laboratory with the plutonium inhalation experiments they are conducting with their dogs. Three of the dogs that were counted were exposed to plutonium from an experimental explosion about a year ago. These dogs did not contain any detectable plutonium, and it is now possible to release them for another program.

At the request of the AEC, a review of the X-ray dosimetry being used in a spermatogenesis study was completed. No gross errors were uncovered in the review conducted at a Pacific Northwest Research Foundation laboratory.

The positive ion accelerator was shut down twice this month for extensive repair work on the magnet power supply and the main vacuum system. At the present time, most experimental work in the 3745-B Building has been interrupted while repairs are made on the building and equipment damaged by flooding December 23.

Dr. L. W. Seagondollar visited Hanford for ten days to continue development work on the charged particle separator for the Van de Graaff. A technique was developed which will allow a more positive identification of the ions which have passed through the ion separator. An Einzel lens is being developed which will improve the focusing characteristics of the system.

An experiment was designed to discriminate optically between gamma and neutron induced pulses in organic scintillators that are used in neutron dosimetry. Necessary circuits and techniques have been developed and tested. Preliminary data are encouraging; however, a more detailed study is being undertaken.

Experimental measurements were made to determine the scattered neutron flux detected by a precision long counter (PLC) under different scattering conditions. The results of these measurements indicate the PLC is seriously limited when used to measure a neutron flux where unknown scattering conditions exist.

The temperature control box for the small calorimeter galvanometer amplifier was completed and operated with a dummy thermopile. The amplifier still has an unexplained drift when it is used to measure the temperature difference in the calorimeter. Drifts corresponding to room temperature changes are no longer present.

Instrumentation and System Studies

An improved respiration cycle sensor, used in canine masks, was placed in service at the Biology Toxicology Laboratory for use in lung tidal volume studies. Several additional pressure transducers were also prepared and a diaphragm-operated pressure relay was assembled for installation in the mask.

Development progressed on the signal conditioning circuitry to be used with the blood pressure measuring portion of the animal physiological function and radionuclide uptake telemtry system. Both differential mode and single-ended solid state amplifiers were designed and tested for use with the system. A circuit was developed to demodulate the amplifier output and to drive the incorporated voltage-controlled oscillator. Initial testing proved acceptable. A device to simulate blood pressure is now being assembled to provide synthetic testing of the blood pressure channel. Bandpass filters are being procured for use in the radiotelemetry transmitter.

Circuit development work progressed on the algae counter being developed to monitor radionuclide uptake in algae cultures at the Biology Laboratory. The planned monitor will provide continuous measurement of the uptake as opposed to current methods which require removal of the algae from glass slides before counting is conducted. A miniature GM tube will be used as the detector with the algae grown on a tubular slide in which the detector is inserted. Tests indicated that satisfactory algae growth is obtained on styrofoam; thus, it will be used as the slide material. Since the algae incubator was destroyed in the Biology fire, experimental work will be suspended until a replacement incubator is obtained.

A new technique was devised, and is being tested, using a diaphragm switch to alleviate difficulties from damage and tarring to the thermistor transducers used in the Biology automatic smoke inhalation control instrumentation. In addition, an improved cigarette holder was designed to incorporate a capillary tube to contain the thermistor; this should prevent smoke passage directly across the transducer.

The solid state, logarithmic response, instrument used to monitor dose rates to fish which was severely damaged during the recent Biology fire has been repaired and returned to useful operation.

Further developmental progress was achieved on the miniature wound probe, which will be used by Hanford medical personnel in detecting Pu-239 contained in human wounds. The scintillation probe detects the 17 keV X-rays from Pu-239. A solid state preamplifier was designed and tested with the stainless steel clad probe and calibration tests were initiated. It appears that 5 nanograms of Pu-239 (approximately 0.3 nanocuries) can be easily detected.

The prototype solid state electronic circuitry, for the gamma dose rate measuring portion of the combination neutron-gamma (mixed) dose rate instrument, was completed and is being tested. This portion includes a preamplifier, main amplifier, discriminator, integrator, and readout. Development work was initiated on the circuitry for the neutron dose rate measuring portion of the instrument and further tests at the TTR are planned for the neutron detector. In addition, it may be necessary to use the Van de Graaff again for tests with the lithium-foil covered surface barrier detector.

Satisfactory operation was obtained with the experimental solid state circuits for use in counting single electrons from multiplier phototubes. This work is applicable to dosimetry studies. The two incorporated discriminators provide a pulse width (at half-maximum) of about one nanosecond and satisfactory isolation was obtained between the discriminators. A pulse-pair resolution of better than 10 nanoseconds is being consistently achieved and comparisons of direct and single electron pulse height spectra demonstrate good agreement.

Progress was achieved on the development, using field effect transistors, of a low noise, charge sensitive, preamplifier. Open loop gains of about 3300 and 7000 were obtained using the FET-FET and FET-transistor design approaches, respectively. Noise measurements at dry ice temperature indicated a noise level of about 1000 electrons (RMS). The low frequency instability previously noted was corrected through the use of bias stabilization feedback.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples at a reduced rate because of the lack of experienced Technologist-Spectrometer operators. The mass spectrometer operated satisfactorily during the month. The new vacuum-lock sample changer for the spectrometer has satisfactorily operated on the test bench. The sample changer is ready to be installed on the spectrometer when the required high-voltage power supply is received. Detailed design of the vacuum-lock sample changer and source components for the new mass spectrometer is about 50-percent complete.

EXPERIMENTAL REACTOR PHYSICS FACILITIES

PRCF

The experiments on the Pu-Al H_2O moderated core were completed on December 18. The facility is presently in a 4-5 week outage to allow conversion back to D_2O capabilities.

U^{235} and Pu^{239} fission chambers were traversed vertically in the center lattice position. These are the first successful fission chamber data obtained for the H_2O cores. Reactivity measurements were made on simulated 7-cell voids in the center and at various radial positions in the core. Measurements were made on various densities of styrofoam simulated voids of about 85% and 35% which permitted an extrapolation to determine 100% void worth. Measurements were also made on a simulated void of similar volume made of teflon. Measurements indicated teflon is a good material for use in void measurements producing an effective moderator void of about 70%. Reactivity measurements were also made of a 1/2 Al rod in a void and an empty EBWR fuel element in the center lattice hole.

Moderator temperature worth measurements were made over a range of 19° to 35°C. The temperature coefficient was positive over the range of measurements going from + 0.043 mk/°C at 20.1°C to + 0.024 mk/°C at 32.5°C.

Relative reactivity worth measurements were made on 44 inch and 36 inch Pu-Al fuel pieces using Pu content and percentage of Pu^{240} as parameters. Cadmium covered Pd and Au were irradiated in the center of the core as a part of the neutron spectra studies.

A three zone load to critical with the higher percentage Pu^{240} fuel elements in the center zone was completed. This loading was the opposite of the previous three zone loading, i.e., the lower percentage Pu^{240} fuel in the center zone. The critical loading was found to be within 1/2 of a fuel element of the previous three zone loading. One point moderator worth and rod calibration measurements were also conducted. Two traverses were also made; a copper pin traverse for previous core comparisons and copper pin traverses in the environment of a fuel piece, a void, and a column of water.

A flux trap attempt in the center of the core was terminated due to the lack of fuel elements.

Construction continued on Project CAH-119 throughout the month.

PCTR Operation

Operation of the PCTR continued routinely during the month. The NPR coproduct experiment-dry was completed during the month. The experiment to measure uniformity of mixed PuO_2-UO_2 fuel rods was started during the month.

TTR Operation

The TTR was operated one night for the University of Washington Graduate Center.

Critical Approach Facility

The critical approach facility was not operated.

COMPUTER FACILITIES

Design of the digital multiplexer for the PDP-5 Digital Computer-Mass Spectrometer Control Unit has been completed and a letter was sent to the customer listing the estimated cost of hardware required to build the unit.

The design of a relay multiplexer system has been started for the containment systems evaluation program. A series of experiments are planned to evaluate reactor rupture problems and containment systems. Included in the instrumentation for this project will be a small general purpose digital computer. One of the functions of the computer will be to provide high speed data recording for over 150 instrumentation points. The relay multiplexer will allow the computer system to sample these points during the experiments.

Analog computer utilization was as follows:

EASE 1132	EASE 2133	
157	144	Hours Up Time
0	16	Hours Scheduled Down Time
3	0	Hours Unscheduled Down Time
160	160	Hours Total

Problems considered during the month were:

1. Meteorology Study.
2. N Reactor Pressurizer.
3. N Reactor Over-all Simulation.
4. N Reactor Secondary (3-Surge Tank Model).
5. Pot Calciner Study.

Disassembly of the analog computers was started December 21 for the move to the 3201 Building in North Richland on December 28. It is expected that the computer will be down two to four weeks.

The fall quarter analog computer class was concluded. Thirteen out of the sixteen starting students completed the ten-week, twenty-hour course. Thirteen engineers and scientists attended the first two sessions of the winter quarter class.

CUSTOMER WORK

Weather Forecasting and Meteorological Services

A technical review was given the meteorological presentation of a proposed manual on Environmental Monitoring and Emergency Situations, sponsored by the International Atomic Energy Agency, Vienna.

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	93	78.0
24-Hour General	62	79.4
Special	163	87.1

December was much colder and wetter than normal. Snowfall amounted to 19.1 inches, which was a new all-time record high for the month.

Mass Spectrometry

Isotopic analyses were provided on 4 uranium samples in support of HAPO U^{233}-production studies, and on one plutonium and three uranium samples in support of Reactor Lattice Physics experimental programs.

Analyses were also provided on three uranium samples in support of Metallurgy Development Operation experimental programs (U^{235} content of thorium alloy thermocouple, U^{235} content of scrap U for safety reasons).

Satisfactory operation of the vacuum-lock of the heavy element mass spectrometer has been restored.

Instrumentation and Systems Studies

Development continued on an ultrasonic test for measuring and permanently recording wall thickness information during the nondestructive testing of solidified waste storage pots. In early discussions, the possibility of incorporating this test into the welding station was considered. However, the pot rotation requirements for welding and ultrasonic testing are sufficiently different to impose difficult design problems. The chuck which holds the pot during the welding operation also "hides" a significant portion of the pot making this region impossible to test. Consequently, a separate mechanical system is being designed.

The mechanical system will rotate the pots at 50 rpm in their vertical positions. An ultrasonic transducer will then be translated through the pot length to obtain 100% inspection of the pot's cylindrical surface. This mechanical unit will be 18 inches square by approximately 12 feet long. By inserting it into a water tank in the hot cell, full pots can be inspected, or by inserting it into a tank in the laboratory, empty tanks can be inspected.

Design and fabrication of the electronic portion of the thickness tester was also initiated. The unit is approximately 25% complete and design criteria for the remaining portion have been outlined. With a broadband amplifier (1 to 10 megacycle), both wall thickness and laminar types of defects should be detectable. Gating and logic circuits, which will present the ultrasonic information to quantizing circuits, are being developed. The quantizing circuits will then present the ultrasonic information in step shades of grey on an X-Y type recording.

A study was conducted to determine the feasibility of measuring the spacing between the aluminum can and the Pu-Al core of a development fuel. A dummy fuel was built with a brass core and paper shims to simulate various gap spacings. An eddy current tester was used to evaluate the effects of core-to-can spacing, probe-to-can spacing and wall thickness variations in the can. The results of this investigation indicate that core-to-can spacings greater than .002 inch can be measured to an accuracy of approximately ± .001 inch with present techniques.

Ultrasonic measurement of a thin Pu coating on a dissimilar metallic substrate theoretically appears to be possible using a resonance technique provided there is a reasonably large impedance mismatch between the plutonium and the substrate. The test is based on a technique to isolate and measure that portion of the through-transmission energy which is frequency dependent solely on the Pu thickness. Samples are being procured to evaluate this technique. Attempts are being made to ascertain the ultrasonic properties of the proposed test materials. It appears that the acoustic impedances of Pu and Be are too close to utilize a resonance technique on this combination. There is, however, apparently a fair mismatch between Pu and steel and between Pu and U. Requests have been made to prepare lab test samples of Cu on Be and a Sn-Pb alloy on Cu--both of which will be analogous to possible actual combinations.

For engineering development to Waste Solidification Engineering, HL, a 7-ft. long liquid level detector probe was completed and installed in a waste solidification tank in 321 Building. Using a transmission pulse with a 1 nsec ristime, three insulators (used for calibration) were easily visible on the oscilloscope readout, equating to 20, 40, and 60 inches in distance. Boiling water was used in the tests and the levels achieved were recorded and checked against a purge-type level indicator. In addition, a 22-in. long probe was assembled for use in a glass melter and tests were satisfactory. Further work planned includes development of a circuit to convert oscilloscope presentation data to a dc output for use with a process recorder.

Engineering assistance was provided to Geochemical and Geophysical Research, HL, in obtaining seismic measurements during two explosions conducted during construction of a nearby highway. Both measurements were successfully recorded and the data will be employed in determining seismic constants for the material underlying Hanford.

Engineering design work was initiated on a new gamma monitor to be employed in moving vehicles by Environmental Monitoring, HL. This instrument is planned to complement the completed aerial gamma monitor now in routine use.

The dual-detector support mechanism for the assault mask monitor, being developed for use at the Hanford Laundry Facility, was completed and is being tested. Both scintillation detectors and all of the associated solid state electronic instrumentation are completed. The project is nearly finished.

Further engineering assistance was rendered to Geochemical and Geophysical Research, HL, concerning instability problems associated with the scintillation well counter and with the Am-241 counting system.

Development continued on a method of remotely measuring the displacement of metal specimens in a high temperature oven. An improved optical arrangement and a ten megacycle electronic counter improved resolution. The system can now detect 0.2 mil displacement. A further improvement in the resolution is expected to be obtained by slowing the light scanning speed down by a factor of three.

The creep data logger at 105-KW is now in operation on an in-reactor creep test. It operates satisfactorily when logging thermocouple data automatically and when micropositioner data are logged manually on demand. Minor logic circuit modifications were made in the logger and debugging of the digital clock circuit continued.

Design was started on a mercury-wetted relay data processor to replace the obsolete one now in service at KW. The processor consists of a relay matrix which stores the data from the micropositioner digitzer and processes it into the data logger.

Development work continued on an ionization transducer to measure the small mechanical displacement of metallurgy test specimens. Tests using a neon filled tube with a 10 Mc signal of 110 volts RMS provide the principle to be feasible. The circuit produced a dc voltage output that was linear with the displacement of the sliding ring. In an effort to improve sensitivity, a new nitrogen filled transducer was fabricated in accordance with specification of work previously completed. Development work is now being done on a 320 volt RMS 20 megacycle exciting source for the new transducer.

Fabrication of the electronic components and transducers for the variable permeance transducer readout system was completed. The system is used in conjunction with the Instron tensile testing apparatus to obtain stress-versus-strain data of metallurgy test specimens. Tests were made to optimize the response of the transducers. The system was installed with the tensile testing apparatus and effort is being applied to make the two systems compatible.

Optics

Modifications to the Pan Bore camera to be used to take a complete picture of the inner surface of N Reactor fuel elements have been completed. The picture is taken in eleven steps per 8 x 10 inch film sheet. The camera has five probes of different sizes: 50 caliber, 60 caliber, 20 mm, 27 mm, and 30 mm. Three of the probes worked fine. Due to modification to the 30 mm probe, the rotational alignment was off by 180° and the advancement indicator was no longer calibrated for the probe. The advancement indicator was recalibrated by taking a picture of a rule laying inside the tube and then calculating the proper distance to advance the probe per step. The rotational alignment was made by taking a number of pictures until a continuous clear picture was achieved. The 20 mm probe was also out of alignment by 180°, and it was aligned in the same manner as the 30 mm probe.

The Optical Shop is in the process of cleaning and repairing the optical equipment damaged in the fire at Biology. The optical equipment consists mainly of microscopes damaged by smoke and water.

A monocular eyepiece was designed for the 105-C basin viewer. The new eyepiece will have a magnification of 10 times and will be used to take pictures of fuel elements.

Two simple quartz lenses, with a focal length of 3", were designed and fabricated.

The installation of the Elgin lapping machine is nearing completion. The three electric furnaces have been moved into the lapping room and will be in operation as soon as the electric power is available.

The following shop work was performed:

1. Repaired a tank farm periscope.
2. Two opton stero microscopes were repaired.
3. The relay lenses in the East and West cranes at Purex were replaced.
4. Three microscopes were cleaned and repaired for Biology.
5. A viewer at 100-H was cleaned and polished.
6. A glass plate was fabricated with hole location tolerance of ± .001" and hole size tolerance of ± .001".
7. A thin film of cadmium sulfide was deposited on the ends of siz quartz rods.

8. An aluminum film was deposited on a cylindrical mirror.
9. A quartz X-ray furnace liner was fabricated.
10. A borescope was repaired for 327 Building.
11. A periscope for PRTR was cleaned and repaired.
12. A sinusoidal heat system was fabricated.
13. A cross hair was installed in a microscope eyepiece.
14. One camera shutter was cleaned and repaired.

Physical Testing

Engineering assistance was provided Plutonium Metallurgy Development for the final acceptance tests on the microprobe analyzer at the vendor's plant. Performance will be rechecked when the installation is completed at Hanford.

An ultrasonic resonance test, using the portable Vidigage, and a mechanical depth indicator, were used to measure the corrosion pitting in the No. 3 Superheater Boiler tubes at KW Reactor. Data were provided for safety and insurability considerations by third-party inspection.

Ultrasonics were used to measure the bonding of the aluminum cladding on lithium-aluminum co-extruded target elements. These elements are designed to be unbonded; bonding is cause for rejection.

Stresscoat was applied on several fuel rods during vibratory compaction to determine the principal stress areas encountere during fabrication. More sensitive methods, photostress or strain gages, will be employed after the patterns have been established.

Nondestructive tests are now applied on the rod and bar stocks prior to specimen fabrication for the Radiation Damage to Reactor Metals Program. Defects, as well as grain size variations, have been observed that would lead to faulty specimens.

A sensitive ultrasonic test has been developed for large, three-inch diameter, Zircaloy-2 sheath tubing. Axial and circumferential defects four mils deep are readily resolved. The four tubes examined were accepted.

Radiography of welds during equipment fabrication continues to increase. A new Annular Dissolver was started by J. A. Jones Company, in North Richland, along with the H-4 Vessel, the Titanium Calcinator, and the Corrosion Loop. Radiography assures compliance with design specifications.

Nondestructive tests and inspection criteria were outlined for the 100,000 psi isostatic press for Chemical Processing Department. Stresses in the vessel wall from these pressures must be monitored at the first application, and periodically thereafter, to determine the areas where fatigue cracking could initiate.

Changes in the Stainless Steel HWS-8000 Series Procurement Specifications have been authored to provide for the nondestructive and radiographic records to be maintained at Hanford.

INSTRUMENT EVALUATION

Instruction and maintenance manuals were partly completed for the accepted scintillation, solid state circuitry, combination alpha-beta-gamma hand and shoe contamination monitors. These 10 units are of Hanford design and were fabricated offsite. In addition, eight more monitors of an advanced design are now undergoing offsite fabrication by another manufacturer.

A modified scintillation area monitor was tested for a 4-hr. period in a 4×10^3 R/hr gamma field. During the first hour, a -16% calibration change occurred with a -5% additional change during the final three hours. The instrument is being modified to provide readout only during the time of engagement of a momentary-contact switch; this method may materially reduce the fatigue-effect problem.

Temperature-effect evaluation tests were conducted on a group of self-indicating dosimeter "pencils" (small ionization chambers) as used in the Hanford Emergency Monitoring kits. For cycling over the range from 75°F to 0°F, a maximum calibration change of ± 5% was noted. In addition, test data were analyzed regarding some 500 self-indicating dosimeters, which have a range of 0-500 mR.

Evaluation tests were initiated on a new prototype Scintran (Hanford design) of an order of 60 units being fabricated offsite. The fabrication quality appears to be marginal.

RS Paul

Manager
PHYSICS AND INSTRUMENTS LABORATORY

RS Paul:mcs

1234884

CHEMICAL LABORATORY

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Overheated N-Reactor Fuel Study

A ninth unirradiated co-product target element heating experiment was completed in preparation for thermal stressing of similar irradiated elements. Initial attempts to carry the element to the goal temperature in a steam atmosphere were thwarted by thermocouple shorting apparently due to the steam atmosphere. Modified thermocouple insulation and reduced steam flow then permitted a cycle of 935 C and a water quench. Restricted water flow due to element swelling at the downstream end and an exothermic release of heat generated steam over a period of about one minute of such pressure that water was forced from the first 50-gallon water seal drum, breaking the seal. The element did not survive this stress and fragments were carried into the trap provided for this eventuality. Further examination of the residue is planned.

Irradiated elements obtained for subsequent tests were found to exceed anticipated gamma radiation levels, and the furnace assembly was modified and shielded to reduce exposure to reasonable levels. Other changes were incorporated in the experimental arrangement based on earlier tests and a hazards review. The heating of the first irradiated element is now planned for month end.

Electroplating of Nickel on Uranium

Current studies of uranium metal surface pretreatment, preparatory to nickel plating, have been concluded. Recent work has shown the need for careful control of H_2SO_4 and HCl concentrations in the H_2SO_4-HCl anodic etch step, the optimum solution composition being 6.0M H_2SO_4 and 0.35M HCl. The EMF-measurement technique (HW-79582) was used to determine minimum effective pickling times for the HNO_3 pickling step which follows the H_2SO_4-HCl anodic etch. Sample results were as follows: for 8.0, 10.0, and 12.5M HNO_3 at 25 C, the minimum times were 80, 40 and 33 seconds, respectively; for 6.0, 8.0, and 10.0M HNO_3 at 40 C, the minimum pickling times were 60, 30 and 23 seconds, respectively. The effect upon nickel plate quality of exceeding the minimum effective pickling time was also studied. No difference in plate porosity was noted for specimens plated under identical conditions, following anodization in 6.0M H_2SO_4-0.35M HCl and pickling in 8M HNO_3 for 40 C for times ranging from 30 seconds to five minutes.

[redacted]

A brief study was made of the use of 40 Kc/sec ultrasonic waves to provide agitation when electroplating nickel onto uranium. No improvement in plate quality was observed.

Ground Disposal of Reactor Effluent Water

Preparations for a pilot reactor effluent disposal test at 100-F Area were completed, and a portion of the reactor effluent was diverted to a disposal trench for 7 days. The average infiltration rate during this time was 1000 gallons per day per square foot of trench. The volume discharged was 1520 times the trench volume.

SEPARATIONS PROCESSES

Disposal to Ground

No significant changes were observed in the gross beta activity levels in the ground water beneath the 200 Areas during this month. Beta emitter concentrations in well 699-30-31 are apparently reflecting the peak discharges of activity from the Purex Plant in 1960-61. About a year and a half ago well 699-34-39A showed a similar increase. These data continue to corroborate the travel time estimates made for ruthenium in December, 1963. No long-lived radionuclides were detected in the ground water beneath active cribs in the 200 Areas during this reporting period.

Plutonium Aerosol Studies

Final shakedown tests and dry runs were completed prior to experiments which will initially involve oxidation of small specimens of plutonium metals and the characterization of the particulate material formed. The capability of the microscope installation in the gloved box was demonstrated. The first plutonium specimen was prepared for oxidation.

Polonium Chemistry

Transport of polonium from bismuth oxide in flowing argon at 1000 C at a flow rate of 55 ml/min decreased rapidly after removal of about 10%. The recovery of polonium as a percentage of the amount originally present was respectively 12.0, 25.3 and 26.5% after 4.28, 27.93 and 51.73 hours. These slow removal rates are not encouraging for the recovery of polonium by a transpiration process; however, many of the important parameters of the process have not received detailed attention at this time.

[redacted]

Electrodiffusion in Bismuth

The possibility of removing polonium from bismuth metal by solid state electrodiffusion is being examined. In preliminary experiments using tellurium as a simulant for polonium, no migration was found after passage of current at 200 amp/cm^2 for 230 hours at 170 C. Negative results were also observed in a second run at 225 C, of similar duration, but at 260 amp/cm^2. On the chance that tellurium may not be an acceptable stand-in, a migration experiment using Po-210 traced bismuth has been undertaken.

Thermobalance Studies

Thermobalance studies of the thermal decomposition of plutonium(IV) oxalate have continued, in search of the mechanism by which carbon is left in the PuO_2 product. It now appears that a major source of carbon is the catalytic decomposition of CO coming from the decomposing oxalate. Supporting evidence for this conclusion comes from the observation that PuO_2 produced by thermal decomposition of $Pu(C_2O_4)_2 \cdot 6H_2O$ in a CO atmosphere was completely covered with a carbon coating.

Thoria Processing

For the various steps in the proposed processing of irradiated thoria in the Purex plant, maximum corrosion of 304-L stainless steel will occur during the boil-down procedure for feed preparation. Tests on 304-L stainless steel indicate corrosion rates of 1.5 and 4.5 mils/mo at nitric acid concentrations of 12.3 and 15M, respectively. Intergranular attack is also increased at the higher nitric acid concentration.

Test specimens of 304-L stainless steel, HAPO-20 and Corronel 230 have been suspended in the 321 Building thoria dissolver. Corrosion rates observed were 1.6, 0.5 and 0.8 mils/mo for the three alloys in the order given.

Chemical destruction of nitric acid is an attractive alternate to thermal concentration. When 1M thorium nitrate-8.5M nitric acid solutions were treated with sugar to decompose the nitric acid, precipitation of thorium (presumably with organic acid anions) occurred. Reduction of the nitric acid to 0.08M via destruction with formaldehyde produced no solids.

WASTE MANAGEMENT AND FISSION PRODUCT RECOVERY

Technetium Chemistry

Extent of volatilization of technetium from boiling ammonium pertechnetate-nitric acid solutions was measured as a function of nitric acid concentration

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to determine the threshold at which excessive technetium loss occurs. This information is important to plant operations since past experience has shown excessive technetium losses from 12 molar nitric acid solutions. Surprisingly, no volatilization (beyond entrainment losses) was observed, even from 15M HNO_3. The difference is believed due to the care taken to avoid superheating the walls of the distillation vessel.

In other studies of technetium chemistry, measurement of the solubility of ammonium pertechnetate and of quaternary ammonium pertechnetates continued. The ammonium common-ion effect on NH_4TcO_4 was measured, and found to be smaller than might have been expected. At 20 C the presence of two molar ammonium nitrate decreased the technetium solubility only from 0.461M to 0.227M. Similar results were obtained at other temperatures.

Semiworks Support

Close support of the Semiworks rare earth program continued during the month. The various steps of the plant process were piloted in the hot cells with actual plant concentrates. Highlights of this effort were (1) finding that rare earth extraction coefficients are much higher into D2EHPA when HEDTA is used to control iron, aluminum and lead than with DTPA, and (2) the discovery of large concentrations of manganese in the feed. Presence of manganese is particularly troublesome since its extraction behavior (as Mn^{++}) is almost identical to Sr^{++} and it reacts with persulfate thereby interfering with the cerium-rare earth partition. On the basis of laboratory experiments, it appears that manganese removal may be effected in the A contact, provided a sufficiently low pH and an efficient scrub section are employed. Accurate pH control is essential.

Because of the difficulties which have been encountered in separating cerium at Semiworks, the necessity of removing cerium prior to ion-exchange purification has been re-examined. It has been concluded that a well-aged promethium concentrate (e.g., approximately one promethium half-life) probably will not require cerium separation to ensure adequate resin performance.

Techniques for removing cerium by a modified ion-exchange process are also being examined. One potential process, which appears promising on the basis of initial laboratory experiments, takes advantage of the fact that promethium forms a significantly tighter complex with DTPA than does cerium. (The effect is ten-fold greater with DTPA than with EDTA or HEDTA). Other approaches being scouted include formation of anionic complexes and the use of inorganic absorbers.

Strontium Semiworks Solvent Studies

Further studies of the marked reduction in Ce(IV) extraction capacity which occurs when used or degraded D2EHPA-TBP-diluent (NPH or Soltrol-170) solvents are washed with sodium hydroxide were continued. Solvents which were degraded by irradiation, washed with caustic, washed with nitric acid and allowed to stand a week at room temperature regained Ce(IV) extraction capacity to near that of the unwashed solvent. A solvent prepared from chemically degraded Soltrol-170 did not recover Ce(IV) extraction capacity during a similar standing period after washing with sodium hydroxide and nitric acid. Cerium(IV) extraction capacity of degraded solvents, whether or not washed with sodium hydroxide, was significantly improved by treatment (one hour at 25C) with an equal volume of 2 to 6$\underline{M}$ HNO_3-0.25$\underline{M}$ $KMnO_4$.

Shipment of Fission Product Cesium Adsorbed on Zeolites

Synthetic zeolites are being considered for use in the shielded transfer casks which carry cesium from Hanford to ORNL. Increased cesium loading in the casks is desired to reduce shipping costs and relieve problems with shipping schedules. Based on laboratory data, cesium loadings may be increased by a factor of three, from 50 kilocuries to 150 kilocuries, by using Linde AW-500 in place of the Decalso presently used. A test is being conducted by Chemical Processing Department to evaluate the physical stability of AW-500 during rail transit.

PROCESS CONTROL DEVELOPMENT

Electrorefiner Control System

The capability of the previously developed electrorefiner control system was extended by adding two new functions: time duration, and an automatic back voltage check with an overlimit trip. The earlier control modes included voltage or current control plus over- or under-limit trips on voltage or current. Additional flexibility is provided by the new functions, including the ability to perform round-the-clock experiments without continuous surveillance.

Temperature Compensation of pH Measurements

An investigation was made of the error introduced into pH measurement by temperature variations using the existing type of thermo-compensators in HAPO in-line pH flow cells. Laboratory measurements indicated that an increase from room temperature to 75 C gave a pH error of about 0.5 pH unit. A method of minimizing the error was devised, in which a tip-sensitive thermo-compensator is substituted for the standard device.

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REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Processing PRTR Fuels

The flowsheet recommended for the dissolution of PRTR UO_2-PuO_2 fuels in the Redox Plant was successfully used to dissolve short pieces of fuels irradiated to 2000 and 5000 Mwd/ton. Fluoride present in the core solution due to conversion of UO_2 to UF_4 during decladding was low and in excellent agreement with predictions based on earlier runs. These results give added confidence that the fluoride in the core solution can be kept sufficiently low for reasonable corrosion rates in the Redox Plant stainless steel dissolvers.

Irradiated PuO_2-UO_2 fuel pieces (2000 and 5000 Mwd/ton) were declad in 5.7<u>M</u> NH_4F-0.055<u>M</u> NH_4NO_3 solution to 0.6<u>M</u> dissolved zirconium. These conditions are recommended for use in a proposed Purex Plant service dissolver (RL-SEP-32). The UF_4 formed during decladding was sufficient to make the core solution 0.15<u>M</u> in fluoride at 1.4<u>M</u> uranium. This is a concentration suitable for dissolution of PuO_2 in the core.

Salt Cycle Process Studies

In support of the C-Cell test of the Salt Cycle concept, the effect of graphite upon the electro-codeposition of PuO_2 and UO_2 was studied. Conditions which gave a plutonium enrichment factor of 1.33 across the electrodeposition using pyrolytic graphite anodes, gave a plutonium enrichment factor of only 0.03 when AUG mold graphite anodes were used and powdered AUG graphite was put into the melt. It is quite apparent that graphite interferes with electro-deposition of PuO_2, probably through its effect as a reducing agent.

Salt Cycle Process Development

In a continuing program for preparation of mixed oxide containing 2 weight percent PuO_2 for recycle to the PRTR, approximately 12 pounds of specification oxide have been electrodeposited at 550 C with a 50 percent oxygen-50 percent chlorine gas sparge. Prior scouting electrolysis produced 7 and 1.2 weight percent PuO_2 mixed oxide with X-ray diffraction studies showing no evidence of a separate PuO_2 phase in the deposit.

RADIOACTIVE RESIDUE PROCESSING DEVELOPMENT

Preparation of Glasses

Thirteen samples of waste calcine were prepared simulating possible spray calciner products from the systems 1WW-B_2O_3, 1WW-CaO-B_2O_3, 1WW-B_2O_3-P_2O_5, 1WW-P_2O_5 and 1WW-PbO-P_2O_5. The calcines were heated to 950 C and held 16 hours at that temperature in a thermobalance. Calculations based on weight change indicate that almost 100 percent of the sulfate was lost from all of the calcines at temperatures below 950 C.

Fusion points and physical characteristics of melts containing 25 - 70 w/o 1WW calcine and varying proportions of B_2O_3 (added as boric anhydride) and P_2O_5 (added as 85% H_3PO_4) were determined. Lowest fusion point observed was 870 C for a mixture containing initially 40 1WW calcine-20 B_2O_5-40 P_2O_5 (expressed as weight percent). A mixture with the initial composition 70 1WW calcine-15 B_2O_3-15 P_2O_5 yielded an apparently homogenous melt which fused at 980 C. This melt contained a higher fraction of 1WW calcine than has been incorporated in any melt prepared previously in any of the systems studied.

Spray Calciner-Melter

A new filter chamber and a calciner-to-melter transition section were installed on the 14-inch calciner to gain performance data on the shorter more compact unit proposed for the Solidification Prototypes. A resistance heated draft tube was also installed and tested.

Feed for three runs was simulated 80 gal/ton U 1WW waste containing 0.75 mole of H_3PO_4 per liter of waste. A fourth run was made with simulated 80 gal/ton U 1WW with 85 grams of borax per liter of waste. Feed rates were maintained at 5 gal/hour. Run durations ranged from four to sixteen hours. Calciner furnace and heated draft tube temperatures were set at 700 C for all runs.

During the first three runs, filter performance was satisfactory when operated with a blowback cycle time of five minutes but a cycle time of ten minutes produced excessive pressure drop across the filters. In the fourth run using a five-minute pulse cycle with borax feed, filter pressure drop was excessive with the new filter section.

The heated draft tube showed negligible powder deposition on the tube except for chunks of powder which built up around thermocouples extending into the center of the draft tube. Calciner wall deposition occurred only at the bottom, below the lower end of the draft tube.

Satisfactory operation of the platinum melter was achieved, except for the overflow and freeze valves, at an induction heated susceptor temperature of 1050 to 1125 C and a melt temperature of 1000 to 1050 C. Extensive premature freezing occurred in the insulated housing section between the melter and the receiver. More heat and a larger housing section than the present two-inch is required for free passage of the glass to the receiver.

Continuous Phosphate Glass Experiment

The Hot Cell Glass Experiment - a cooperative test of the Brookhaven continuous glass process using full level Purex waste - was terminated near the end of the fifth hot run when catastrophic failure of the pot caused glass to flow into the furnace windings, destroying the furnace. Fortunately, all or most of the initial program objectives had been accomplished and only one further run had been planned. Mechanism of the failure is believed to have been an undetected hair-line crack in the platinum liner of the Inconel melter. Contact of molten phosphate glass with Inconel (or any other reactive metal) produces elemental phosphorus, which attacks platinum in a vigorous, thermit-like reaction. The entire bottom of the melter was "burned" off.

No analytical data is yet available for run No. 5; however, data is now complete for run No. 4. In this run the evaporator was operated at a lower temperature (<145 C) to prevent plugging, and the resultant feed to the melter was somewhat "wet", i.e., contained more than the usual amounts of water and nitric acid. Result was the volatilization of a significant fraction of the ruthenium (5.6% vice 0.4%) and considerable splattering and entrainment. Entrainment carried off other fission products and salts equivalent to 0.2% of those in the feed. As a result of the low evaporator temperature, the sulfuric acid condensate from the melter contained enough nitric acid to result in a very corrosive solution.

A formal report is being prepared describing the results of the continuous glass experiments.

Phosphate Glass Studies

A large number of additional measurements of melting point, drip temperature, solubility (or leaching rate) and tendency to devitrify were made during the month to complete data required for an interim formal report on the system: phosphate-(simulated) fission product-process oxide. Results are summarized in the October-December Waste Fixation Quarterly.

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Waste Solidification Instrumentation

A remote viscosimeter was installed in a test unit for evaluation as an in-cell instrument. Viscosity measurements in the 10-1000 centipoise range compared closely with actual viscosities of glycerine solutions. Additional measurement of molten glass viscosities were made, using the falling sphere method, in support of glass melter flow studies. A gas purge system for glass melter level measurement was calibrated in the laboratory prior to use with molten glass.

Waste Solidification Engineering Prototypes

Design verification tests were continued with an 8-inch diameter pot calciner by using simulated Purex high sulfate feed with $Ca(NO_3)_2$ added directly to the feed to prevent excessive sulfate volatilization. In 19 hours, 530 liters of feed were added to the pot for an average rate of 28 liters per hour when using an overall net furnace power of 36 kw. Eighty-three kilograms of calcine were produced. Total furnace operating time was 32 hours. Significant findings were: (1) induction-heated furnace operation was quite satisfactory, (2) performance agreed with earlier ORNL results, and (3) the pot fill volume was 15 percent greater with this run than with a previous run using a draft tube to net the same weight of calcine.

In other water boilup tests, a 12-inch diameter pot produced a 90-liter per hour boilup rate at 110 KW furnace power. Gross entrainment was encountered at this rate.

Waste Solidification On-Line Data Processing

The computer programs required for communication between the GE-412 computer and the process instrumentation were completed and successfully demonstrated. On-line, real-time processing of run data was accomplished during a recent pot calcination run using simulated Purex feed. Continuous material balances, integrated kw-hr input to the induction heaters, and indication of out-of-limit control points were included in the information generated by the computer. Based on experience with this run, minor program revisions and further debugging work were undertaken.

CONTAINMENT SYSTEMS EXPERIMENT

Major Facilities and Equipment

The contract for the reactor simulator vessel was approved by all parties. Delivery will not be until about January 1, 1966. Erection of the containment vessel is progressing on schedule. The design of the vessel is better than 95 percent complete with only the simulator support details still requiring work.

Installation of the demineralizers, deoxygenator, and other auxiliary equipment was started late in the month.

Instrumentation

A study was completed on the sensitivity requirements for the CSE instrumentation for leak test measurements. The standard deviations of ± 0.05 psi for pressure, ± 0.05 psi for humidity, and ± 0.1 F for temperature were found to be reasonable instrument sensitivity requirements for tests of 24 hours duration. These sensitivity requirements can be relaxed, if necessary, by extending the duration of the tests to longer periods.

Specifications for the digital data logging equipment and the justification for this equipment were completed. Equipment delivery is expected about the middle of May.

Two alternative methods for determining the average temperature in the containment vessel were studied. In one, a single nickel wire is strung throughout the interior of the vessel; the wire resistance is a function of the average ambient temperature around the wire. In the second method, readings of individual temperature transducers located throughout the vessel volume are averaged. The second method, using precision platinum resistance temperature detectors (RTD's), was adjudged to be superior for the proposed application on the basis of improved accuracy for about the same cost.

In support of this investigation, a method of welding platinum RTD leads to copper wire was developed using a capacitance discharge welder.

In other CSE studies, specifications were developed for several types of strain gauges to be evaluated under simulated CSE environmental conditions. Consideration was given to acquiring additional information on dry well and containment vessel stresses by appropriate positioning of strain gauges on these vessels.

Program Planning

A detailed outline of leak rate tests on the bare containment vessel following its acceptance from the fabricator is nearly complete. The outline will serve to coordinate plans and activities of all concerned groups by defining specific test objectives, establishing nominal test conditions and parameters, and specifying the necessary facility preparations and equipment.

Mathematical Models

An analog program for pressure-temperature transients in a single containment system consisting of a reactor vessel discharging directly to the containment shell was completed. A digital program for the same situation was also written and is being debugged because the analog facilities are temporarily not available. The situation being simulated includes heat removal by condensation, spray suppression, and "external" cooling loop, and decay heat addition. The last two are included in a simplified form; the cooling loop is simply a fixed rate of heat removal after delay time and the decay heat addition is approximated by an initial fixed rate followed by a decaying rate. The portion of the decay heat which goes into an increase in sensible heat of the reactor vessel is roughly compensated for by the use of an initial fixed rate of decay heat addition rather than a very high initial rate with an immediately decaying rate. More sophisticated forms for both these terms may be easily included when they are warranted.

The program does not include discharge to an intermediate region (i.e., a reactor well or dry well region) or vapor suppression in a wet well. These aspects of containment systems are currently being developed into the program.

Fission Product Simulation

Tests were made in the Aerosol Development Facility (ADF) using a steam-air mixture in the aerosol receiver. The aerosol was generated remotely and injected into the receiver through 18 feet of 1-inch SS pipe. Elemental iodine spiked with I-131 was passed through a furnace tube containing molten UO_2 (SS clad). The maximum initial total iodine concentration in the receiver was about 0.6 mg/m^3. Losses in the 1-inch injection line were acceptably low.

The total iodine "airborne" concentration in the presence of steam fell rapidly with a half-life of 2 to 4 minutes until the concentration was 0.001 of the original, after which time the half-life was measured at about 400 minutes. In comparison, air atmospheres gave an initial half-life of 15 minutes until the concentration was 0.1 of the original, after which a 125-minute half-life was measured. The more rapid attenuation in steam was caused by a combination of washout by steam condensate and by a higher deposition velocity on the aerosol receiver walls in the presence of steam.

Iodine sorbed on the receiver walls in the steam tests was not removed by further steaming, washing with water or scrubbing with detergent. In similar cases with aerosols deposited in an air atmosphere up to 90% of the iodine was easily removed.

One test was made in which SS-clad UO_2 was kept molten for 60 minutes in an air atmosphere.

OFF-GAS DISPOSAL TO GROUND

Work is under way on programming the LaPlace gas-flow solution into the stream program. The "Gas-Steady" program is being run, and initial results look promising from the application standpoint.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

A short study was made to define the magnitude of errors in travel times and path length when a three-dimensional flow system is solved both two-dimensionally and three-dimensionally. Twenty randomly-picked travel paths were calculated for the test model system used previously to check electrical analog techniques. Percentage deviations in both cases (path lengths and travel times) were over 100% from those determined by a three-dimensional analysis. In this case the travel times, based on two-dimensional analysis, were considerably in excess of those calculated three-dimensionally. Also evident from this exercise was the fact that the relative accuracy of evaluation of flow systems by either two- or three-dimensional methods is a function of the complexity of the system- and that there is no rational way of estimating the deviations short of actual analysis by both methods.

The counting equipment for measuring soil saturation, using Am-241 gamma ray transmission, is working satisfactorily and data are being collected on typical project soils. A problem in using Tygon tubing with the core-test fluid, Soltrol, was noted recently. Apparently the fluid dissolves the plasticizer out of this tubing. Thus, when Tygon is used on one side of a U-tube manometer with glass on the other side, errors in pressure measurement result. On the Tygon side, the density is increased, causing the interface in the glass tube to be higher than corresponds to the capillary pressure in the soil. The errors due to different densities can result in permeabilities in error by as large as a factor of four. Tubing containing no plasticizers (Saran or polyethylene) is being tested.

An 8-hour pumping test and subsequent recovery test was run on well 699-10-54 to attempt determination of aquifer characteristics. The well is located in Cold Creek Valley where there is little hydrologic information. Field coefficient of permeability is 260 gpd/ft^2. The aquifer occurs in the Ringold formation sands and gravels and the measured characteristics are consistent with Ringold coefficients determined at other locations.

Seismic monitoring of two additional Rattlesnake Hills quarry blasts were made. Data obtained supported information acquired from an earlier shot. Refracted wave travel times were on the order of 14,000-15,000 ft/sec, and the seismic signals were nearly pure sine waves at frequencies of 6-7.5 CPS. The first arrival waves did not have as great an amplitude as later waves, indicating wave reinforcement by reverberation.

Statements made in 1960 that ground waters theoretically could flow beneath the Columbia River between the Hanford Works and the Columbia Basin Irrigation Project are further substantiated now by new field data. Data earlier reported (March and April, 1964) suggested that the basalt aquifers beneath the north part of the Hanford Works are now charged partly from the Priest Rapids Dam pool and the upper Ringold Coulee area. Piezometric heads and other data from wells in other areas of the project are being evaluated to determine possible connections and recharge of confined aquifers on both sides of the river east of the project.

Soil Chemistry

Laboratory studies of anion-soil reactions are continuing. Equilibrium distribution coefficients (K_d) were determined for arsenate adsorption on a Hanford soil. Values ranged from 5.3 to 6.8 ml/g for adsorption from solution concentrations ranging from 10 to 100 ppm arsenic. K_d values of 10.0 and 13.0 were determined for trace arsenic adsorption from distilled water and tap water, respectively. No adsorption of the chromate ion ($CrO_4^=$) by soil could be detected.

RADIOLOGICAL AND HEALTH CHEMISTRY

Uranium Ore Inhalation Study

The final tissue samples from the rats exposed in the uranium ore inhalation experiment were analyzed during the month. The next step will be the analysis of the excreta obtained during the experiment to determine the excretion patterns during the exposure. Upon completion of this phase of the study the second phase utilizing dogs will begin.

Technetium Metabolism Study

Arrangements are currently in progress to collaborate in a technetium metabolism study with Dr. Will Nelp of the University of Washington Hospital and H.E. Palmer of the Radiological Physics Unit. The study will consist of measurements of the body distribution and excretion patterns of injected Tc-95 in ambulatory patients. The shadow shield whole body counting and slit scanning techniques will be used similar to the techniques used in the recent ore inhalation studies conducted in Colorado uranium mines.

Radiation-Induced Formation and Degradation of Aqueous Jellies

Textural change in the structure of mammalian tissue and other foodstuffs is a serious form of radiation damage. Synthetic aqueous jellies are convenient systems for studying this type of damage since they closely resemble tissue and since the types of chemical association and bonding in the jellies can be easily reproduced and varied. They also serve as model systems for testing the protective effects of chemical additives. Agarose, polyvinyl alcohol (PVA) and polyvinylpyrrolidone (PVP) were examined this month. A 0.20% solution of agarose forms a very firm jelly at room temperature which liquifies when an adsorbed dose of 6.8×10^5 rads is received from a Co-60 source. In the presence of 0.01M erioglaucine or 0.44M NH_4SCN liquefaction doses of 7.1×10^6 and $>2 \times 10^6$ rads are required, indicating that liquefaction occurs via hydroxyl radical intermediates. Neither PVA nor PVP in one percent solutions form jellies in water at room temperature if unirradiated but do so after absorbing 7.1×10^5 rads of γ-radiation. This cross-linking and jelly formation is inhibited by the presence of 0.01 M erioglaucine so that $>7.1 \times 10^6$ rads are needed for jelly formation. This indicates that the jelly formation also occurs via hydroxyl radical intermediates.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Aerosol Sampling Studies

The impaction efficiency for the filters and holders under test was redetermined for 30 mph. The calculated impaction efficiencies are consistent with impaction data for lower wind speeds, as well as consistent with efficiencies for very low sampling rates.

An analysis of the impaction efficiency data for filter supports used in the sampling study showed that the results could be reasonably correlated if the impaction efficiency for a given wind speed were plotted as a function of the Stokes Number,

$$\frac{\rho_p d_p^2 u_p}{18\mu r_f}$$

in which:

ρ_p = density of the particle

d_p = diameter of the particle

u_p = velocity of approach (wind speed)

μ = viscosity of air

r_f = filter radius

A family of curves results with wind speed as a parameter. At a low windspeed of 2.7 mph impaction efficiency increases linearly with the Stokes Number but with efficiencies less than predicted theoretically by Sell for a flat disc(1). At higher wind speeds the efficiency decreases as the Stokes Number is maintained constant. At the highest wind speed, 30 mph, the efficiency increases as the square of the Stokes Number. Through the intermediate range of wind speeds from about 9 to 15 mph a transition is observed producing a "knee" in the curve, indicating that the impaction efficiency varies with somewhat less than the first power of Stokes Number, then rapidly approaches the second power dependence on Stokes Numbers. Qualitatively, the lower velocities in this transition appear to correspond to the beginning of significant non-uniformity of deposit on the filters. At higher wind speeds deposition is prominently in an annular ring near the filter retainer, which pattern is frequently observed in particle sampling. The filter retainer is believed to have an important effect on flow distribution and deposition pattern on the filter.

The particle stopping distance is an extremely important factor in the behavior of particles as they approach an object. The casual calculation of stopping distance essentially ignores the progressive change in the drag coefficient as the particle stops or is brought to a constant velocity along its path. An empirical expression was derived using computer techniques to give a more nearly exact expression for the drag coefficient as a function of the instantaneous Reynolds Number. Better agreement with theory is anticipated from the use of more realistic stopping distances. Stopping distance with the refined calculation can differ by as much as 80 percent from that calculated assuming uniform drag in the range of particle sizes and velocities of interest.

Improved Aerosol Generators

Improvements in the design of spinning disc aerosol generators are under investigation. A promising concept is to make the supporting shaft of the spinning disc the armature of a synchronous motor. Speed should be accurately controllable by adjusting the frequency of the power supplied to the stator. A test model is planned which will make use of a gas-lubricated bearing. Improvements in air-driven systems are also being studied.

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(1) W. Sell, "Dust Precipitation on Simple Bodies and in Air Filters," Translation from FORSCH. GEBIETE INGENIEUR, 2 Forschung-sheft 347 (August 1931), by Paul Joseph Domotor, University of Illinois, February, 1951.

Theory of Thermal Repulsion of Particles

A critical review of the derivation of the currently accepted equation for the velocity of a particle in a thermal gradient revealed that unlike boundary conditions have been assumed for the particle moving under the influence of the thermal forces and the particle under the influence of the air drag force. This inconsistency in assumptions may lead to significant errors in the realistic assessment of particle behavior in a thermal gradient. A method of analysis yielding boundary conditions mutually compatible with the two force systems should yield an improved equation. A possible solution is being considered which will yield an equation with greater reliability.

Fallout Studies

The neutron activation product Ag-110m (249d half life) was observed in air filter collected fallout at Hanford and also in lichen samples from Alaska. This radionuclide has apparently not been previously reported in fallout so it is not known whether it is a normal (but very low level) constituent of bomb debris or if it comes from silver added to a specific bomb for tracer purposes. It is present in air at Hanford at about 1 dpm per million cubic feet.

Co-60 from fallout was measured for what appears to be the first time in a terrestrial animal (caribou). Both bone and flesh were found to contain this radionuclide. Whale flesh and bone were also found to contain Co-60. The concentrations were very low but measurable on ashed samples by low level multidimensional gamma spectrometry.

The Sr-90 concentrations in monthly composited fallout samples have been measured for 1963 and the first half of 1964. Together with the 12 radionuclides determined by multidimensional gamma spectrometry and the Pu-239 obtained by chemical separation and alpha counting this gives data for 14 radionuclides, and makes this set of samples one of the most extensively studied.

ISOTOPES DEVELOPMENT - 08 PROGRAM

Promethium Shielding Requirements

Calculations were made of the radiation dose rates associated with promethium heat sources and the results were compared with those of Pu-238 sources of equivalent heat output. The comparison was made both for Hanford-produced promethium, containing 0.5 ppm Pm-146, and for promethium from power reactor fuels containing 4 ppm Pm-146. Since

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shielding is concerned with reducing the effects of radiation to an acceptable level for the system under exposure, the dose rates should be compared in units representing equal radiation effects for the particular system in question. For example, in Arnold's shielding report, ORNL-3576, radiation dose rates are reported in terms of rep/hr, a unit appropriate only if energy absorption is important. In exposure of personnel and a great number of transistors, however, neutrons are respectively ten- and one-thousand-fold more damaging than gamma rays for equal energy absorption. When the factor of relative damage effectiveness is taken into consideration, it is found that transistors require greater shielding from a Pu-238 source of any size, then from a source of promethium, even one containing 4 ppm Pm-146. If dose rates to man are considered, however, promethium requires less shielding than Pu-238 for large sources, but more shielding for small sources.

Prototype Promethium Heat Source Program

Work on synthesis of candidate heat source compounds, measurement of compatibility with cladding, and refinement of burn-up calculation techniques continued. The fact that burn-up of many proposed capsules appeared marginal with the present somewhat simplified burn-up code has required that the code be amended to include the effect of second-order variables which were initially ignored.

Development

The model 1220F Dynapak high energy rate impaction machine was installed in the mock-up of C Cell in the 324 Building. Acceptance tests were performed.

Installation and shakedown has been completed on process equipment for converting fission product solutions to various fuel forms necessary for supplying feed to pneumatic impaction process. Several problem areas, such as cracking of the ceramic filters, have been identified and are being corrected.

Scope design was completed on a heavy duty transfer mechanism for C Cell and on a mechanism for heating isotope capsules and loading them into the pneumatic impaction machine. An engineering flow diagram and tentative equipment arrangements for C Cell were prepared for comment.

M. T. Walling, Jr.
Manager
Chemical Laboratory

MT Walling:cf

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BIOLOGY DEPARTMENT

A. ORGANIZATION AND PERSONNEL

M. S. Stack, Secretary transferred from I.P.D. to Biology Department - General on December 14th.

W. B. Peterson, Biological Analyst transferred from H.U.&P.O. to Experimental Animal Farm on December 14th.

Y. W. Hendrickson, Biological Analyst transferred from Technical Information to Biological Analyses on December 28th.

H. M. Robinson, Biological Analyst transferred from Technical Information to Biological Analyses on December 28th.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Columnaris

A monthly survey of agglutinating titers for C. columnaris in river fish was initiated. River fish will be taken near Bonneville Dam, Patterson's Ferry Landing, Hanford Slough, Priest Rapids and Wenatchee. Variation in titer due to season, species and sampling-area will be observed for a one-year period. An attempt will be made to correlate specific columnaris agglutinin titers to columnaris infectivity of river fish. Single samplings from Patterson's Landing, Hanford Slough and Priest Rapids showed average agglutinin values of 1:60, 1:150 and 1:200, respectively.

Reactor Effluent Monitoring

An experiment was started at 100-KE on December 7 to test the effect of different water temperatures and reactor effluent concentrations on the development of young chinook salmon. Approximately 1,300 freshly spawned eggs from Priest Rapids are being incubated for each of the eight experimental conditions. Egg mortality has thus far been negligible in all groups.

The holding of large adult chinooks for egg taking purposes in concrete raceways at 100-F now appears more favorable. Two out of six females held since early October matured before death. A greater proportion of males ripened before death.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY AND TRANSFER OF RADIOACTIVE MATERIALS

Phosphorus

Columbia River whitefish were force-fed P^{32} to obtain estimates of such parameters as biological halflife which cannot be measured in the field. Twenty-seven whitefish captured at Priest Rapids were each fed 27 μCi P^{32} and killed serially. Results at one and three days post administration show a much greater variation in body burden than observed in previous studies with rainbow trout. This variation may be related to species difference or differences between wild and hatchery reared fish.

Technitium

Procedures were developed for the analysis of Tc^{99} in animal tissues by scintillation counting. These involved wet ashing under total reflux followed by hydrogen peroxide treatment and careful addition of formaldehyde. Analysis of feces and urine samples still presents problems because of the formation of colored solutions.

Cesium

Nine male sheep have been fed 25 μCi Cs^{137}/day for 500-600 days. Cesium-137 burdens, as determined by whole-body monitoring, increased rapidly after initiation of feedings in January 1963 and plateaued approximately 100 days later, at levels ranging from 14 to 18 x the daily Cs^{137} intake in June and July when the mean environmental temperature was ∿ 70 F. During the fall months body burdens increased reaching levels of 16 to 24 x the daily Cs^{137} intake during December and January when the mean environmental temperature was 30-35 F. Following this peak during the coldest months, body burdens again declined to typical summer levels. This apparent correlation of Cs^{137} burden with environmental temperature may be related to underlying seasonal changes in metabolism or diet. Further studies are needed to substantiate and elucidate these findings.

Cerium

In contrast to previous findings with Pu^{239}, carrier free Ce^{144} was retained as well in livers of partially hepatectomized rats as in control rats. Addition of 0.2 mg of carrier cerium did not affect retention in liver but increased deposition in kidney, spleen, femur and blood.

Gastrointestinal Radiation Injury

A collaborative study with R. J. M. Fry, of Argonne National Laboratory, has shown that the transit time of cells from the crypts to the villi tips of bile duct cannulated rats was increased over that shown by control rats. This may be interpreted as an effect of bile salts on cell removal and hence on transit times.

Results from a study of the distribution of C^{14} labeled bile salts in rats, one day after intravenous injection, showed that exposure to 1500 R of X-ray doubled the amount of labeled bile salts in the lower bowel. The concentration of labeled bile salts in the small intestine of irradiated animals was one-fifth that of control animals, indicating dilution by newly synthesized bile salts. These results supported previous findings which suggested that irradiation brought about a decreased absorption of bile salts from the small intestine and a resultant increase in their excretion via the large intestine.

Co-carcinogens

Feeding the liver carcinogen diacetylaminofluorene (DAF) increased retention in the liver of previously injected Ce^{144}. However, animals injected with Ce^{144} after prolonged feeding of DAF showed less retention of the radioisotope than control animals. The increased retention of prior injected Ce^{144} is probably due to effects on biliary excretion. Such excretion is decreased by DAF feeding. Ligation of the bile duct practically eliminates fecal excretion of Ce^{144}.

Inhalation Studies

Only 1% of inhaled Sr^{85} fluoride remained in the lungs of rats at 24 hours post-inhalation. The major portion of the retained Sr^{85} was in the skeleton. These results agree with those obtained utilizing other routes of entry.

Pulmonary clearance of inhaled $Cr_2{}^{51}O_3$ was tested in dogs that had smoked 20 cigarettes/day, 5 days/week for 11 months. No alteration in pulmonary clearance was observed. Since the normal rate of clearance of $Cr_2{}^{51}O_3$ is very low the test was relatively insensitive. Similar tests will be made with $Fe_2{}^{59}O_3$ which appears to be cleared more rapidly.

Rats exposed repeatedly to uranium ore dust showed higher levels of uranium than thorium in the kidney, an effect opposite to that noted in the lungs and bronchial lymph nodes.

Several dogs were exposed to aerosols of Fe^{59} labeled microspheres to determine the deposition and retention of large particles. Animals exposed to either 10 or 20 μ spheres show deep lung burdens about one-third of initial deposition levels at one week post-exposure.

Plant Nutrition

Bentonite dust particles lost over 60% of adsorbed $I_2{}^{131}$ when 100 C air was drawn across them for three hours. About 50% of the adsorbed $I_2{}^{131}$ was lost in similar experiments with 50 C air. Similar particles were dusted onto two geranium plants. One of these placed in the airflow of a hood lost 30% of the $I_2{}^{131}$ over a 48-hour period. The other plant, retained in the static air of a pasteboard carton, lost none of the $I_2{}^{131}$. It has not been established whether loss of $I_2{}^{131}$ from the plant in the hood was due to desorption from the clay particles or to dislodgment of the particles from the plant.

Terrestrial Ecology

Pitfall trapping of autumn emerging darkling beetles in a sagebrush community was continued in 1964. The emergence period was essentially the same as in 1963 and extended from September to the first week in December with peak populations occurring in mid-October. Populations of Pelesayphorus were reduced to 65% of 1963 levels, and an even more drastic reduction to 41% of 1963 levels was observed in Stenomorpha populations. The population decline appeared to be related to the drought conditions of 1964. Data of the past two years indicate that a sizable population of darkling beetles can be expected every year. Present laboratory studies of radiation and insecticide effects in flour beetles could be extended to field populations of darkling beetles.

Population Dynamics of Hanford Wildfowl

From aerial census estimates about 94,000 ducks and geese were utilizing the Columbia River between the Yakima River mouth and Priest Rapids Dam. This represents a substantial increase over the 18,000 observed last month, but is only half the number estimated for the same period last year.

Alaskan Studies

Whole-body counting data on Anaktuvuk Pass Eskimos, obtained on 15-17 November, is summarized below:

Age Group	No. of Persons	Cs^{137} body burden (nCi)	
		Average	Maximum
Children (< 15 yrs)	32	150	500
Juveniles (15-20 yrs)	9	490	820
Adults (> 21 yrs)	39	760	1660

Average burden among 24 persons in the control group was 760 nCi, a decrease of 27% from September. Sampling of caribou and carnivore flesh and bone continued.

Special Problems

A technique for detection of latent fingerprints was developed. This technique involves the use of UO_2 as an alpha source, ZnS as a phosphor, and high contrast film to detect the emitted light. Prints were developed from a variety of materials including wood and paper.

R. Thompson

R. C. Thompson
BIOLOGY LABORATORY

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TECHNICAL INTERCHANGE DATA
BIOLOGY DEPARTMENT

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposia

None

b. Seminars (Off-Site and Local)

None

c. Seminars (Biology)

J. L. Palotay - Information Systems. December 16, 1964.
W. J. Clarke - The Use of Cell Organ Culture and Cytochemical Techniques in Toxicological Research. December 23, 1964.

d. Miscellaneous Lectures

R. E. Nakatani and J. M. Dean. Careers. Columbia High School. December 7, 1964.

II. Articles Published

a. Open Literature

Bair, W. J. Conclusion and acknowledgements. Health Physics 10: 1259 (1964).

Clarke, W. J., J. F. Park, J. L. Palotay and W. J. Bair. Bronchiolo-Alveolar Tumors of the Canine Lung Following Inhalation of Plutonium Particles. Am. Rev. of Respiratory Diseases 90: #6, 963-967 (1964).

Proceedings of the Hanford Symposium on Inhaled Radioactive Particles and Gases, Richland, Washington May 4-6, 1964. Health Physics 10: #12 (1964).

Palmer, H. E., R. W. Perkins and B. O. Stuart. The Distribution and Deposition of Radon Daughters Attached to Dust Particles in the Respiratory System of Humans Exposed to Uranium Mine Atmospheres. Health Physics 10: 1129-1135 (1964).

Park, J. F., W. J. Clarke and W. J. Bair. Chronic Effects of Inhaled Plutonium in Dogs. Health Physics 10: 1211-1217 (1964).

Stuart, B. O., H. W. Casey and W. J. Bair. Acute and Chronic Effects of Inhaled $^{144}CeO_2$ in Dogs. Health Physics 10: 1203-1209 (1964).

Tombropoulos, E. G. Review of Therapeutic Procedures for Removal of Inhaled Radioactive Materials. Health Physics 10: 1251-1257 (1964).

Kornberg, H. A. Introduction, Hanford Symposium on Inhaled Radioactive Particles and Gases. Health Physics 10: 861 (1964).

O'Brien, R. T. Effect of D_2O on uptake in Yeast. Proc. Soc. Exptl. Biol. Med. 117: 555-558 (1964).

Smith, V. H. Prevention of Plutonium Deposition by Desferrioxamine-beta. Nature 204: 899-900 (1964).

Sullivan, M. F. and R. C. Thompson. The Influence of Fractionated X-irradiation on the Intestine of Rats Protected by Cysteine and Partial-body Shielding. Radiation Research 23: 551-563 (1964).

b. HW Documents

None

III. Visits and Visitors

a. Visits to Hanford

Dr. Charles Kendeigh, Professor of Zoology, University of Illinois. Discussed mutual research problems with J. M. Dean and R. E. Nakatani and presented a seminar. December 8, 1964.

Fifteen members of the Washington State Medical Society and Health Department toured the Biology facilities with R. F. Palmer and W. J. Clarke, December 14, 1964.

Dr. S. Abraham, Department of Physiology, University of Calif., Berkeley. Introduced by J. M. Dean, Dr. Abraham presented a seminar December 15, 1964.

Twenty-five Catholic nuns, all teachers from Christ the King School in Richland, toured the Biology facilities guided by R. F. Palmer and W. J. Clarke.

b. Visits Off-site

12/1-6 - E. M. Uyeki visited Dr. N. K. Das, University of California, Berkeley and presented a paper at the Radiation Protection and Recovery Conference.

12/7-9 - J. V. Dilley traveled to the University of Chicago to discuss research problems dealing with radiation, and pesticide and oxygen toxicology with Drs. J. Doull, K. P. Dubois and E. L. Simmons.

12/8-9 - H. A. Kornberg discussed the Biology symposium and Biology program with the Division of Biology and Medicine, A.E.C., Washington D.C.

1234908 12/10-11 - D. D. Mahlum presented a lecture to Student Affiliate of Am. Chem. Soc. at University of Idaho.

IV. Achievements

None

V. Honors and Recognitions

None

VI. Professional Group or Organization Assignments

None

APPLIED MATHEMATICS OPERATION

MONTHLY REPORT - DECEMBER, 1964

ACTIVITIES FOR OTHER HAPO COMPONENTS

N-Reactor Department

A statistical test for outliers was specified and employed to detect and remove outlying observations from a set of pre-irradiation fuel data.

Work was begun on an experimental procedure to obtain tolerance limits for use with a gamma radiation fuel enrichment tester.

Irradiation Processing Department

An experimental design was formulated for the purpose of determining an optimal mix of four components in fuel fabrication.

A set of experimental data was analyzed concerning the comparative effects of 6-quench anneal treatments used in fuel fabrication. The comparative effects of these treatments on weight and on each of several dimensions of the fuel element were evaluated.

A statistical quality control procedure was devised which allows incoming fuel ingots to be weight-checked in sets of three rather than individually.

Data were analyzed relative to the frequency of reactor "temperature surges" as a function of specific causes.

A set of attribute sampling plans was developed for a quality control and maintenance program for certain reactor hardware.

ACTIVITIES WITHIN HANFORD LABORATORIES

2000 Program

Work was continued on the analysis of micron-size solid particle deposition on vertical conduit walls.

In the interest of economy, a two-dimensional EDPM program for ground water flow has been constructed from the current three-dimensional program and is now in use.

Preliminary analysis was finished on the experimental data from the calibration of the organic photometer to be used to monitor organic uranium stream concentration in the A-column test facility. Screened data will now be used to construct the final calibration function.

Work continued on the clean-up and documentation of functional programs and the general diagnostic program for the GE 412 process control computer. Clean-up work consists of minor rewriting of the programs to economize memory requirements and speed up arithmetic calculations.

4000 Program

Consultation, analytical and programming aid continued on heat conduction problems arising in the theory of nondestructive testing by thermal methods.

Analyses continued on alternative mathematical models being used to simulate carbon burnout in the graphite-zirconium compatibility study.

An analysis was made and an EDPM program written to describe the phenomenon of the spreading of a bounded ultrasonic beam as it travels through layered visco-elastic media.

Work continued on a formal Hanford report discussing the mathematical problems of unfolding spherical partical-size distributions. Current efforts are directed toward constructing appropriate illustrative examples which compare the results of these mathematical techniques with earlier ones due to Saltykov and Johnson.

A regression analysis was carried out on experimental data for the purpose of obtaining estimates and confidence intervals for a coefficient of thermal expansion (by exposure level) for several types of graphite.

A linear statistical model was formulated for estimating the overall error associated with current determinations of the plutonium content in Pu-Al billets.

5000 Program

New mathematical techniques for evaluation of program data are being coded for use with the digital computer. In this connection several computer routines were converted from Fortran 2 to Fortran 4. Entry of data into the BLU system continues. Presently BLU contains the equivalent of 1574 white cards.

Power function calculations were started for the uniformly most powerful unbiased test of the hypothesis that a set of observed counts arose from a pure background uniform intensity source as opposed to the alternative hypothesis that they arose from a decaying source. The power function tables will be included in a formal report describing the mathematical and statistical techniques associated with testing this hypothesis.

Several subroutines in the main calculation pass of the IRA system have been modified to reduce running time. The modification reduced the number of times standard spectra files must be searched by saving pertinent information from previous searches.

6000 Program

A comparative study was completed of the 1963 (monthly) and 1964 (quarterly) personal film badge exchange plans.

Data were analyzed relative to the effect of DTPA treatment on the removal of inhaled cerium dioxide.

A statistical model was formulated in connection with the amounts of intravenously and orally administered C_s^{137} retained by sheep.

Statistical services were provided on a study of the protection against radiation-induced damage afforded by the sulf-hydral compound dimethyl sulfoxide (DMSO).

Carl A. Bennett
Manager
Applied Mathematics

CA Bennett:lg

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF DECEMBER 1964

A. ORGANIZATION AND PERSONNEL

H. E. Hanthorn completed the advance engineering and related work for Chemical Processing Department and returned to the Nuclear Health and Safety group. Elma Liddel transferred from the Analytical Laboratories to CDS&R on December 28, 1964. Agnes Hanke transferred from CDS&R to Personnel Accounting on December 7, 1964. W. E. Parker transferred to IPD Radiation Monitoring, effective December 21, as part of the Displacement Transfer Procedure. L. L. Monroe transferred to CPD effective December 28, 1964. O. M. Hanson, Maude H. Holeman, K. K. Knapp, J. M. Wirth, R. M. Allen, R. L. Farber, W. H. Neal, and C. A. Tompkins terminated for transfer to U. S. Testing Company, Inc., on December 31. O. M. Hanson, Maude H. Holeman, and K. K. Knapp also optionally retired from the General Electric Company on that date. R. D. George transferred from IPD to Radiation Monitoring effective December 21. E. P. Bowling, J. E. Coleman, F. C. Hatch, F. L. Meeks, and A. E. Pentilla transferred from IPD to Radiation Monitoring effective December 28. D. O. Erickson terminated from Radiation Monitoring effective December 18.

B. ACTIVITIES

There were no new plutonium deposition cases confirmed by evaluation of bioassay data during the month. The evaluation of additional urine analysis data resulted in one employee and two previously terminated employees being reclassified as nondeposition cases. The total number of individuals who have received internal depositions of plutonium at Hanford is 316, of which 225 are currently employed.

There were five incidents involving six employees which required special bioassay sampling for plutonium analysis. A CPD employee received facial contamination of 5000 d/m and nasal contamination indicated by smears of 1800 d/m plutonium as a result of work in the Plutonium Purification and Fabrication Facility (234-5). The employee was working at a hood when one of the hood gloves ruptured. Exposure time was estimated at less than two minutes.

A CPD employee received a plutonium nitrate contaminated injury on the tip of the small finger of the right hand on December 5, 1964 as a result of work in the Plutonium Purification and Fabrication Facility (234-5). The object causing the wound was not determined. Initial contamination of >40,000 d/m at the wound site was reduced to 500 d/m. Blood smears indicated 10,000 d/m on a portable survey instrument. The initial examination, using the Plutonium Wound Counter, indicated 1.4×10^{-2} μc plutonium in the wound. Re-examination in the wound counter, following surgical excision, indicated less than the detection level of 1×10^{-4} μc plutonium remaining at the wound site.

An HL employee received a plutonium contaminated wound on December 4, 1964 as a result of work in the Plutonium Metallurgy Facility (231-Z). The employee was placing plutonium turnings in a metal can when a piece of turning penetrated his glove and punctured the middle finger of his left hand. An initial examination, using the Plutonium Wound Counter, indicated 1.2×10^{-3} μc plutonium in the wound. No excision was performed. A recount three days later indicated 1.1×10^{-3} μc plutonium in the wound.

There were two plutonium contaminated injuries this month. The total number for the year is fourteen, with eleven requiring excision. In 1963 there were twenty plutonium contaminated injuries, with fourteen requiring surgical excision.

In addition to the incidents involving plutonium, there were two incidents involving four persons that required evaluation for possible intake of other radioisotopes. Neither of these are considered significant.

The processing of the fourth quarter, 1964 and the December, 1964 beta gamma film badge dosimeters was completed and readied for data processing and distribution of reports on January 4, 1965. Processing of finger ring and neutron dosimeters could not be completed by January 1, 1965 and those processed and on hand were given to U. S. Testing Company, Inc., for further processing on December 31, 1964.

Other preparations were completed for the transfer of film dosimeter processing to U. S. Testing Company, Inc., effective January 1, 1965.

A method for estimating the amount of dose received from N^{16} radiation in a field of Ra γ radiation was developed. Because the film darkening per unit dose for N^{16} is greater than that for Ra γ, the net result for film dose from the November processing when adjusted for N^{16} was a lowering of the gamma dose by up to a factor of three.

A new wound counter probe was received from the Nucleonic Instrumentation group for use with the semi-portable wound counting equipment which was installed at the 700 Area First Aid building. This probe is a 1/4" x 1 mm NaI detector for use in more accurately pinpointing the location of contaminated material in wound sites. An additional large-area probe plus two geiger-type probes were also installed to provide the capability to detect any gamma emitter which may be encountered in a contaminated injury.

Plans for new phosphorus-32 detectors were drawn up for submission to Technical Shops for fabrication. Plans for modification of the Mobile Whole

Body Counter were completed. This modification will allow the steps and exterior cabling to be carried with the trailer, rather than requiring use of our auxiliary vehicle.

Environmental Experience

Concentrations of fallout materials in the air of the Hanford environs remained at the low average of 0.2 pc β/m^3 for the four week period ending December 25, 1964.

Levels of radioactivity found in the Columbia River and in fish and duck flesh were normal for this time of year. No unusual circumstances were noted during the month.

Failure of a heat exchanger tube at 100-N resulted in the release of primary coolant to the river. Although the radioactivity released was estimated to be 1000 curies, such a large fraction was short half-lived that nothing was detected in our river water samples.

Vegetation samples, collected at the edges of the contaminated Gable Mountain and 'B' Swamps, showed that a significant vegetation uptake of strontium and cesium has occurred. Contamination levels as high as 2×10^{-4} μc/gm Cs^{137}, 0.04 μc/gm Sr^{89}, and 0.02 μc/gm Sr^{90} were noted. ES&EO personnel had already formally indicated to CPD that a soil sterilization program would be necessary and such a program has now been carried out. When the growing season starts, close surveillance will be given to assure that the soil sterilization program is effective.

Nuclear Safety

The safeguards studies of Plutonium Recycle Test Reactor Operation, completed last month and issued for comment, are now ready for publication as HW-61236 SUP7, Plutonium Recycle Test Reactor, Final Safeguards Analysis, Supplement 7, Analysis of Increased Power Level, by L. J. Nitteberg and N. G. Wittenbrock, dated December 4, 1964. Comments within the Laboratory were satisfactorily resolved.

Safeguards studies needed for the high power density core operation mode of PRTR continued. Comments were provided on proposed operation of oxide fuel elements in the PRTR with a substantial fraction of the fuel in a molten condition. An independent review of the information gathered during the leakage tests of the PRTR containment vessel neared completion.

A special condition was prepared to permit simulation of a Bonneville Power Administration outage to demonstrate the adequacy of the primary system to withstand such an outage.

The Richland Operations Office of the AEC issued Operating Safety Limits - Plutonium Recycle Critical Facility, document number RL-OSL-PRCF, dated November 10, 1964. This document is in complete agreement with our transmittal to the Commission on October 26, 1964.

Operating Safety Limits for the Physical Constants Testing Reactor and the Thermal Test Reactor were transmitted to AEC-RLOO on December 14, 1964, as HW-84557, Thermal Test Reactor, Operating Safety Limits, December 9, 1964; and HW-84564, Physical Constants Testing Reactor, Operating Safety Limits, December 11, 1964, F. Swanberg, editor, after preliminary agreement was reached on the content.

Preparation of operating limits for the Critical Mass Laboratory and the Critical Approach Facility were deferred at the request of the AEC-RLOO. A Final Safeguards Analysis for the Critical Approach Facility, HW-84585, neared completion.

A formal safeguards review was conducted of target element meltdown experiment. The potential exposure which could result from accidental release of tritium was estimated and advice was provided to the groups conducting the experiment or interested in the potential release of tritium.

Work on the computer program "Intake Distribution and Excretion of Radionuclides in Mammals" was interrupted while the analog computer was moved to the 3201 Building.

Additional progress was made on computer programming and utilization of existing programs in dose calculations from postulated accidents, in the calculation of reactor excursion transients, and in calculation of radiation level through various types of shielding. The latter program was initiated to examine the adequacy of casks for shipment of fuels and can include many other types of calculations of a similar nature. The postulated accident calculations include anticipated future operating conditions of the PRTR.

Routine audits performed during the month were as follows:

PRTR - 20
PRCF - 7

Consultations

Information was provided the Chemical Processing Department regarding the potential hazards of shipment of neptunium-237 as a nitrate. This information is being included in a document now being prepared by CPD.

ES&EO personnel supplied consultation services to PRTR Operation personnel on effluent monitoring requirements for proposed changes in containment procedures for PRTR.

Consultations were provided IPD staff members on reactor effluent monitoring and sampling, and with NRD on radiation protection procedures and river water activity.

Consultation was provided for IPD and NRD on the placement and alarm trip point for Critical Radiation Detectors in the 300 Area.

Consultations were provided to IPD personnel on possible methods for eliminating tritium exposure problems in the reactor drying rooms during personnel entries.

Consultations were held with Metallurgical Development concerning Tc^{99} and associated hazards; with IPD - Fuels Engineering pertaining to the establishment of a thorium laboratory in the 3706 Building; with Property Management concerning the disposal of 11,000 pounds of aluminum tubing containing low-level uranium contamination; and with Receiving concerning the monitoring and release of empty pallets from 300 Area.

The stability of a Sinclair Phoenix Photometer was studied during the past several months at the request of Analytical Laboratory. The instrument was found to drift downscale from 25 to 50% of the chart range under normal cold start conditions during a warmup of about two hours. Downscale drift was limited to about 12% over a two-hour period by the installation of a heater element to maintain a relatively constant temperature.

Studies and Improvements

An alphabetical card file containing data for approximately 23,000 individuals who worked for construction contractors at Hanford from 1944 to 1956 was produced. This file consists of data processing cards containing the names of the individuals, the names of the companies for which the individuals worked, and the payroll numbers assigned the individuals.

New exposure record folders and microfilm jackets were prepared for all employees transferring to Battelle-Northwest. Alphabetical and numerical cross index files were also established.

A river water sampler was installed at the 100-F forebay. This sampler obtains a sample of the raw river water by a timer and solenoid valve arrangement on the line that delivers raw water to Biology Operation.

Hanford Laboratories' experiences with high exposure plutonium are being collected into a summary report at the request of the AEC. RDCO is participating with other Hanford Laboratories groups in contributing to this report. Specifically, the objective of the report is to analyze the facility design and cost differences of processing high exposure plutonium compared to processing low exposure plutonium. It appears, based on the data analyzed to date, that the cost for facility design or modification and for routine processing would increase very slightly.

Shielding requirements for a special waste calcination sampling hood were calculated. The radiation source for the calculation consists of 10 cc of a 50 gallon solution containing 1 PRTR fuel element that was irradiated for 10,000 megawatt days per ton and cooled 15 days. Calculations indicate that 6-1/4 inches of lead shielding will be required.

The final draft of the design criteria for the Radio-Surgery facility did not include shielding specifications for walls of the surgery room, although they were included in the earlier draft of the design criteria. The necessary changes were given to the Engineering Services and Projects personnel who have now revised the shielding section of the design criteria. Other shielding ambiguities were clarified.

Eighteen circuit diagrams of the new mechanized densitometer logic system are being prepared by Drafting. When completed, drawings detailing all mechanical, electrical and physical detail layouts for this device will be available. A total of 61 drawings per set were required for detailing this information. The use of semiconductor transistors and diodes throughout the design of this equipment has provided maintenance-free operation for all test evaluations and for the first three personnel dosimeter exchanges. A total of 3074 semiconductors are included within the densitometer.

One day was devoted to testing and adjusting the "old" mechanized film densitometer to assure its satisfactory operation prior to release for transfer to the U. S. Testing Company.

A study of low level plutonium air concentrations in the Weapons Fabrication Facility was continued with the collection of weekly samples at six selected locations. These samples collected over the last 10 weeks were all from locations within Zone III. A detailed analysis of the plutonium content of these samples indicate that air concentrations to a maximum of 8.4×10^{-12} µc Pu/cc and a minimum of 2×10^{-15} µc Pu/cc were observed. The use of Lexan plastic and the fission fragment damage principle for analysis of air samples with very low plutonium concentrations was studied.

The thermoluminescent neutron dosimeter reader was modified to permit processing and reading of much larger samples of LiF phosphor. The use of larger phosphor quantities should result in increased sensitivity and a lower neutron dose detection limit. Special larger planchets are now being fabricated for use with the larger samples. Neutron exposure time on the positive ion accelerator was delayed considerably due to building flooding necessitating repairs to equipment and the building structure.

Three criticality surveillance instruments were received from the Technical Shops and calibrated. These devices are ready for installation about the 300 Area to verify, not signal, the occurrence of possible criticality events. To avoid false verification, possibly due to the movement of highly radioactive materials, the detectors were designed to have a sensitivity to neutrons over a wide energy range, but no gamma dose response. The instruments were designed by the Nucleonics Instrumentation group. The recorders to provide a record of the neutron flux at each instrument location were not received, delaying the actual installation.

Substantial improvements in the calibration and servicing of G.M. instruments reduced operating costs during 1964. The average monthly cost for G.M. calibration and repair for 1963 was $3938. For 1964 this was reduced to $3246 per month, representing an annual savings of $8300. The savings resulted primarily from a change in the batteries used to power the instruments.

A total of 118 radiation monitoring instruments received an audit calibration as part of the Calibrations Unit quality control program. In 1964 1168 instruments received audit calibrations, which indicated a failure rate of about 3% due to all causes. A total of 11 Emergency Monitoring Kits were serviced during December. As a part of the year-end servicing of these kits, all face masks were exchanged and sent to the mask decontamination station for inspection and testing. The portable instrument repair frequency was 4.7% above the November rate. The 1964 repair frequency was 5.2% below that observed for 1963. A total of 108 hours were spent on special studies; the 1964 total for special studies is 1132 hours.

Research Studies

Effect of Reactor Effluent on Quality of Columbia River Water

A study is in progress to determine the effects of reactor effluent on Columbia River water quality with emphasis on temperature effects. The interim report covering progress on the temperature phase of the study was issued. The report of 1963 work on the chemical effects phase of the study was held to be combined with 1964 progress.

River work came to a halt as a result of the severe weather conditions. Compilation and reduction of routine and previous test data continued.

Mechanisms of Environmental Exposure

An experiment to study the uptake and retention of P^{32} and Zn^{65} by volunteers eating Columbia River fish ended with measurements of P^{32} in excreta to study elimination rates. The experiment involved eleven individuals and covered an ingestion period of eleven weeks. The maximum P^{32} body burden attained from fish consumption was slightly more than 300 nc. The fish flesh available for the experiment contained somewhat less P^{32} than was originally anticipated but the resulting body burdens were several times the detection level on the new P^{32} whole body counter.

Nuclear Facilities Monitoring Guide

An attempt to describe a method for evaluating the radiological significance of each of the radionuclides appearing in power reactor wastes was unsuccessful. The intent of the effort was to develop, by theoretical calculations, a logical method for identifying the few radionuclides that nearly always require the major environmental radiation protection effort. Current reactor designs and practices appear to so effectively control wastes that, based on past experience, only the release of radioiodine and, in some cases, the noble gases, require special attention. A summary description of the characteristics of radioiodine which are important to its control is three-quarters completed.

C. TRAINING

Two 2-hour sessions on Civil Defense Monitoring were provided for members of HL Radiation Monitoring and Environmental Monitoring personnel.

One 8-hour training session on Civil Defense Monitoring was presented at Moscow, Idaho on December 11, 1964 for employees of the U. S. Department of Agriculture, including Soil Scientists, Veterinarians, and State Civil Defense workers.

An orientation on radiation practices at the Plutonium Fuels Pilot Plant was presented to two Japanese exchange employees.

The final sessions on "Radiological Civil Defense" and "Control of Tritium Radiation Problems" were held in the Second Radiation Monitoring Refresher Course. In addition, the first four sessions on "Application of Dose Rate Instruments to Field Use" were held.

D. SIGNIFICANT REPORTS

HW-80888 (Secret) "Effect of Reactor Effluent on Columbia River Temperatures - Interim Report No. 2," J. P. Corley, December 10, 1964.

HW-83723 "Evaluation of Radiological Conditions in the Vicinity of Hanford, January - June, 1964," R. H. Wilson, Editor.

HW-80892-11 "Radiological Status of the Hanford Environs for November 1964," R. F. Foster.

HW-84605 "Radiation Monitoring Operation Monthly Report," December 1964, A. J. Stevens.

HW-61236 SUP7 "Plutonium Recycle Test Reactor, Final Safeguards Analysis, Supplement 7, Analysis of Increased Power Level," December 4, 1964. L. J. Nitteberg and N. G. Wittenbrock.

HW-84557 "Thermal Test Reactor, Operating Safety Limits," December 9, 1964. F. Swanberg, Editor.

HW-84564 "Physical Constants Testing Reactor, Operating Safety Limits," December 11, 1964. F. Swanberg, Editor.

A paper entitled "Plutonium Aerosol Particle Size Distribution in Room Air," by B. V. Andersen was published in the Health Physics Journal, December 1964 issue, Volume 10, No. 12.

A paper entitled "Safety Aspects of Transport of Radioactive Materials" was presented by L. L. Zahn during the 57th Annual Meeting of the AIChE in Boston, Massachusetts on December 10, 1964. R. J. Junkins, Coauthor.

AR Keene

Manager
RADIATION PROTECTION

AR Keene:np

FINANCE AND ADMINISTRATION OPERATION

ACCOUNTING

Cost Accounting

The following minor adjustments were reflected in the Hanford Laboratories' control budget during the month:

	Increase (Decrease)
Fueled Graphite Studies	$ 50 000
Major Conference Display	7 000
HAPO Service Assessments	(11 000)

In addition, a work order for $30,000 was received from RLOO-AEC to help cover costs associated with fire damage in the 146-FR Building. The work order is to help meet restoration costs at 146-FR and to partially offset costs of repairing equipment.

The following Midyear Budget Review program data was submitted to RLOO-AEC during the month:

1. Monthly expenditure pattern estimates for FY 1965 for the HL 02, 04, 05, 06, 07 and 08 Research and Development Programs.

2. Latest FY 1966 estimates for the 04 and 05 Research and Development Programs.

3. Equipment expenditure forecast for FY 1965 for all Research and Development Programs and FY 1966 estimate for the 04 and 05 Programs.

This completed all activities associated with the FY 1965 Midyear Budget Review.

Activities for which special accounting codes were established are:

.2X - Fabrication of uranium rod and wire for Monsanto Research Corporation - $180.

.2Y - Laboratories has been requested to Nupak UO_2 for Atomic Power Equipment Department, GE - $1,240.

.2Z - Laboratories has been requested to ship 3 Kg. of 1% of Pu to Atomic Power Equipment Department - $402.

.5N - Requested to fabricate and ship enriched uranium wire to Israel-AEC - $335.

.5R - Work order received from RLOO-AEC for the ATR Gas Loop Project - $29,500.

General Accounting

Approval letter No. AT-364, AEC Monograph on Plutonium - Its Industrial Hygiene Aspects - Dr. R. C. Thompson, was approved by the Commission on December 7, 1964. Approval letter No. AT-372, 146-FR Fire - Personal Property Losses, was approved by the Commission on December 30, 1964.

Hanford Laboratories' net material investment at December 1, 1964 totaled $27.9 million as detailed below:

	(In thousands)
SS Material	$ 26 551
Reactor & Other Special Materials	997
Spare Parts	388
Exotic Materials	56
Subtotal	27 992
Reserve: Spare Parts	(84)
Net Inventory Investment	$ 27 908

The cumulative value of nuclear material consumed in research by Hanford Laboratories during FY 1965 (at December 1, 1964) is shown below:

02 Program	$ 31 246
03 Program	329 969
04 Program	(17 006)
05 Program	42 483
Total	$386 692

Credit indicated in the nuclear material consumed in research account is due primarily to return of material to Redox at full value.

A total of 613 Extended Property Passes covering the routine movement of specific categories of property within the plant areas during CY 1965 were prepared for distribution to BNW personnel in January 1965.

The status of Hanford Laboratories' heavy water inventory at December 31, 1964 is as follows:

	Current Month		FY to Date	
	Pounds	Value	Pounds	Value
Beginning balance	56 061	$ 751 883	32 762	$ 447 572
Acquisitions	--	1 087	27 445	380 102
Scrap returns (SROO)	(17 491)	(217 375)	(17 491)	(217 375)
Removals (Off-site)-1)	(30)	(417)	(38)	(522)
Consumption-2)				
PRTR: Loss	(965)	(13 337)	(5 103)	(70 540)
Scrap	--	(5 311)	--	(22 707)
Ending Balance -3)	37 575	$ 516 530	37 575	$ 516 530

(1- Heavy water shipped to the University of Oregon Medical School.
(2- Consumption - Scrap reflects value charged to Cost only.
(3- Includes 2,819 pounds of heavy water scrap valued at $35,045.

A billing was submitted to Savannah River Operations Office for the December shipment of heavy water, consisting of 17,491 pounds, with a fund value of $217,375 and a nonfund value of $107,729.

The following contracts were processed during the month:

CA-481	Clyde R. Jensen and Walter A. Ricker, d/b/a, Laboratory of Clinical Medicine
CA-482	Richard S. Miller
CA-484	Dr. David C. England
CA-485	Charles H. Naundorff
DDR-198	Basic Products Corporation
SA-376	Schwarzkopf Microanalytical Laboratory
SA-382	Scientia Research Laboratories, Inc.
SA-384	W. E. Welsh
SA-385	L. M. Bodie
SA-386	P. M. Aldrich
SA-387	Robert J. Donohue
SA-388	Betz Laboratories, Inc.

Personnel Accounting

Suggestion awards for December, which amounted to $1,150, were paid to 33 employees. There were no outstanding suggestions at December 31, 1964.

M. H. Holeman, K. K. Knapp, and O. M. Hanson took optional retirement effective January 1, 1965.

Personnel statistics follow:

Employee Changes	Total	Exempt	Nonexempt
Employees at beginning of month	1 851	798	1 053
Additions and transfers in	111	21	90
Removals and transfers out	32	10	22
Employees at end of month	1 930	809	1 121

Overtime Payments During Month	December	November
Exempt	$ 7 265	$ 8 132
Nonexempt	27 378	29 169
Total	$ 34 643	$ 37 301

Gross Payroll Paid During Month	December	November
Exempt	$ 828 511	$ 819 973
Nonexempt	773 754	621 971
Total	$1 602 265	$1 441 944

Participation in Employee Benefit Plans at Month End	December		November	
	Number	Percent	Number	Percent
Pension	1 840	99.3	1 752	98.3
Insurance Plan - Personal	385		381	
- Dependent	1 520	99.0	1 460	99.6
U. S. Savings Bonds				
Stock Bonus Plan	201	37.0	170	35.6
Savings Plan	72	3.7	70	3.8
Savings & Security Plan	1 203	87.0	1 199	86.9
Good Neighbor Fund	1 376	71.5	1 336	71.9

Insurance Claims	Number	Amount	Number	Amount
Employee Benefits				
Life Insurance	-0-	$	-0-	$
Weekly Sickness & Accident	10	1 037	6	437
Comprehensive Medical	67	4 732	37	2 146
Dependent Benefits				
Comprehensive Medical	130	10 963	89	9 351
Total	207	$16 732	132	$11 934

TECHNICAL ADMINISTRATION

Forty-nine nonexempt employment requisitions were filled; 62 remain to be filled.

Suggestion Plan activity:

Suggestions received	0
Suggestions adopted	28
Suggestions rejected	12
Suggestions in process	0

Visitors Center activity:

December attendance	860
Average attendance per day open	33
Cumulative attendance since 6-13-62	83 226
Conducted groups	3 (totaling 74 people)

Plant tour activity:

	Number	Total People
General public relations tours	3	74
Special tours	1	2

Overall recruiting results for December were as follows:

Offers extended	7
Offers rejected	0
Offers accepted	2
Added to roll	3

Advanced Degree - Four Ph.D. applicants visited HAPO for employment interviews. One offer was extended; no acceptances and no rejections were received. Three offers are currently open.

BS/MS (Direct Placement) - One offer was extended. One acceptance and no rejections were received. There is one offer currently open.

BS/MS (Program) - Five offers were extended. One offer was accepted and no offers rejected. There are four offers currently open.

Technical Graduate Program - One Technical Graduate was placed on permanent assignment. No new members were added to the roll and no one terminated from the roll. Current Program numbers 18.

FACILITIES ENGINEERING

Projects

At month end, Facilities Engineering Operation was responsible for 13 active projects having total authorized funds in the amount of $10,729,000. The total estimated cost of these projects is $11,917,000. Expenditures on them through November 30, 1964 were $5,585,000.

The following summarizes project activity in December:

Number of authorized projects at month end -------------------- 13

Number of new projects authorized in December ---------------- 0

Number of projects completed in December --------------------- 0

New projects submitted to the AEC in December ---------------- 1
CAH-159, Geological & Hydrological Wells - FY 1965

Projects awaiting AEC approval ------------------------------- 3
CAH-123, Laboratory Fire Protection System
CAH-157, Services for Biology Laboratory and Future Facilities - 300 Area
CAH-159, Geological & Hydrological Wells - FY 1965

Project proposals complete or nearing completion ------------- 2
Power Supply - Million Ampere Welder
PRTR Increased Power Level

Other project proposals under way ---------------------------- 2
309 Building - Experimental Facility Addition
Shielded Creep Test Facility - 3707-C Building

The status of active projects follows:

CAH-100 High Temperature Lattice Test Reactor

Detailed design is about 97 percent complete compared to a scheduled 100 percent. The graphite stack drawings were approved. The reactor case, insulation, gas piping, and Phase II electrical drawings have been commented upon, but are not approved. Phase II instrument drawings, other than the PMAC System, have not been issued for comment.

Construction is about 1.5 percent complete compared to a scheduled 2.7 percent. The contractor moved in temporary construction buildings. Excavation for the building is essentially complete. Work on utility systems has been started. The J. A. Jones Company started procurement of Phase II equipment including heat exchangers, filter case, vacuum pump, and gas blower.

Bids on PMAC Systems were opened December 21, 1964 and are currently being evaluated.

CAH-116 PRTR Decontamination and D_2O Cleanup

Design is complete; however, construction specifications for the decontamination facility and all ATP's are yet to be completed. Construction of the D_2O cleanup system is 17 percent complete compared to a scheduled 18 percent. Fabrication of the tanks, enclosure hood, and supports and procurement of process equipment items were started. The Commission deferred action on the project proposal revision requesting authorization of funds to initiate procurement and fabrication of the PRTR decontamination equipment.

CAH-119 PRTR Storage Basin and Experimental Facilities Modifications

Construction is about 22 percent complete compared to a scheduled 24 percent. The concrete basin walls have been poured up to the (-) 15 foot level. Piping and electrical work is continuing. Structural steel was shipped from Seattle on December 28. Bids for the underwater equipment were opened December 21, 1964. Only two bids were received. They are currently being evaluated.

CAH-123 Laboratory Fire Protection System

No action has been taken by the Commission on the project proposal submitted on March 3, 1964.

CAH-126 Waste Transport System

Design is 96 percent complete compared to a scheduled 100 percent.

Vitro is preparing a study of the waste unloading system.

CAH-136 Service Addition - 327 Building

Design is 99 percent complete. The detailed design drawings were received for review and approval. They were approved with minor exceptions. The final issue of the construction specifications has not been issued for approval.

CAH-137 Temporary Physical Sciences Center

Design is 100 percent complete. Construction at the 3201 Building is 98 percent complete. The fire protection sprinkler system contractor completed piping and is ready to initiate testing. The telephone serial cable installation is complete.

CAH-146 Atmospheric Physics Building

The architect-engineer is working on preliminary drawings of the building. Conceptual drawings were submitted for Company and Commission comments. Our comments were transmitted to the Commission. Title I design is 95 percent complete; the A-E indicates it will be complete the first week in January.

CAH-151 Office Addition - 308 Building

Design is complete. All drawings have been approved.

CAH-153 Plutonium Recycle Critical Facility, Irradiated EBWR Fuel Handling

Design is complete. Procurement of the fuel element handling cask was initiated. Construction is scheduled to start January 4, 1965.

CAH-157 Services for Biology Laboratory and Future Facilities - 300 Area

The project proposal was approved December 17. The design criteria document has been approved and was transmitted to the AEC on December 31, 1964.

CAH-159 Geological and Hydrological Wells - FY 1965

A project proposal requesting authorization of total project funds was submitted to the Commission on December 14, 1964.

CGH-161 PRTR Increased Power Level

The project proposal was returned from the General Electric Company, General Manager's office, and will be submitted as CBB-001, the first Battelle project proposal.

CAH-916 Fuels Recycle Pilot Plant

Construction is 83 percent complete compared to a scheduled 81 percent. Insulated metal siding was installed on most of the building exterior. Metal roof decking was placed over most of Area 5. Plastering, concrete block laying, and painting of interior partitions are continuing. Installation of ductwork, electrical wiring, and piping is continuing. The top mat of reinforcing steel was installed for the roof slab of the Hot Pilot Cell.

The directive completion date has been extended to September 30, 1965, as a result of time lost due to the plumbers' and pipefitters' strike.

CAH-962 Low Level Radiochemistry Building

Construction is 45 percent complete and on schedule. Construction of clean room partitions was started. Installation of heating and ventilating equipment, piping, and electrical services continued. Installation of the aluminum front entrance and interior plastering were started.

CAH-977 Facilities for Radioactive Inhalation Studies

The Commission project review board approved the project proposal. A transmittal letter, recommending authorization of the project by AEC-Washington, is being prepared. The approved design criteria were transmitted to the AEC on December 23, 1964.

CAH-982 Addition to Radionuclide Facilities

The Commission's project review board approved the project proposal and is preparing a letter recommending authorization by Washington. The approved design criteria were transmitted to the AEC on December 23, 1964.

Engineering Services and Plant Engineering

Engineering work was performed in support of design and construction on active projects, project proposals, preliminary planning, and design criteria for new projects. The following projects were involved: CAH-100 - HTLTR, CAH-119 - PRTR Storage Basin and Experimental Facilities Modification, CAH-146 - Atmospheric Physics Building, CAH-153 - PRCF Irradiated Fuel Handling System, CAH-916 - FRPP, and CAH-962 - Low Level Radiochemistry Building.

Engineering and consulting work was provided to research and development personnel as requested. Major work items included: (1) change in scope for a sample load-out hood (324 Building) requiring a new set of drawings, (2) study of 324 Building manipulator handling, (3) assistance to the Waste Calcination Engineering group in reviewing "packages" of work for additions to the 324 Building as part of the waste calcination program, and (4) design of a furnace and tilting dissolver holder for 292-T hot cell.

Plant engineering work included: (1) preparation of sketches to procure and install a tone generating system for the plant standard evacuation alarm over the 308 Building public address system, (2) review of sketches and purchase specifications for a proposed 440 volt service for a materials engineering laboratory in 314 Building, (3) preparation of purchase specifications for the process water system in 300 Area, (4) development of procurement specification and purchase requisition for a replacement shielding window assembly for the 325 Building Analytical Hot Cell, (5) preparation of comment for a proposed modification of the electron microscope laboratory in the 326 Building, (6) determined engineering feasibility of transmitting an alarm signal via telephone to patrol headquarters when any of the 3201 Building vault doors are opened, (7) simplified 3201 Building computer air conditioning installation design to permit completion of work within allowable funds, (8) outlined a special ground system for the computer installation in the 3201 building, (9) recommended corrective action for the floor settlement in the 3745-B Building caused by excessive drainage water from melting snow, and (10) prepared electrical layout for office changes and additions in the 3760 Building.

Pressure Systems

The PRTR Corrosion and Film Loop is now 60 percent complete. During the last outage the core drilling was finished and both shielding plugs were installed. All remaining piping has been prefabricated, and it is estimated that the job can be completed during the ten-day outage scheduled for January.

The four PRTR moderator vessels for ion exchange moderator cleanup are 50 percent complete. The third-party inspector checked materials for the vessels on December 16 and 17 and fabrication inspection on December 28. Especially stringent inspection is being made, since these vessels will be stamped as Class C nuclear vessels. The Nuclear Code is quite new and very restrictive.

Containment Systems Experiment

Chicago Bridge and Iron is progressing with field erection of the containment vessel on schedule. A purchase order has been placed with Struthers Wells Corp. for the Simulator Vessel. Delivery is scheduled for November of 1965. Scope work is complete and design, by Vitro, is being expedited. Further building modifications, equipment procurement, and laboratory work are awaiting design.

Expenditures to date appear in the following order of magnitude:

Design (Vitro)	$ 55 000
Scope, administration, estimating, etc.	30 000
Construction and procurement	260 000*
Total	$345 000

*Excludes Containment of Simulator Vessels which amount to about $390,000 as commitments for FY 1965 and FY 1966.

Facilities Operation

Costs for the month of November are $202,341, which is 105 percent of the forecast for the month.

Waste Disposal

The following table summarizes the Waste Disposal Operation:

Item	October	November
Concrete waste barrels disposed to 300-Wye burial ground	0	0
Concrete waste barrels disposed to 200-W burial ground	6	0
Loadluggers of dry waste disposed to 300-Wye burial ground from 300 area sites other than 325 building	68	21
Loadluggers of dry waste disposed to 300-Wye burial ground from 325 building	23	6
Loadluggers of dry waste disposed to 200-W burial ground from 300 area sites	5	5
Containers of high level dry waste disposed to 300-Wye burial ground waste tanks and tubes	82	60
Crib waste volume, gallons	280 000	290 000

No unusual incidents occurred during the month at 300-Wye burial ground.

C. L. Brown, Specialist, Nuclear Safety, has been informed of the estimated plutonium concentration in the 200-W tubes and will establish a safe working limit. A crucible containing 130.3 grams of Pu and 8.2 kilograms of depleted uranium was disposed to tube #5.

None of the retention basins exceeded the Class II activity levels.

Several instances of tanker drain valve freeze-up were experienced during the recent cold weather. A temporary steam supply for thawing was provided at the 200-W unloading site.

Building Operations

Both alum pumps at the filter plant were recalibrated by 309 maintenance. The two microphotometers have been unreliable in determining turbidity. A study is being made to improve turbidity measurements. Severe winter temperatures (below 0° F) have created freezing problems at the filter plant which require closer operator surveillance. Service has continued uninterruptedly.

All other buildings serviced by this group have had additional problems created by sub-zero weather, and inspection tours have been more frequent. On December 18 outside temperatures dropped to a record low for this date and -12°F was recorded. The power house steam capacity was nearly reached, and some reduction of steam was made by stopping all supply air washers, some vent fans, and some laboratory steam usage. During heavy melting of snow, the number 3 well pit was sandbagged, and the fire department pumped the excess water to prevent equipment damage. Both wells remained in service. Overall inside building temperatures were maintained during this sub-zero weather.

The centrifugal switch on number 1 cold exhaust, 327 Building, failed on December 2, and units were operated manually until repairs were completed.

Following the overhaul of the air sample vacuum pump, 325 Building, December 17 and 18, the circuit breaker coil failed. Delivery of the new coil has been delayed by floods.

The Supplement No. 2 power operator transition was completed on December 28. Six operators from other areas are in training for this operation.

Drafting

The equivalent of 130.8 drawings was produced for an average of 36.6 manhours per sheet. Work continued on essentially the same major designs as described last month.

Construction

	Unexpended Balance
Orders outstanding beginning of month	$736 652
Issued during the month (incl. suppl. & adj.)	138 511
J. A. Jones expenditures during month (includes CO cost)	429 976
Balance at month end	446 187
Orders closed during month	133 398

Maintenance Work Orders active - 3, Face Value - $8,924.

In addition to the major nonproject jobs in progress as listed in the November report, the following were started: (1) reroof 146-FR Building, (2) construct maintenance shop in 2709-E Building, and (3) repair sunken floor in 3745-B which was flooded by melting snow.

INVENTIONS

One invention for the month of December: E. F. Antal - "Tamper-indicating Seals and Safing Bands" (jointly with C. D. Brons, C. H. Gydesen, H. F. Jensen, S. J. Ostrom, and F. B. Quinlan).

W Sale
Manager
Finance and Administration

W Sale:whm

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output for December was 1,105 MWD for an experimental time efficiency of 68% and a plant efficiency of 51%. There were fourteen operating periods, four of which were terminated manually, and ten were terminated by scrams (six of the scrams were because of high differential pressure on the rupture loop). A summary of the fuel irradiation program as of December 31, 1964, follows:

	Al-Pu		UO_2		UO_2-PuO_2		Other		Program Totals	
	No.	MWD	No.	MWD	No.	MWD	No.	MWD	No.	MWD
In-Core	0		6	1717.3	76	14799.4			82	16516.7
Maximum				375.5		413.2				
Average				286.2		194.7				
In Basin	7	572.5	27	3078.9	55	7557.5			89	11208.9
Buried							1	7.3	1	7.3
Chem. Process.	68	5465.8	35	1965.8					103	7431.6
Program Totals	75	6038.3	68	6762.0	131	22356.9	1	7.3	275	35164.5

(Note: MWD/Element x 20 ≈ MWD/TU for UO_2 and UO_2-PuO_2)

Heavy water loss and indicated helium loss for the month were 963 pounds and 128,564 scf., respectively.

Equipment Experience

The final phase of the automatic controller activation was initiated during the month with the period startup control section testing. Three startups were accomplished from subcritical to 2 MW with progressively decreasing periods.

A total of 76 reactor outage hours were charged to repair. Main items were angle valve repacking, HPHC repairs, 5-L leak repair and rupture loop repairs.

Preventive maintenance utilized 513 manhours, or 11.6% of available manhours of assigned craftsmen.

Improvement Work Status (Significant Items)

Work Completed

DC Solenoid Failure Alarm
Containment Valve Additions

Work Partially Completed

Corrosion Loop Installation
Air Lock Door Operators
Modification to PRTR Warehouse 3718-C
Vibration Snubbers for Earthquake Protection
Supplemental Emergency Water Addition
PRTR Steam Utilization
Additional Fuel Storage and Examination Facility
Creep Test Facility
Flux Wire Scanning System
D_2O Cleanup Facility
Alarm Annunciator - High Helium Flow to RLT-1
Pneumatic Irradiation Facility

Design Work Completed

Decontamination Building

Design Work Partially Completed

Instrument Power Supply
PRTR Experimental and Building Facility Addition
PRTR Increased Power Level
"B" Cell Instrument Tubing Penetration

Process Engineering and Reactor Physics

The Zr-0.5 w/o Cobalt wire which was mounted on F.E. #6001 was removed from the element during the December outage. The wire was scanned using a NaI detector and single-channel analyzer and scaler system, to determine the relative flux profile. The peak-to-average flux along the axis of the element, neglecting end effects, was quite low, about (1.12), compared with a predicted value of (1.16). The flux peaking on the ends of the element was quite pronounced, producing maximum-to-average values of 1.41 and 1.87 on the upper and lower ends, respectively. Flux profiles indicated the possibility that the heat flux at the end caps of the 2% enriched short clusters might be exceeding the 650,000 BTU/hour ft^2 limit. Calculations disregarding axial heat transfer indicated a heat flux of 720,000 BTU/hour ft^2, but calculations considering heat transfer through the end cap indicated a heat flux of 480,000 BTU/hour ft^2, well within the limit.

The document, Calculated Reactor Parameters for the High Power Density Core in PRTR, HW-84544, was issued on December 3, together with Appendix I, Power Density and Flux Distributions, Appendix II, Moderator Level Coefficients of Reactivity, was also issued during December. Appendices III and IV are still in preparation.

All scheduled November and December corrosion monitoring items were completed with the exception of moderator shim well inspection and examination of the interior of a shroud tube. Shim well inspection is scheduled for January and shroud tube examination will be completed when the next process tube is removed.

The status of the various test elements at the end of December 1964, is shown below. Those elements discharged prior to December 1, 1964, have been deleted from this table.

Test No.	Channel Location	Element Number	Description	Date Initial Charge	Date Discharged	Approximate Accumulated MWD
14	1956	5097	Swaged - 0.48%	4/2/62		250.8
14	1758	5099	Vipac - 0.48%	5/8/62		260.3
37	1548	1098	UO_2-Physics	5/12/62	12/12/64	255.2
37	1550	1097	UO_2-Physics	5/12/62		281.7
48	1156	5150	0.48% ($\frac{1}{2}$" x $\frac{1}{2}$" Pads)	8/1/62		263.8
54	1542	5116	0.48% (Clip-on pads)	5/8/62		271.0
54	1554	5118	0.48% (Clip-on pads)	5/8/62		413.2
61	1249	5186	0.48% - Physics	5/28/63		284.2
61	1445	5192	0.48% - Physics	6/13/63		272.4
67	1047	5117	0.48% (Repaired wire)	10/20/63		231.6
72	1253	5230	1% (Zr Coupons)	9/1/64		105.1
80	1746	5214	Swaged - 1%	11/18/63		243.4
85	1855	5230	Vipac - 1%	1/30/64		180.6
105	1354	6000	Vipac - 2%, HPD	10/26/64	12/12/64	49.0
107	1255	6001	Vipac - 2%, Flux wire	11/20/64	12/12/64	23.6

Ten irradiated fuel elements were inspected during the month. Eight fuel element rods were shipped to Radiometallurgy for examination.

Twenty-one process tube sections were shipped to Radiometallurgy for testing and evaluation.

Fuel Element Rupture Testing Facility

Replacement of the Rupture Loop Temperature Control Valve trim has improved the temperature control significantly. The small differential pressure variation permitted across the test section by the Process Specifications, however, continues to cause problems.

The irradiation of swage compacted UO_2 fuel element #1030 with a 6-1/2 inch longitudinal slit was completed with the reactor shut down on December 11. The test element had endured 390 hours of reactor operation and had accumulated 7.8 MWD or 127.8 MWD/Ton exposure for a total fuel element exposure of 170.8 MWD. A thorough examination of the element is scheduled in January.

Vibrationally compacted UO_2 fuel element #1067 was defected with a 3-inch slit and irradiation started December 20. The fuel element endured nine reactor scrams from part power and one scram from full power during the month. During this time the fuel element accumulated 166.4 hours operating time and 2.7 MWD or 44.7 MWD/Ton exposure.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 24,557 hours. This includes 17,166 hours performed in the Technical Shops, 4,581 hours assigned to J. A. Jones Company, and 2,810 hours assigned to off-site vendors. Total shop backlog is 17,308 hours, of which 90% is required in the current month with the remaining hours distributed over a three-month period. Overtime worked during the month totaled 737 hours or 3.4% of the total available hours. Distribution of time was as follows:

	Man Hours	% of Total
N Reactor Department	1 845	7.5
Irradiation Processing Department	4 587	18.7
Chemical Processing Department	355	1.4
Hanford Laboratories	17 770	72.4

LABORATORY MAINTENANCE OPERATION

Total productive time was 15,900 hours of 18,000 potentially available. Of the total productive time, 95% was expended in support of Hanford Laboratories components, with the remaining 5% directed toward providing service for other HAPO organizations. Craft overtime worked during December was 330 hours or 1.8% of total available hours. Manpower utilization (in hours) for December was as follows:

A. Shop Work		2 200
B. Maintenance		5 200
1. Preventive Maintenance	1 300	
2. Emergency or Unscheduled Maintenance	1 800	
3. Normal Scheduled Maintenance	2 100	
C. R&D Assistance		8 500

WD Richmond

WD Richmond:bk

INVENTIONS OR DISCOVERIES

All persons engaged in work that might reasonably be expected to result in inventions or discoveries advise that, to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during the period covered by this report except as listed below. Such persons further advise that, for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

INVENTOR	TITLE OF INVENTION OR DISCOVERY
E. T. Merrill and E. A. Eschbach	Multizone Spectral Control Reactor
R. L. Brown and R. G. Chafin	Nondestructive Testing (HWIR-1795)
R. L. Brown	Nondestructive Testing - Improved Penetrant Crack Detection System with Automatic Detection and Readout Features (HWIR-1796)
C. L. Frederick and E. F. Perrizo	A Method of Performing an Ultrasonic Defect Test on Multi-Layer Variable Attenuation Materials Accessible From One Side (HWIR-1797)
C. N. Jackson	The Extension of Nondestructive X-Ray Fluorescence Testing to Small Inaccessible Test Pieces by the Use of a Radioactive Beta Capsule (HWIR-1798)
W. L. Bunch, M. R. Wood, and L. D. Phillipp	Technique for Reactor In-Core Flux Monitoring (HWIR-1789)
W. L. Bunch	Reactor In-Core Neutron Flux Monitoring Method (HWIR-1790)
B. E. Dozer	Self-Calibrating Liquid Level Measuring Device (HWIR-1791)
D. R. Green	Thermal Transducers for Nondestructively Testing Conductors, Semi-Conductors and Insulators (HWIR-1792)

INVENTIONS OR DISCOVERIES (contd)

INVENTOR	TITLE OF INVENTION OR DISCOVERY
R. W. Steffens and M. F. Zeutschel	Capacitor Type Ultrasonic Transducer (HWIR-1801)
N. E. Dixon	A Device for Measuring Fatigue Orientation and Magnitude in Questionable Materials and Components (HWIR-1802)
R. L. Brown	Nondestructive Testing - A Method of Improving the Interpretation of Differential Eddy Current Testers at Small Signal Levels (HWIR-1803)
W. W. Schulz, J. F. Phillips, and R. E. Burns	A Process for the Dissilution of Aluminum-Clad Thorium Dioxide Fuel Elements
A. G. Blasewitz, G. L. Richardson, and M. R. Schwab	Miniature High-Capacity Air Life Circulators for Liquid-Solid Mixing
R. L. Moore and L. A. Bray	A Solvent Extraction Process for the Recovery of Silver
J. J. C. Hsieh, F. P. Hungate, and S. A. Wilson	Technique of Using ZnS:Ag Crystals for High-Speed Alpha Radioautography (HWIR-1786)
P. E. Bramson	A Motor Powered Data Output Tape Reel, December 29, 1964 (Patent applied for)
E. F. Antal, C. D. Brons, C. H. Gydesen, H. F. Jensen, S. J. Ostrom, and F. B. Quinlan	Tamper-Indicating Seals and Safing Bands
G. H. Strong	High Pressure High Heat Transfer Single Cycle Tu e Type Boiling Reactor

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INVENTIONS OR DISCOVERIES (contd)

INVENTOR	TITLE OF INVENTION OR DISCOVERY
G. H. Strong	Expansible Type Fuel Element Which Can Produce Polonium
G. H. Strong	High Temperature Thermal Gas Cooled Breeder Reactor
G. H. Strong	High Temperature Ceramic "Tire and Tube" Fuel Element

R S Paul

for Manager, Hanford Laboratories