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HANFORD LABORATORIES MONTHLY ACTIVITIES REPORT

FEBRUARY 1964

MARCH 16, 1964

HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

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HW-81019

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Compiled by
Section Managers

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Table I - Hanford Laboratories Force Report

Date: February 29, 1964

	At Beginning of Month		At Close of Month		
	Exempt	Salaried	Exempt	Salaried	Total
Chemical Laboratory	147	130	146	130	276
Reactor & Fuels Laboratory	207	197	209	197	406
Physics & Instruments Laboratory	102	75	104	75	179
Biology Laboratory	44	66	44	66	110
Applied Mathematics Operation	21	5	21	5	26
Radiation Protection Operation	41	95	44	97	141
Finance & Administration Operation	149	118	144	117	261
Programming Operation	18	4	21	3	24
Test Reactor & Auxiliaries Operation	65	315	63	311	374
General	4	5	4	5	9
TOTAL	798	1010	800	1006	1806

BUDGET AND COST SUMMARY

February operating cost totaled $2,480,000, a decrease of $413,000 from the previous month; fiscal year-to-date costs are $21,507,000 or 66% of the $32,723,000 control budget. Hanford Laboratories' research and development costs for February, compared with last month and the current control budget, are shown below:

(Dollars in thousands)	COST Current Month	Previous Month	To Date	Budget	% Spent
HL Programs					
02	$ 92	$ 98	$ 650	$ 1 180	55
03	7	8	238	250	95
04	996	1 331	9 379	13 754	68
05	96	109	929	1 403	66
06	243	275	2 140	3 352	64
08	27	28	125	240	52
	1 461	1 849	13 461	20 179	67
Sponsored by					
NRD	135	175	1 284	1 798	71
IPD	25	31	379	490	77
CPD	205	166	1 109	1 784	62
Total	$1 826	$2 221	$16 233	$24 251	67%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels Laboratory

Swelling incurred in prototype N-Reactor fuel elements irradiated to tube average exposure of 1600 and 2000 Mwd/ton has been determined. Maximum swelling observed in the 1600 exposure test was 1.33 vol% in an inner component and 3.44 vol% in an adjacent outer component.

In support of N-Reactor hazard studies the effect of temperature transients has been investigated by annealing an N-Reactor inner fuel component previously irradiated to 2500 Mwd/ton. Portions of the element were induction heated to 980 C for hold times of 0, 5, and 30 min. The resulting incremental volume increases were 5, 8, and 13%.

Postirradiation of eight target elements irradiated 124 days in a KER loop show good irradiation performance.

Additional studies of aluminum extrusion have shown good surfaces produced by the use of an aquadag lubricant on a 400 F billet combined with the use of a solid film lubricant on the die.

In the study of alternate candidate target materials, lithium aluminate has been compacted by swaging to 94% TD.

Metallographic examination of Zircaloy-2 clad rods of U–2 wt% Zr irradiated to 0.25 at.% burnup at temperatures below 600 C show extensive grain boundary tearing.

A thorium fueled thermocouple element has been charged in a KER loop and is operating normally. Damage to a connection between thermocouple and recorder has caused a loss of readings from the thermocouple.

Two N-Reactor inner tube fuel elements of the fluted type have been irradiated to 1800 Mwd/ton and show a volume increase of 0.9%; irradiation is continuing.

Continued destructive examination of clad uranium rods containing intentional clad striations is forming a progressive picture of the deformation of the cladding from the first indications of instability to ultimate rupture. For example, on one specimen in which 50% of the total cladding strain occurred in the striation, necking progressed from both inner and outer surfaces. The Zircaloy-2 in the region of maximum deformation showed severe grain elongation and some void formation.

In the development of a second generation N-Reactor outer support, T section supports are being hot forged from Zircaloy wire. Metallographic examination shows the support is being formed in the beta phase, followed by an extremely rapid die quench.

Corrosion studies are continuing on brazing alloys of varying beryllium content, with and without uranium contamination. Almost every

[redacted]

composition has gone into breakaway corrosion after 3500 hr in pH 7, 360 C water. The same alloy series in a pH 10 environment has accumulated 1400 hr of exposure and also shows breakaway corrosion.

An irradiated N-Reactor inner fuel piece with an unbonded tapered end cap was ruptured in the KRP at 300 C. The current N-Reactor shutdown cooling program was followed after a 62 min incubation period. Examination of the fuel element showed that the end cap had come loose, and the rupture extended 2 in. back from the end cap over the entire circumference.

An electrically heated zirconium-clad device with N-Reactor fuel supports welded to it was exposed 19 days to 288 C, pH 10 (NH_4OH) water at a heat flux of 600,000 Btu/(hr)(sq ft). Virtually no corrosion was found in the crevice under the support. Severe attack had been encountered in similar tests with LiOH.

Crud deposition was monitored in KER-1 Loop during operation with NH_4OH at pH 10 following a decontamination which produced freshly exposed surfaces of carbon steel piping. Thus far, water quality has been good, solids concentrations in water very low, and no crud deposition has been detected with thermocouple elements.

Analysis was continued of laboratory data obtained in heat transfer and fluid flow experiments with the full scale, electrically heated model of the downstream half of an N-Reactor fuel column and outlet fittings. It was found that an outlet riser pressure of 1200 psig, a flow reduction caused by plugging upstream of the fuel column, would not produce flow instability in a maximum power tube.

An in-reactor test was started to evaluate a mixture of sodium nitrite and sodium silicate as a corrosion inhibitor to eliminate Cr^{51} activity from the once-through cooled reactors.

Laboratory experiments and data analysis were concluded for the tests with a full-scale, electrically heated model of a charge of kv

self-supported fuel elements in a smooth-bore K-Reactor Zircaloy tube. It was concluded that the kv fuel elements will require modified heat transfer limits like those specified for the self-supported fuel elements in the over-bore tubes at C-Reactor.

Graphite burnout monitors from B-, C-, and KE-Reactors showed very low burnout rates.

The activation energy for the reaction of CO_2 with H_2 was determined as 52.5 kcal/mole from 762 to 975 C. Specific rate constants were also determined. Dependence of the reaction kinetics on H_2 concentration seemed anomalous in the low pressure region.

The rate of oxidation of TSX by water vapor in helium can be expressed by a function containing a half-power dependence on water vapor concentration.

Swaged UO_2 fuel elements as well as swaged and vibrationally compacted UO_2–0.5 wt% PuO_2 elements have exceeded 5000 Mwd/ton exposure in PRTR. Continuing FERTF tests of a deliberately defected swaged UO_2 fuel assembly show no fuel loss or dimensional changes.

Two UO_2 PRTR fuel elements (1034 and 1099) with broken wire wraps were successfully repaired in the PRTR basin.

Irradiation tests of prototypic EBWR fuel are continuing satisfactorily. Reconditioning and inspection of EBWR fuel rod cladding increased yields by approximately 25%.

Four PRTR elements containing impacted mixed oxides were fabricated by vibrational compaction. Gamma scanning and powder outgassing are being eliminated.

A stainless steel clad, 80 vol% UO_2–stainless steel, 4-rod cluster, complete with stainless steel end caps and spacing brackets, was fabricated by a "one-shot" pneumatic impaction process.

PRTR fuel element design studies were directed toward fuel element rejuvenation concepts. Out-of-reactor loop tests of an eddy current instrumented, PRTR Mark I element demonstrated its pendulum behavior rather than flexing.

Irradiated ThO_2–8.8 wt% PuO_2 pellets revealed an unexpectedly low thermal conductivity, based on irradiation conditions requested from MTR-ETR.

Examination of a section of PRTR pressure tube in TF-7 disclosed that 1/16 x 1/2 in. end bracket supports of a previously tested PRTR fuel element had caused fretting corrosion 10 mils deep. A 360° ring support and 1/4 x 1/2 in. supports had produced no measurable depth of penetration under similar conditions.

Preliminary data from laboratory tests indicate that sulfamic acid solutions have low effectiveness in dissolving PuO_2.

A preliminary study of increasing the specific power in the PRTR by reducing the number of tubes and/or the fuel length was completed. Specific power would be increased by a factor of up to 3.6 and enrichment up to about 2.1%. Although control system strength would be changed, inherent reactor safety would not be appreciably affected.

Assembly of the second generation PRTR shim control rod driving head was completed and the unit is operating satisfactorily in the environmental test facility.

Vibration measurements from the instrumented 19-rod bundle fuel element assembly exposed to prototypical PRTR coolant conditions in the EDEL-I Loop indicate that the fuel element vibrates about 1 mil at a frequency of less than 1 cps. Pressure tube vibration is recorded at 0.2 to 0.5 mil at 20 to 25 cps frequency. This compares to a vibration amplitude of 8 mils at 25 to 30 cps on the inlet connections to Tube 1348 in the PRTR at normal 70 Mw operation.

The first corrosion test of aluminum-clad fuel elements in C-1 Loop, conducted at 290 C outlet temperature, resulted in a very rapid rise in cladding temperature (after a 50 hr induction period) from crud deposition. A second test is being run at 260 C. The cladding temperature has risen slowly for 10 days, but has leveled out since.

Thermal stability studies show that PuN, UN, and a PuN–50 mole% UN solid solution are highly volatile in a 10^{-5} torr vacuum at 1800 C. In contrast, ThN, TiN, and ZrN are very stable under these conditions.

Three capsules containing UN–20 wt% PuN were charged into the MTR this month. Samples of PuO_2 and PuN with burnups of approximately 20 x 10^{20} fissions/cm^3 are being examined in the radiometallurgy facility.

Significant corrosion apparently did not occur during irradiation of a deliberately defected, water cooled ThO_2 fuel element. Gas analyses from nondefected, irradiated elements revealed a high concentration of hydrogen at approximately 3 atm pressure (STP).

High temperature transmission microscopy of UO_2 at 500 C in H_2 shows fracture of UO_2 single crystals resulting from the hydriding of uranium inclusions.

A 10 μ zone of porosity was observed at the surface of a UO_2 single crystal irradiated to an exposure of 5 x 10^{20} fissions/cm^3.

Work has progressed in the development of two new methods of testing–low cycle fatigue of reactor structural materials and a cleavage technique for studying the fracture behavior of materials.

The construction and assembly of a dynamic loading system for fracture studies was completed. This system, which is capable of producing loading rates as high as five million pounds per second, is presently being checked out for performance.

Two creep capsules containing annealed AISI 304 SS have been shipped to the reactor for charging. To obtain control data two

[redacted]

out-of-reactor creep tests have been started at 30,000 psi, one at 500 C (932 F), the other at 600 C (1112 F).

Button-head specimens of Inconel X-750, representing two test heat treatments, are being irradiated in the G-6 position of the ETR during Cycle 61, and others representing a third test heat treatment are being prepared for shipment for Cycle 62.

Recent tensile tests performed by extensometry at 300 C (572 F) on stainless steel specimens yielded results indicating that this high temperature method may be successfully applied to remotely operated elevated temperature tests.

The determination of the functional dependence of strain rate and effective stress and the quantitative calculation of effective stress (the rate controlling stress) is the most important single obstacle to the formulation of a rate theory of deformation. Data from stress relaxation experiments will provide information about this functional relation. Estimates of the effective stress at specific strain rates agree well with estimates made by other methods.

The fast neutron spectrum distribution in N-Reactor was measured using a Li^6 detector. Agreement of the measured spectrum and previously calculated spectrum was very good.

Various gaseous mixtures made up from hydrogen, methane, carbon monoxide, or nitrogen were examined after irradiation in a Hanford reactor at 550 C or in the Co^{60} facility at 30 C.

The long-term EGCR graphite irradiations continue successfully. A capsule was discharged during the report period.

Evidence of a strain rupture limit was revealed in studies of the effects of tensile loading on the irradiation-induced contraction of graphite.

The thermal conductivity of several graphites has been determined over the temperature range -195 to 525 C. A maximum is observed at -10 C.

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The irradiation of boronated graphite samples continues satisfactorily; all temperatures are within acceptable limits.

Three prototype Zircaloy-2 clad Th-U fuel elements have been irradiated to 4600 Mwd/ton at fuel temperatures of 540 C and higher. The elements show a volume increase of only 0.5%; irradiation is continuing.

Previous defect tests of Zircaloy-2 clad Th–2.5 wt% U–1 wt% Zr in 300 C water have shown a tendency to longitudinal splitting of the cladding in as-extruded rods. Specimens heat treated at 750 C for 1 hr or 800 C for 3 hr have now been defect tested. No clad splitting occurred in the 10 hr tests.

A method has been developed for evaluation of Th-U bond strengths by measuring the force required to peel the cladding in a circumferential direction. Forces required to peel the cladding were measured on as-extruded rods. Heat treatment of rods at 800 C for 3 hr strengthened the bond to the point that cladding failure occurred before bond peeling started.

An irradiation test is in progress to investigate the ability of a submicron dispersion of the uranium carbide to reduce swelling. One capsule has achieved an exposure of 0.2 at.% with discharging targeted for 0.3 at.%. Two additional capsules await charging.

Preliminary results from the U^{235}:U^{238} burnup control cases for the Fuel Reuse study show only a 0.2 to 0.3 mill fuel cycle cost variation in the control cases among the four reactor models (HWR, BWR, PWR, and spectral shift) for either Zircaloy and stainless steel cladding, but about 0.5 mill fuel cycle cost difference between the two cladding types in a single reactor type at the same fuel element fabrication and jacketing cost.

The recently completed study of plutonium fueling in fast reactor types similar to the Military Compact Reactor (MCR) illustrated once again that within the size constraints of a U^{235} fueled core, plutonium

will permit greater reactor power or endurance. For a similar power and endurance, shield and core weights can be reduced up to one-third with plutonium fueling.

The Segmented Fast Reactor study has started. The reactor is a sodium cooled, PuO_2-UO_2 fueled, 1000 Mw_e, fast power reactor.

Additional improvements have been made on the bright field etching of alpha plutonium using a nitric acid-alcohol-water electrolytic etchant. There appears to be much promise for use of this etchant to distinguish retained beta from the alpha matrix.

Los Alamos Scientific Laboratory (LASL) electrorefined plutonium has been recast into a slab suitable for rolling. Although the density increased from 19.5013 to 19.5983 g/cm^3, the presence of microcracks is still too extensive to provide optimum rolling characteristics.

The transformation kinetics of LASL electrorefined plutonium was further studied for comparison with high purity, as-reduced and vacuum cast plutonium.

Refinements in the experiments on the creep of the alpha, beta, gamma, and delta phases of plutonium were incorporated in the investigation of transformation creep of plutonium. Results have improved. The relationship between strain rate and stress was examined in detail. The ratio of the steady-state creep of the beta phase to that of the alpha phase at 5200 psi and 150 C (302 F) was determined to be 35.

Sections cut on (111) plane of irradiated and subsequently strained single crystals of molybdenum show channels intersecting with each other at an angle of 60°. The channels correspond to the traces of (110) planes on the plane of the sections. These results are consistent with observations of irradiated and deformed polycrystalline foils of molybdenum.

Seeded molybdenum single crystals have been prepared by electron beam zone-melting of sintered 1/8 in. diameter polycrystalline molybdenum rods.

[redacted]

Fabrication of a system for introducing controlled amounts of carbon into ultrahigh purity molybdenum crystals has begun. The technique employs removal of interstitial elements with a purified hydrogen gas stream and carburizations with a calibrated CO-CO_2 mixture.

Quenching experiments are being conducted on polycrystalline nickel foils of various purities. Significant increases in yield stress were observed after quenched specimens were aged at temperatures of 500 and 700 K (932 and 1292 F).

A 4 ft, 1-5/8 in. diameter HTLTR prototype graphite heater element has operated satisfactorily for over 500 hr at temperatures in excess of 900 C, including operation between 1400 to 1700 C for over 320 hr.

More than 150 EBWR rods (a total to date of approximately 600) were fabricated for physical tests.

One ton of aluminum-clad thoria fuel elements was charged into the F-Reactor fringe on January 27, 1964, for a planned exposure period of 60 to 65 days. Fabrication of an additional ton of thoria fuel was completed. A total quantity of 4 tons is required for the initial large scale pilot run.

In the fabrication of fuel shapes by high energy impaction of cermet over steel mandrels, bonding of the cermet to the mandrel was completely prevented by autoclaving an oxide coating to the steel before impaction.

A W–20 vol% UO_2 cermet grid shows no apparent breakdown in grid structure after 10 cycles in an arc plasma. Temperature differences across the face of the grid were greater than 2000 C.

Temperature cycling of UO_2-W cermets leads to serious UO_2 loss by a mechanism causing initial migration of UO_2 into large tungsten grain boundaries, followed by rapid UO_2 vaporization.

A demonstration extrusion of a "plate in shell" tungsten geometry has been evaluated showing good bonding between the plates and shell. Grain growth during preheat is considered the cause of interface roughness observed in the extrusion.

[redacted]

In the development of submicron UO_2 particles in tungsten matrices, uranium and tungsten were coprecipitated to produce a product containing 10% by volume UO_2. Evaluation of the product is in progress.

Fabrication of irradiation samples for the Phillips Petroleum Company was started. These elements consist of aluminum-clad, Au-Al and Li-Al alloys for reactor physics experiments.

Fabrication of crud detector elements for Coolant Systems Development was completed. The elements are Mo-UO_2 cermet pellets in stainless steel cans.

In cooperation with Process Engineering Development, the effect of heating rate and temperature during the calcination of sol-gel ThO_2 was investigated. Dense, unshattered specimens are obtained by slow heating to 1150 C.

2. Physics and Instruments

Measurements on the N-Reactor lattice are under way in the PCTR to study the change in k_∞ between the PCTR N-mockup and the as-built reactor. Measurements were also taken to determine the flux ratios to be expected in special fuel and poison columns as compared to regular fuel columns.

Instrument work in support of N-Reactor startup tests included: procurement of special low noise instrument cables, completion of a system for recording the temperature of thimbles for the neutron detectors, modification of a process monitor to provide a sensitive neutron flux monitor for startup tests, and further development of a low-level neutron monitoring system.

The N-Reactor secondary coolant system simulation is being analyzed on the new computer. The system includes a steam generator, dump condenser, surge tank, and piping.

A solid-state portable gamma spectrometer that detects the presence of plutonium in hoods and glove boxes by sensing 384 kev photons has been completed. Other efforts are being made to obtain beneficial use of electronic circuitry suitable for high temperature and high radiation field applications in chemical processing

Criticality experiments with plutonium systems simulating precipitates and polymers (H:Pu atomic ratio of ~15) determined the critical thickness of a slab of essentially infinite extent. The effects of plastic neutron reflectors, with and without cadmium next to the core, were also determined.

Computer techniques were developed for calculating the criticality of enriched uranium rods in uranium solution, a separations problem for which little data exist.

Two documents were issued reporting the results of a nuclear safety review of the N-Reactor fuels process in the 333 Building.

The reason for certain systematic discrepancies in previous measurements of the scattering law for 95 C H_2O has been tentatively identified as order contamination in the monochromatic beam for some experimental conditions.

A series of 3 to 15 Mev total cross-section measurements was completed for 18 samples.

The recently completed 400-channel analyzer was successfully connected to the Vernier chronotron at the Positive Ion Accelerator Facility for neutron time-of-flight measurements. The equipment automatically records data on punched tape in a form suitable for direct processing by the 7090 computer.

The Wilkins, Wigner-Wilkins, and Nelkin models for calculating flux spectra were compared for a highly loaded plutonium-fueled Phoenix core. While the calculated multiplication constants differed appreciably, the isotope variations with burnup were similar because of compensating errors, and because the spectra are highly epithermal.

Data obtained in the D_2O-moderated PRCF experiments continue to be analyzed. A value for β/ℓ of 6.3×10^{-6} sec^{-1} has been obtained from the noise spectrum measurements. The moderator level coefficient measurements are being interpreted. Plans continue to be made for the PRCF H_2O experiments using EBWR fuel. These experiments should be started in March. The approach-to-critical experiment with a 0.71 in. lattice spacing using the EBWR fuel was completed in the Approach-to-Critical Tank.

Theoretical physics development work on fuel cycles progressed through formulation and programming of an isotopic transmutation analysis based upon one-dimensional multienergy transport theory. This code will be particularly useful in fast reactor studies.

Other code development work included: the revision of TEMPEST to include routines for solving the modified heavy gas equation; writing of a program, TRANS, for determining the amplitude and phase shift of the zero power transfer function; preparing of a special version of TRIP for HTLTR studies; modifiying the operating strategy of ZODIAC to improve running efficiency; continuing updating the RBU Basic Library; and checking out RBU code using the PRTR 19-rod cluster and writing documentation for the code.

Burnup analyses were started on the second H_x Pu-Al and the first 1 wt% PuO_2-UO_2 PRTR elements. The modified PRTR Gamma Scan is being checked using irradiated rods from PRTR elements 5109 and 5103.

Plans are underway for general improvement of the PRTR fuel rupture monitor. Detectors, scanners, and electronic circuitry are being reviewed and modified. A new scanning system is being developed for prototypical operation.

Measurements made on the PRTR fuel element underwater gamma scanner with modified collimator show a 30-fold improvement in signal-to-noise ratio.

[redacted]

An analysis of the hazards associated with including special fuel elements (e.g., Pu-Al and UO_2) in the EBWR core has been completed, and the results communicated to personnel at Argonne National Laboratory.

The fourth in a series of high temperature tests of possible HTLTR materials in nitrogen was shut down after 75 hr of operation at 1200 C because of a leak in the gas outlet line caused by what appeared to be corrosives from samples of insulating brick. A detailed examination of the samples is in progress. A 4-1/2 ft graphite heater element, 1-5/8 in. in diameter, successfully operated for 500 hr (equivalent to 6 mo of HTLTR operation) in the temperature range from 900 to 1600 C in a nitrogen atmosphere. The test was shut down so that the element could be examined at this stage.

Work also continued on the design of the HTLTR mechanical oscillators, and a document was written on the proposed HTLTR neutron chopper.

Analysis of mass spectrometer data indicates the sensitivity of regenerating neutron flux monitors deviates from calculated values by 5%. The data show a skewness which makes it unlikely that a consistent set of neutron parameters can be found. In essence, the experimental-calculational techniques developed for optimizing detector composition are adequate for engineering use, but are limited in accuracy by the spectral assumptions in the use of Westcott cross sections. Two experimental regenerating detectors are being fabricated offsite.

The B^{11} neutron flux detector concept was experimentally demonstrated with reactor exposure of a unit employing evacuated connecting cables. Vibration induced by water turbulence put this unit out of commission after a few minutes' operation; a more rigid assembly is being constructed.

Faborable results are being obtained with nondestructive testing eddy current coils mounted in a direction perpendicular to the work piece surface.

This arrangement improves resolution by effectively limiting the surface area linked by the electromagnetic field.

Work is proceeding toward practical application of the multiparameter eddy current test to reactor process tubes. Efforts are currently being made to adopt special electronic adjustments which incorporate three degrees of freedom.

Work has started on the nondestructive inspection of isotope heat source capsules. The wall thickness of several experimental capsules was ultrasonically measured with favorable results.

Sheep body temperature has been successfully measured by implanted sensors that telemeter the data to a remote station. Accuracies of ±0.25 C are achieved. Continuing efforts are being made to sense and transmit animal pulse and respiration data in similar fashion.

The results of the whole body counting at Anaktuvuk Pass last month became available and showed decreases of Cs^{137} in the Eskimos averaging 30%. Urine samples from this village showed the presence of a new isotope, Na^{22}, in the people.

A new accelerator tube made of glass with higher voltage-breakdown strength appears to have solved most of the difficulties that have been appearing in the Van de Graaff operation for almost 2 yr.

Further investigation of the Area-Dosage relationship, using all Hanford atmospheric dispersion data obtained between 1959 and 1963, progressed during the month. These data extend over a greater range of atmospheric stability classifications than the earlier stable case studies.

3. Chemistry

A study was completed that incorporated both field data and theoretical evaluations to define the ground-water travel time from the 1301-N Crib to the Columbia River. Results indicate the actual minimum travel time will be about 96 days.

Electromotive force measurements as a function of time, made on various uranium specimens in $NaClO_4$ solution, have revealed surface differences between ingot and dingot uranium. Development of this technique may yield a method to distinguish the quality of uranium coating surfaces.

Laboratory tests of a sample of straight chain hydrocarbon produced by South Hampton Refineries have shown it to possess excellent properties for use as a Purex-type diluent.

Laboratory tests show that sodium sulfate-ferrous sulfamate solutions are satisfactory reductants for neptunium and plutonium in TLA systems and do not have the potential for generating ammonia that exists with hydroxylamine sulfate reductants. Tests of sodium formaldehyde sulfoxalate as a plutonium reducing agent continue to show promise.

Hanford capability to manufacture U^{233} very low in U^{232} (ca. 1 ppm) was demonstrated with successful isolation of the first sample of Hanford-produced, high-isotopic purity U^{233}. The product, slightly less than one-half gram of UO_3, contains only 1.3 ppm of U^{232} and gives a radiation dosage much less than 1 mr/hr.

A survey was made of potential polonium extractants. Promise was shown by xylene solutions of DBBP, Aliquot-336, TOA, and diethoxybutane.

An investigation into the possible use of zeolites in neutron irradiation applications was begun. Five zeolites will be used initially in a study concerned with the sodium-uranyl ion system.

The run plan is nearly complete for purification of the kilogram quantity of CPD-recovered Tc^{99}. The run, to be carried out in equipment installed in the contamination control zone of 325-A Building, is targeted to start in mid-March.

Technology for a second generation in-tank CPD waste solidification unit is being developed; evaporation will be effected by immersed electrical heaters.

In the zeolite sorption process for packaging Sr^{90} and Cs^{137}, helium is used to improve thermal conductivities. Full scale tests show an appreciable helium loss if the container is not quickly sealed.

A novel, single beam, filter photometer has been devised that shows promise; it may directly measure the uranium concentration in solvent regardless of the degradation of the solvent.

Scouting studies of the use of molten chloride salt media for U^{233}-Th separations have been started. ThO_2 dissolution rates (the reaction is slow) and the selective electrodeposition of UO_2 from Th-containing melts will be observed.

Operations of the C-Cell Salt Cycle Process with irradiated fuels were started. Decladding, dissolution, and electrolysis steps were concluded; the data are under analysis.

Electrolysis in a chloride melt has been used to prepare a W-UO_2 cermet containing 37 mole% tungsten metal.

Brookhaven continuous glass-making equipment arrived. Preparations are under way for hot-cell installation and testing.

Appreciable difficulty was experienced in attempts to spray calcine a simulated Redox feed with silica, borax, and sugar additives. A plugging problem appears to be associated with the borax.

The electrodialysis unit was successfully operated for extended periods with Purex acid condensate in a study aimed at radioruthenium decontamination.

Travel times for Columbia River water have been calculated from observations of the rate of movement of I^{131} discharged to the river following a fuel element rupture. It required 14.6 days to reach Vancouver, a distance of 259 miles.

A mathematical model of the Hanford ground-water flow system is to be built rather than the electrical analog previously planned. This change

in approach resulted from cost evaluations, the broader applicability of the mathematical model (evaluated by digital methods), the success achieved in development of the Hanford "steady-state" flow program, and (most significantly) the recent advances made in devising techniques for solving large systems of simultaneous algebraic equations (diakoptics).

A comparison of the Cs^{137}, Cs^{134}, Na^{22}, and Zn^{65} levels in urine samples of local origin with a sample from Alaskan Eskimos showed that the Eskimo sample contained 50 times as much Cs^{137}, Cs^{134}, and Na^{22}, but only one-half as much Zn^{65}. These numbers reflect the fallout origin of the four isotopes, and also the presence of Zn^{65} in local drinking water.

Arrangements were completed to perform uranium-daughter inhalation measurements on uranium miners at the Union Carbide mines in Colorado in April.

In tests of the Tracerlab low temperature asher, which uses atomic oxygen, 1 g rods of graphite were almost completely oxidized in 6 hr without disruption of the rod form of the remaining gray ash.

A cell counting and sizing system using a Coulter counter and an RCL 256-channel, pulse-height analyzer has been adjusted to measure blood cell volumes in suspensions with a precision of about ±10%. The equipment will be used in the investigation of chemical protective agents for the living cell and, in addition, should yield considerable information on the nature of the radiation-induced, cell-lysing process itself.

Treatment of charcoal with bromine vapor was shown to aid the retention of methyl iodide, a form of radioiodine found in certain process off-gas streams.

Pneumatic compactions have successfully produced good compacts of SrF_2 and Sr_2TiO_4. An attempt to prepare Ce_2O_3 met with partial success and will be repeated. Several potentially interesting heat source materials and chemical intermediates were prepared--SrS and a range of high-cesium borate "compounds," e.g., Cs_3BO_3, $Cs_2B_4O_7$, etc.

[redacted]

4. Biology

The lack of correlation between incidence of columnaris organisms and mortality in trough-held Rainbow trout appears to be related to an as yet incompletely defined cycling state of infection, recovery, refractory period, re-infection, etc.

Chronic feeding of P^{32} to cichlids over a 2 yr period resulted in depressed testes growth at the 1.0 μc P^{32}/g food level. No effect was noted at a feeding level of 0.25 μc/g of food.

Pigs on a 625 μc Sr^{90}/day feeding level showed a typical infiltration of myeloid cells in certain soft tissues, particularly kidney and heart muscle. These lesions were suggestive of myeloma formation that may be a significant indication of early Sr^{90} effect.

Pigs exposed to a constant Sr^{90} intake from conception accumulated a skeletal burden equivalent to approximately 10 times the daily intake, with a uniform distribution of the Sr^{90} throughout the skeleton. When Sr^{90} feeding was started at age 9 mo, the skeletal accumulation at 1 yr was about three times the daily intake, and distribution throughout the skeleton was quite nonuniform--the skull exhibiting about half the average skeletal Sr^{90} concentration and the thoracic vertebral body exhibiting about twice the average.

Eight pigs whose skin was exposed at 46 sites to Ru^{106} particles nine years ago (200 to 19,000 μc hours per site) showed no changes suggestive of neoplasia.

Pigs injected intradermally two years ago with Pu^{239} nitrate at levels of 0.0016 to 5.0 μc/site were re-examined. At none of the injection levels were all sites visible. Analyses on one of these animals indicated retention at the injection sites of about 1% of the total administered dose. About 9% was translocated to the liver and about 7% to the skeleton. Highest concentrations were observed in regional lymph nodes which showed extensive damage. Considering that a large amount of the

administered plutonium was lost externally during the first 2 wk after injection, these findings indicate a very substantial translocation to internal organs.

Liver homogenates prepared from female rats injected 18 hr before sacrifice with toxic doses of Np^{237} showed a markedly decreased ability to oxidize C^{14}-labeled palmitate.

Using intestinal perfusion and bile duct cannulation techniques in rats, it was shown that the chelating agents DTPA and TTHA when administered orally are absorbed almost exclusively from the jejunal segment of the small intestine.

Mice showed only a small increase in the leakage of polyvinylpyrollidone into the intestinal tract followed irradiation. This is in contrast to the much larger effects observed in rats.

Unirradiated mice injected with rat bone marrow or rat spleen cells showed a marked leukocytic infiltration of the liver within 1 to 3 days. In irradiated mice this leukocytic infiltration does not occur until about 3 wk posttreatment. This observation, if confirmed, should provide a valuable tool for the study of graft-host response in radiation chimeras.

Following inhalation exposure to uranium ore dust, rat lungs showed a varying ratio of Th^{230} to U^{238}, increasing from an average value of 1.5 at 1 wk postexposure to a value of 2.7 at 8 wk postexposure.

Analysis of more than 100 white fish collected at Priest Rapids in January showed that approximately 80% of these fish had been exposed to reactor effluent, as indicated by their above-background content of Zn^{65}.

A study of soil profiles beneath greasewood shrubs indicated that flow of rain water from the branches and down the stems was not an important factor in concentrating moisture beneath the shrubs. Highest pH values were noted on the east and northeast sides of the shrubs under areas of greatest leaf litter accumulation, supporting the theory that leaf litter is the significant factor in increasing the sodium content of soil in the vicinity of these shrubs.

5. Programming

Power-conversion for the small Hanford production reactors (including multiproduct operation) is being examined on the basis of once-through water coolant, and without overboring the graphite channels. Thermal-to-electrical efficiencies of up to 20% appear possible, and preliminary fuel cycle costs appear satisfactory. Situations examined require relatively low capital costs, compared with overboring and installation of pressurized recirculating coolant.

Initial analysis of the possible conservation of U^{235} with spike-enriched-uranium-fueled-thermal reactors vs. uniformly enriched-thermal reactors indicates that the uniformly enriched systems generally achieve slightly higher U^{235} utilization as measured by Mwd of heat-per-ton of natural uranium used.

Tentative recent physics data for U^{233} suggest that thermal breeding with U^{233} in Th^{232} may be limited to soft-spectrum reactors such as large D_2O or graphite moderated machines.

TECHNICAL AND OTHER SERVICES

AEC-RLOO issued a request for expressions of interest from commercial firms desiring to perform radiation protection services (e.g., the processing of film badges, neutron badges, and finger rings; the collection and laboratory analysis of bioassay samples; and the laboratory analysis of offsite environmental samples). A preproposal conference for interested firms is scheduled for late April.

There were four new plutonium cases confirmed by special bioassay analysis during the month. All cases were estimated to be less than 1% of the maximum permissible body burden (MPBB, plutonium with bone as reference, is 0.04 μc). Three cases resulted from inhalation during fires in the Weapons Manufacturing Facilities (234-5 and 232-Z), and the fourth case resulted from absorption through the skin when an employee received plutonium nitrate contamination of 20,000 dis/min while working in the Purex Processing Facility (202-A).

The total number of individuals who have received internal plutonium depositions at Hanford is 338. With the termination of one employee deposition case there are currently 245 employed.

Consulting work on economic-biological interactions for the Division of Biology and Medicine was started.

An appropriate discriminant model was developed to obtain confidence intervals for the cladding thickness of standard N-Reactor fuels.

A set of sequential sampling plans was developed for use in testing a new form of resistance temperature probe.

A number of analyses were performed in connection with hot-die-sizing experiments for fuel element production.

Discussions were held regarding future work required to construct a model of the Z-Plant plutonium balance system for accountability purposes.

Continued assistance was provided on formulating and fitting mathematical models to the problem of radionuclide retention in reactor effluent streams.

The first of the three phases of the computational programming of the "shoe box" model of ground water flow has been completed and is functioning.

A mathematical study was completed describing sources and magnitude of errors which could result from possible misalignment of gears on a rotary contour gage.

The main calculation program of the IRA system was modified to permit the use of several different models in fitting gamma ray spectrometer data.

The statistical analysis was completed on an analysis of beetle populations in sagebrush and greasewood communities.

SUPPORTING FUNCTIONS

PRTR output for February was 1224 Mwd, for an experimental time efficiency of 77% and a plant efficiency of 60%. There were 16 operating periods during the month, 5 of which were terminated manually and 11 were terminated by scrams. A summary of the fuel irradiation program as of February 29, 1964, follows:

	Al-Pu		UO_2		PuO_2-UO_2		Other		Program Totals	
	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd
In-Core	0		7	1 199.3	78	10 173.2			85	11 372.5
Maximum				264.6		240.1				
Average				171.3		130.4				
In Basin	43	3 726.4	25	2 754.9	27	1 454.1			95	7 935.4
Chemical Processing	32	2 309.3	35	1 965.8					67	4 275.1
Buried	—	—	—	—	—	—	1	7.3	1	7.3
Program Totals	75	6 035.7	67	5 920.0	105	11 627.3	1	7.3	248	23 590.3

Note: (Mwd/Element) x 20 ~Mwd/ton_U for UO_2 and PuO_2-UO_2.

Heavy water and indicated helium losses for February were 1588 lb and 144,623 scf, respectively.

A potted electrical penetration of the containment vessel deteriorated since the last inspection. The reactor was shut down and the potting compound replaced. This completes the potting replacement of those penetrations installed during the same time period and noted to be of a different hardness from the original installations.

Startup work for the H_2O moderated PRCF continued through month end. Estimates of fuel availability continued to dominate scheduling of experiments.

FERTF Test No. 8, Irradiation of an Intentionally Defected UO_2 Fuel Element, was started on February 2, 1964, and continued through month end. During the month, the fuel element was cycled through 16

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critical periods and gained about 8.3 Mwd exposure. No increase in fission gas release was observed which might indicate propagation of the failure area.

Shipment of Al-Pu fuel elements to Redox for chemical processing was begun.

Total Technical Shops Operation productive time for the period was 17,098 hr. Distribution of time was as follows:

	Man hr	% of Total
N-Reactor Department	2 731	16.0
Irradiation Processing Department	3 324	19.4
Chemical Processing Department	413	2.4
Hanford Laboratories	10 630	62.2
Hanford Utilities and Purchasing Department	--	--

Total productive time for Laboratory Maintenance Operation was 17,900 hr of 19,400 hr potentially available. Of the total productive time, 87.7% was expended in support of Hanford Laboratories components, with remaining 12.3% directed toward providing service for other HAPO organizations. Manpower utilization (in hours) for February was as follows:

A. Shop Work		1 500
B. Maintenance		7 300
1. Preventive Maintenance	2 000	
2. Emergency or Unscheduled Maintenance	1 500	
3. Normal Scheduled Maintenance	3 800	
C. R&D Assistance		9 100

Cumulative data of Hanford visits:

	Number of Visitors	
	In February	Since June 13, 1962
Visitors Center	1 338	62 657
Plant Tours	237	n.a.

The heavy water inventory at the end of February 1964 showed a loss of 1589 lb valued at $21,817 for the PRTR. Heavy water scrap generated during the month amounted to 3290 lb, resulting in a $4244 charge to operating cost. Heavy water accumulated at February 29, for return to SROO, amounted to 14,298 lb valued at $177,856.

HAPO professional recruiting activity this month:

	Plant Visits	Offers Extended	Acceptances Received	Rejections Received	Open Offers at Month End
Ph.D.	7	2	1	3	2
BS/MS (Direct Placement)		0	1	0	0
BS/MS (Program)		40	1	9	57

Seven technical graduates were placed on permanent assignment. Two new members were added to the roll and none terminated, bringing the program total to 62 at month end.

An agreement was reached with CBC and the American Society of Metals to offer Metals Engineering Institute courses in CBC's Extended Day Program begining with the spring quarter.

Press releases on the recovery of Pm^{147} and Hanford Laboratories' newly acquired analog computer were prepared and distributed.

One hundred fifty-six students and teachers participated in the February 11, Edison Day program at HAPO.

A survey of classification and related security practices in the Laboratories, performed at the request of AEC-RLOO, was completed and reported to the Commission.

Authorized funds for eight active projects total $7,443,500. The total estimated cost of these projects is $10,650,000. Expenditures on them through January 31, 1964 were $2,051,000.

[redacted]
Manager, Hanford Laboratories

HM Parker:JEB:esl

REACTOR AND FUELS LABORATORY MONTHLY REPORT

FEBRUARY 1964

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 02 PROGRAM

1. Metallic Fuel Development

N-Reactor Fuel Evaluation. In support of N-Reactor hazard studies the effect of temperature transients of various durations have been investigated by annealing experiments on an intact N-inner element irradiated to 2500 MWD/T. Isolated zones along the length of the element were induction-heated to 980 C for times at temperature of 0, 5, and 30 minutes. Fuel volume increases of 5, 8, and 13%, respectively, were measured for the three zones. These volume increases are about a third of those previously obtained on specimens cut from an irradiated element, presumably because the restraint of the geometry and the clad were lost in the case of the cut specimens. Final testing and evaluation of the performance and behavior of full length (24") testing of both N-inner and outer tubes at 980 C will be accomplished next month.

The irradiation-induced swelling in the inner and outer components of two or more KER loop charges of NAE's have been determined. The charges were irradiated to tube average exposures of approximately 1600 and 2000 MWD/T at prototypic temperature and 1600 psi coolant pressure. Maximum fuel swelling observed in the 1600 exposure test was 0.85 vol% in an inner and 2.50 vol% in the corresponding outer component. Maximum fuel swelling in the 2000 MWD/T exposure test was 1.33 vol% in an inner component and 3.64 vol% in an adjacent outer component. Observed swelling in some of the outer components of both charges is more than predicted by the previously developed empirical expression. It appears that there is less restraint offered by the outer tube geometry and more offered by the inner tube geometry than allowed for in the developed expression.

Target Element Development. Post-irradiation examination of the eight target elements which were irradiated for 124 days in a KER loop has continued in Radiometallurgy. The single phase lithium-aluminum alloys continue to show good irradiation performance as judged by visual examination. In addition, measurements of the outside diameters of the eight Zircaloy-4 cans were in agreement

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with pre-test values. One of the target elements was selected for extensive examination. The OD's of the Al-can and the Li-Al target core again agreed with pre-test values. The core of this element is presently being sectioned for further analyses.

Four additional target elements are being irradiated in KER Loop 2 to evaluate the performance of two-phase lithium-aluminum target alloys (nominal 3.2 and 8.0 wt% natural Li) under simulated N-Reactor conditions.

Lithium-Aluminum Alloy Corrosion. Corrosion testing of aluminum alloys with up to 3.5 wt% lithium is continuing. The tests are being conducted at 100 C in pH 4.5, pH 7, and pH 10 water, and in a steel autoclave at 200 C in pH 7 water. Weight gain data to date indicate that the corrosion rate of aluminum is slowed by additions of lithium up to 1.5 wt%; by lowering the pH and temperature; by annealing; and by the lack of galvanic action. Additions of lithium above 1.5 wt% seem to increase the corrosion rate drastically.

Aluminum Extrusion Studies. Additional billets of 1100 aluminum were extruded to determine a reliable extrusion process which will produce a good surface on coextrusions of aluminum clad materials. Previously, best results were obtained by chemically plating copper on the surface of the billets and using Fiske 604 (a commercial lubricant consisting of a suspension of aluminum fines) as an extrusion die lubricant.

A number of billets, part of which were plated with copper, were coated with "Aquadag Lubricant", first preheating the billets to 300-400 F and dipping the billet into and then spraying the billet with a thin Aquadag solution (thinned 15 to 1 with water) to produce a tough adherent Aquadag film on the billet. Results indicated that the Aquadag film is nearly as good as the copper plate.

The condition of the die is very important in preventing light die pickup when using cone dies for coextrusion. A number of dies were treated with a commercial solid film lubricant to compare with oxidized dies. The endurance of the lubricant is about the same as the oxide layer; however, the die can be retreated in about one-half the time it takes to reoxidize the die.

Higher extrusion speeds, with die exit speeds of about 1000 to 1200 inches per minute, have been found to result in a slightly better extrusion surface.

A means of compacting lithium aluminate or lithium aluminate plus aluminum powder has been developed. The latest swaging attempt has produced compactions equal to 94% of theoretical density. The final product is tightly encased in an aluminum jacket and can be produced with or without integral aluminum end closures.

Uranium and Uranium Alloy Irradiations. Uranium and uranium alloy rods coextrusion clad with Zr-2 are being irradiated in NaK capsules in a program to evaluate their swelling performance as a function of composition, heat treatment, irradiation temperature, burnup, and restraint.

Additional experimental data have le(to a reappraisal of the swelling performance of Zr-2 clad U - 2 wt% Zr alloy. The previous data indicated that the fission gas-induced swelling of the alloyed uranium was two to three times that of unalloyed uranium at an equivalent irradiation temperature. The refined data now show the swelling to be about the same for the alloyed and unalloyed fuels at equal temperatures. It is possible now, from recalculations of temperature distributions to relate the microstructures observed in three rods of irradiated U - 2 wt% Zr fuel to that expected from the U-Zr equilibrium phase diagram. In the central region of these fuel rods a second phase which was previously thought to be delta phase is now shown to be retained gamma-1. These new results will require a re-evaluation of the effects of zirconium additions on the swelling of metallic uranium.

Uranium and Uranium Alloy Irradiations. Metallographic examination of another rod of U - 2 wt% Zr irradiated to about 0.25 at% burnup in the alpha plus delta temperature range (less than 600 C) shows extensive grain boundary tearing. The tearing is observed in a band approximately 0.060" wide at the outer surface of the fuel. Also observed in the center of this rod is a core, approximately 0.3" diameter in which large irregularly-shaped pores are aligned perpendicular to the major axis of many of the twins. The appearance of this microstructure is similar to that observed in unrestrained uranium cylinders irradiated in the Basic Swelling Program by the Hanford Physical Metallurgy Operation. This is the first evidence of grain boundary tearing and extensive aligned porosity in Zr-2 clad fuel rods.

Fuel Fouling Detector. Two thorium-fueled thermocoupled elements were charged in KER Loops 2 and 3, respectively. During pressure testing the pressurizer on Loop 4 was damaged and the elements in that loop were transferred to Loop 3, replacing the thermocouple train. The element in Loop 2 continued to operate efficiently

during initial startup procedures, but just prior to startup the connections between the thermocouple and recorder were damaged causing a loss of readings from the element. The element is presumed to be undamaged, however, and has been under irradiation for one week with no indication of failure. It will be necessary to wait until reactor shutdown to determine the nature of the damage to the thermocouple leads and to effect repairs.

Fabrication of Crud Probe Stock. A long life-constant power heat source will be fabricated from a Th - 1½ wt% (fully enriched) U - 1 wt% Zr alloy for the purpose of studying the variation in heat transfer with film buildup on a Zircaloy-2 surface during irradiation. Two 65-pound ingots have been primary melted and will be extruded into electrodes for remelting. Primary melting was done using 3600-3800 amperes, 27-28 volts at a vacuum of 16 micron.

Fluted Fuel Element Irradiation. Weight measurements made on two N-single tube fluted elements after nine cycles of irradiation in the ETR M-3 loop indicate that they have undergone a volume increase of 0.9%. The elements have been weighed after each reactor cycle, which represent exposure increments of about 200 MWD/T. During the last (ninth) reactor cycle no measurable volume increase occurred. The elements now have a total exposure of about 1800 MWD/T and operate with a maximum core temperature of 520 C. The irradiation will continue.

Cladding Deformation Studies. Post-irradiation examination continued on Zircaloy-2 clad uranium rods from capsule irradiations to determine the strain capabilities of Zircaloy-2 cladding as a function of cladding thickness, cladding thickness uniformity, and temperature.

Metallography has been completed on the cladding of four rod specimens with nominal cladding thicknesses of 0.025-inch and 0.008-inch deep 1/8-inch radius intentional striations on the inside surface of the cladding. Observations from these samples form a progressive picture of the deformation of the cladding from the first indications of plastic instability to ultimate rupture.

A longitudinal indent on the cladding outer surface is the first visible indication of the onset of plastic instability. At this point the thinning or necking of the cladding is occurring from the outside surface. Strain in the width of the striation amounts to approximately 15% of the total cladding strain. This was observed on two of the rods examined. On these, the ductility of

the cladding is high - as indicated by the 30% reduction in thickness in the striation. The effects on the Zircaloy-2 microstructure were confined to a narrow band in the striation near the inner surface of the cladding. On the third specimen, which had a pronounced indent in the cladding, 50% of the total cladding strain occurred in the striation, and necking progressed from both the inner and outer surfaces. Grain elongation was severe in this sample and some voids appeared in the Zircaloy-2 at the region of maximum deformation. On the fourth sample the cladding ruptured after approximately 55% of the total cladding strain had occurred in the striation. Necking obviously progressed in this sample from the inner and outer surfaces of the cladding. Grain deformation at the fracture was severe, but voids were not detected in the microstructure. From these four samples it is apparent that the Zircaloy-2 has retained good ductility, but because of the reduced cross section in the striation and the influence of radiation damage the cladding is unable to support the load imposed by the swelling uranium and necking results.

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

The comparative irradiation behavior of single tube fuel elements (1.790-inch OD x 0.975-inch ID) fabricated from standard N-fuel composition and an alloy containing 400 ppm Fe and 800 ppm Al will be determined from high temperature loop irradiations. Structural characterization of the ingots, primary extrusions, coextruded fuel, and completed fuel elements have been made. Hot hardness tests have been made on samples of the primary extrusions and finished fuels with the creep penetration apparatus at 300 and 400 C over a range of loads. Hardness increases occur in both alloys due to coextrusion and beta heat treatment that are primarily due to changes in second phase particle distribution. No differences have been observed between the two alloys in the strain hardening exponent determined from the variable load tests. Limited creep data obtained at constant load and temperature for the two alloys indicate no difference in creep rate at 400 C.

Second Generation N-Outer Support. Analysis work has been carried forward since the initial fabrication demonstration of this support shape. Metallographic examination shows the support is being formed in the beta phase and followed by an extremely rapid die quench. Three heat treatments have been given the formed parts to determine their influence on the permanent set characteristics of the support. A 72-hour autoclave treatment in 400 C steam produced no evidence

of immediate corrosion inadequacy. Supports were also subjected to a stress relief anneal at 600 C, and an alpha anneal of 750 C. In both cases parts were taken to temperature and cooled in vacuum. Ten samples from each heat treatment were dead weight tested without shoes to determine the permanent set behavior. The 750 C alpha anneal increased deflection under load and permanent sets up to three times the as-formed condition. Autoclaving had no measurable effect.

Corrosion of Braze Alloys. Corrosion studies have continued on the various brazing alloys in both pH 10 and pH 7 environments at 360 C. The first group of alloys, Zircaloy + varying amounts of beryllium, have amassed over 3500 hours. In almost every case breakaway has occurred. The same series in a pH 10 environment have 1400 hours and also show breakaway. A striking difference in behavior arises with Zircaloy with up to ½% uranium. These alloys have remained dark black and glossy through more than 3500 hours of corrosion, even in pH 10 environments. The light reflectance reading, made after each test run, has been normalized. The data now follow the weight gain curve almost exactly, except that the breakaway trend is noted in the light reading slightly before the weight change occurs. The normalized light values show a systematic change with a change in composition. This suggests that it is possible to analyze the welded brazed closures with the light reflectance method to determine approximate beryllium and/or uranium contents. The data are being analyzed on the computer program JCT 100 to allow complete characterization of the data with the various compositions.

Closure and Joining. A melting point analysis has been made of 16 Zircaloy + 5% Be braze alloys, representing three different vendors. Melting temperatures ranged between 986 and 1020 C with a median at 998 C. Some of these alloys were reported to have poor flow characteristics. An examination of those alloys which had poor flow showed a third phase inclusion, either a carbide or oxide. The poor flow characteristics did not necessarily correspond to high melting temperature. The high melting temperature is due solely to variation in beryllium content, always on the low side, while fluidity is a function of third and fourth phase inclusions.

Welding. Fifteen four-inch long hot-headed N-Reactor outer tube sections were closed by projection welding of an end cap to one end using the 1200 KVA, 600,000 amp resistance welder at the U.S. Army Rocky Mountain Arsenal in Denver.

Five of the welds are now being sectioned for metallographic examination. Six are being held for possible autoclave tests pending examination of the five. The remaining four represent machine conditions which apparently did not produce the desired welds.

Fuel Component Development. Laboratory tests to study possible damage to N-Reactor process tubes by charging N-fuel elements are proceeding. Tests have been conducted to determine to what extent the reactor process tube thickness would be reduced by repeated corrosion of the bare Zr-2 that is exposed from the scratching by the fuel element supports during each charge cycle. From these tests it was determined that after 2000 passes or approximately 100 charging cycles the tube wall would be reduced by 0.018 to 0.023-inch if each fuel support passes over the same wear track. Most of this thinning results from repeated removal of the corrosion layer formed after each charging when the reactor is at temperature.

Tritium Release from Irradiated Aluminum-Lithium Alloys. Laboratory outgassing studies have been conducted on thin sections of an irradiated aluminum-1% lithium fuel element at 300, 330, 365, and 385 C (572, 626, 689, and 725 F). Rates of tritium escape were measured by pressure rise in a constant volume system. Composition of the evolved gas was determined by mass spectrometer analysis. In initial outgassings at 300 C on a given Al-Li section, the gas was primarily hydrogen. The hydrogen appeared to be associated with the sample surface. The steady-state tritium evolution rate at 300 C was below the system leak rate. At 330 C, the steady-state outgassing rate was slightly higher than the system leak rate; at 365 and 385 C, the evolution rates were clearly defined and linear (after an initial period of nonlinearity) in runs to 26 hours. At 365 C, tritium release was 2.1% in a 25-hour run. At 385 C, the release was 2.9% in a 19-hour run.

Calculation of a temperature coefficient from the outgassing data is currently complicated by appearance of a dark film on the sample surfaces. There is evidence that the film effects the outgassing rates. The source and composition of the film are presently being investigated.

Rupture Testing of Irradiated Fuel Elements in IRP. A rupture test was conducted in the IRP Loop of an N-inner fuel element with an unbonded tapered end cap, which had been irradiated to 1900 MWD/T in KER at 185 C. It was defected by a 0.025-inch hole at the end cap. The rutpure began after a 62-minute exposure to water at

300 C. The water was then cooled at the current N shutdown schedule. The ruptured area extended 1½ or 2 inches from the end cap over the entire circumference. The end cap came loose. The weight loss was 88 grams. The rupture behavior was similar to that in a similar test of an element with a tapered end cap.

2. Corrosion and Water Quality Studies

Evaluation of Nonchromium Inhibitors. An in-reactor test was started in five tubes of the 1706 KE single pass facility to evaluate a mixture of sodium nitrite and sodium silicate as a replacement for sodium dichromate as a corrosion inhibitor in process water in the IPD reactors. This would permit elimination of Cr-51 effluent activity. In preparation for the test the inlet piping was conditioned with dichromate-free water, and the in-reactor tubes were decontaminated or replaced to provide dichromate-free surfaces. Weighed fuel elements, coupons, and resistance corrosion probes were charged to monitor corrosion.

Corrosion of Carbon Steel. Measurements of the corrosion of carbon steel at both pH 6.6 and 7.0 were obtained from large coupons and corrosion probes in the water systems at 100-K, 100-D and 100-H. Both uniform corrosion and pitting were the same at the two pH values. Maximum pit depth in 12 months of exposure was 17.5 mils at pH 7.0; 16.5 mils at pH 6.6. Average corrosion (as measured by weight loss) was 3 mils for both sets of samples.

Corrosion of Zirconium in Heated Crevices. A test of a zirconium-clad heated corrosion sample in TF-2 was terminated after 19 days because of failure of the electrical heating elements. Test conditions were 288 C, heat flux of 600,000 Btu/hr-ft^2, pH 10.1 with NH_4OH. Only slight corrosion of the zirconium (a small fraction of 1 mil) was observed in the crevices under three N-fuel supports, where boiling had occurred. This is good evidence that the severe corrosion in heated crevices encountered with LiOH can be avoided by use of NH_4OH. Three similar heated specimens, operating at heat fluxes of 200,000 and 300,000 Btu/hr-ft^2 are being tested in TF-2 and TF-3 to exposures of 3 or 4 months.

Three samples heated to a heat flux of 600,000 Btu/hr-ft^2 have failed to date; the presently used type of heater is apparently unable to take this load. New designs are being investigated.

Testing of NH_2OH in KER Loops. The ex-reactor portion of KER-1 (carbon steel) was decontaminated using inhibited 5% ammonium citrate in order to study crud behavior during startup with a

freshly cleaned piping. The DF at various points ranged from 2 to 12; cell background readings dropped from 6 to 2 mr/hr. No significant crud problems have been encountered in the seven days since startup using NH_4OH at pH 10. Water quality has been good. Only slight amounts of solids (magnetite) were present in the coolant, and no appreciable change occurred in the cladding temperature of the upstream instrumented crud detector. The thermocouple readings of the crud detector are the same (within 2 C) of those prior to the outage.

Two improved crud detectors were fabricated with ceramic (UO_2-Mo) cores, which will operate at a higher heat flux (600,000 Btu/hr-ft^2). These will be used in future crud studies.

The two Zr-clad instrumented crud detectors, two stainless steel-clad heater elements, and four Zr-clad NIE-1 heater elements discharged from KER-1 on December 2, after exposure in ammoniated coolant, were examined. A reddish-brown crud film covered the four NIE-1 elements, the upstream steel-clad heater, and the upstream crud detector. The deposit was continuous and equally heavy on these six pieces. The downstream crud detector and steel-clad heater had virtually no crud film on the heat transfer surfaces. These two elements had operated with surface boiling. It is believed that this boiling acted to prevent crud deposition. Neither the upstream nor the downstream instrumented crud detectors had shown measurable increase in cladding temperature (greater than 10 or 15 C) due to crud buildup during this exposure.

Corrosion of N-Reactor Components. The corrosion rate of copper tubing was measured as 4000 mg/dm^2/month in a 7-week test in recirculating deionized water at 305 C, pH 10 with LiOH. The silver plating on samples of N-Reactor valve retaining rings exposed during this test was completely removed during the first week of exposure; this behavior is similar to that encountered in a previous test with NH_4OH.

Effect of Chemical Cleaning on N-Fuel Elements. Two additional tests were conducted to determine the corrosivity of a proprietary EDTA-ammonia cleaning solution (A1K-14) on N-fuel elements. In the first test, a 7% solution was recirculated at 120 C for 24 hours in the carbon steel TF-8 Loop. The solution depleted rapidly, and large amounts of iron oxide precipitated on the fuel elements. Most of this deposit was removed by subsequent recirculation of deionized water at pH 10, 260 C. The second test was conducted in the carbon steel TF-5 Loop using a 15% solution at 115-150 C for 6 hours. The solution did not become depleted and no deposition occurred on the fuel elements. Corrosion of the fuel elements was nil in both tests.

3. Gas Atmosphere Studies

Zr-Graphite Compatibility Loop. It has been discovered that the moisture analyzer on the loop was grossly out of calibration voiding earlier data obtained on the diffusion of water vapor through graphite. After recalibration, new diffusion rates of water vapor through graphite were obtained with a 24-inch N-Reactor tube block. These diffusion rates show essentially no temperature dependence from 450-650 C (842-1202 F). At 350 C (662 F), water diffuses more slowly. The initial data indicate that a maximum dew point in the N-Reactor gas of -21 F is required 21 feet into the reactor in order to keep the Zr process tubes oxidized and inhibit gas phase hydriding.

Graphite Burnout Monitoring. Small graphite monitors exposed from September 23, 1963 to January 8, 1964 in Channel 1081 at B-Reactor disclosed no burnout rate exceeding 1% per 1000 operating days (%/KOD).

Burnout monitors exposed from September 22, 1963 to January 25, 1964 in Channels 1889 and 1960 at C-Reactor showed rates below 1.5%/KOD.

Monitors from Channels 3066 and 5080 at KE-Reactor also showed low burnout rates $\leq$ 1%/KOD. Exposure was from July 3, 1963 to January 24, 1964.

Kinetics of the Reaction $CO_2 + H_2 \rightarrow H_2O + CO$. An Arrhenius activation energy of 52.5 kcal/mole has been observed over a temperature range of 762 C to 975 C for the hydrogen-carbon dioxide reaction. Using the rate expression

$$\frac{d(H_2O)}{dt} = k(CO_2)(H_2)^{1/2}$$

values for the specific rate constant, k, were obtained which ranged from 0.0045 at 762 C to 0.338 at 975 C. In these experiments the first order dependence on the concentration of carbon dioxide was verified. At 880 C, the rate was independent of the residence time in the reaction zone for times up to three seconds. No induction period was observed. For all experiments the flow rates were adjusted such that a reaction time of approximately two seconds was maintained, thus eliminating any significant contribution from a back reaction.

The effect of hydrogen on the reaction has also been studied over the experimental temperature range. A plot of the normalized

rate, $P_{H_2O}/P_{CO_2}t$, versus $P_{H_2}^{1/2}$ results in a straight line which does not extrapolate to the origin but to a positive value of the rate. This anomaly could not be explained by the presence of extraneous water and another explanation is being sought.

Graphite-Water Vapor Reaction. The effect of concentration of water vapor in helium on the oxidation of TSX graphite has been determined in the pressure range from 0.25 to 7.60 mm and at temperatures of 790, 820, and 850 C. Nineteen oxidation experiments on a sample of TSX graphite (0.426-inch diameter and 2 inches long) which was oxidized to ~ 5% burnoff in air at 550 C prior to the water-vapor oxidation tests gave the following result:

$$R(g/g/hr) = 2.933 \times 10^7 e^{-59,300/RT} p^{1/2} \qquad (1)$$

where the pressure of the water vapor is in mm. The square-root dependence is in contrast to a first power dependence found by others at higher pressures.

4. Thermal Hydraulic Studies

N-Reactor Studies. Analysis was continued of laboratory data obtained in the heat transfer and fluid flow experiments with the full scale, electrically heated model of an N-Reactor fuel column with prototypic outlet piping. The calculations include provisions for factoring heat losses, estimated from zero power runs, into the calculations. These corrections improved the heat balances, with powers calculated from flow and temperatures, and those calculated from electrical measurements generally agreeing within 2 or 3%. In a few cases, at low flows and powers, where errors would tend to be magnified, disagreements on the order of 5% were observed.

The following conclusions are indicated by calculations completed to date:

(1) At pressures of 1200 and 850 psig, the two-phase to liquid-phase pressure-drop ratios for the outlet piping are about 30% lower than those predicted by Martinelli-Nelson relationships but can be approximated closely by the Levy momentum model. For the downstream dummy section, the two-phase to liquid-phase pressure-drop ratios at 850 and 1200 psig are on the order of 40 to 50% below values predicted from Martinelli-Nelson relationships.

(2) For normal reactor operating conditions, with a riser-to-riser pressure-drop of 135 psi, the tube flow rate predicted from experimental measurements with the model matches that calculated from standard pressure-drop equations.

(3) At an outlet riser pressure of 1200 psig, a flow reduction caused by plugging upstream of the fuel column would not produce flow instability in a maximum power tube.

(4) At normal reactor operating conditions, flow and heat distributions to the three coolant channels in the fuel-column model are approximately as listed below:

Channel	Fraction of Total Flow	Fraction of Total Heat*
Center Hole	0.085	0.089
Inner Annulus	0.520	0.500
Outer Annulus	0.395	0.411

*Assuming 4% of heat comes from outside the process tube; e.g., graphite heating.

Analysis of the experimental data is continuing.

Detailed plans are being made for experiments to determine transient thermal and hydraulic behavior in an N-Reactor fuel column during certain accidents. The full scale model of one-half of a fuel column, with prototypic inlet and outlet piping and fittings will be used in these experiments. Assembly of the experimental piping system, and installation of instruments to measure flows, pressures and temperatures has been completed. The first experiments planned will be designed to study transient events following an inlet piping rupture.

Shipping Cask for N-Reactor Fuel. A heat transfer analysis of a cask for off-site shipment of irradiated N-Reactor fuel was performed. This cask was designed to hold a total of 56 complete fuel elements in four circular wells located symmetrically about the center of the square cross-section of the cask (Drawings SK-2-21106 and SK-2-21107). During transit, the wells will be "dry", and the heat resulting from the decay of fission fragments will be transported by conduction, convection, and radiation across the air gap between the clusters of elements and the well walls.

The heat transfer analysis predicted maximum cask body and fuel surface temperatures of 182 F and 568 F, respectively. After exposure to a gasoline fire for one hour, these maximum temperatures become 621 F and 780 F. These temperatures were judged to be well within allowable limits. The calculations will be included as an appendix of the final design report by Facilities Planning Operation.

Present Reactor Studies. Laboratory experiments and data analysis were concluded for tests with a full scale, electrically heated model of a charge of KV self-supported fuel elements in a 1.724-inch ID smooth-bore Zircaloy tube. The fuel model was 1.520-inch OD by 0.420-inch ID with a 28.1-foot heated length. The peak-to-average power ratio was 1.40 with the stepped cosine power distribution. Ceramic spacers, 0.086-inch high to simulate KV self-supports, allowed a maximum eccentricity of 20.6% from the normal 0.102-inch annulus.

A total of 129 steady state runs were made for determining the pressure drop flow relations across the assembly at constant tube powers of 675, 970, 1360, 1750 and 1940 kw. The experimental results show a nonconservative deviation from the analytical model previously used in establishing the instability limits (T-A-I limits) for individual tube outlet temperatures. Consequently, the KV geometry will require modified instability limits such as those specified in HW-75523[1] for the C-overbore smooth-tube geometry.

The conditions for incipient film boiling were briefly investigated for the KV fuel and tube geometry. Film boiling was initiated at outlet coolant enthalpies corresponding to about 15% steam quality at the end of the fuel charge with tube powers of 1360 to 1940 kw. The flow rates at film boiling are comparable to those found previously for ribbed-tube geometries[2] and can be approximated as gpm at film boiling = 0.014 kw.

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(1) HW-75523, "Instability Limits for Smooth-Bore Process Tubes," P. A. Carlson, Nov. 6, 1962.

(2) HW-71074, "Experimental Boiling Burnout Conditions for Hanford Production Reactors," E. D. Waters and J. M. Batch, Sept. 20, 1961.

It was determined that reducing the rear header pressure below 60 psig had no effect on the film boiling flow rate or the heated surface temperature at film boiling because of critical flow conditions in the outlet fittings.

Circumferential water temperature measurements were made at the end of the fuel model and at five feet upstream from the end. R values, defined as the ratio of maximum water temperature rise at some point to the bulk coolant temperature rise along the same length, $\frac{\Delta T_{max}}{\Delta T_{max}}$, were calculated from these measurements. The R values at the upstream location were larger than those at the downstream location and the upstream values varied considerably with tube power. The R values at the downstream end compare favorably with those obtained previously from laboratory and in-reactor experiments, i.e., R = 1.19 for 20.6% eccentricity. The larger R values upstream indicate that downstream values should be adjusted before use in fuel jacket and process tube corrosion rate calculations or predictions.

Local boiling at the end of the active charge will occur with outlet temperatures in excess of about 138 C (280 F).

Transient upstream plugging tests were not conducted with this assembly because the transient Panellit pressure response is postulated to be the same as that for the C-overbore smooth-tube geometry.

The results of the above study will be reported in HW-80942, "Results of Thermal Hydraulic Experiments with KV Self-Supported Fuel in a Zircaloy Tube - K-Reactor," E. D. Waters and M. E. Shockley, 3/2/64.

Hydraulic Tests. Tests were performed in the hydraulics laboratory to determine the effects on process tube coolant flow when a charge of fuel elements is positioned so that the uppermost fuel slug extends upstream of the front nozzle inlet port. This positioning of fuel elements occurs during the charge-discharge operation when part of the old charge has been pushed out the open rear nozzle by the incoming new fuel elements. The concern in this case is that the old fuel being discharged would not receive sufficient flow of water through the tube for adequate cooling. The results of the tests indicated that while the total flow entering the front nozzle would be reduced from normal charge-discharge rates, the flow of coolant past the old fuel remaining in the tube would be increased. The reason for this is

that the fuel upstream of the inlet port restricts the flow normally passing out the front of the open front nozzle during charge-discharge operations and forces a larger percentage of the flow through the tube.

The increase in tube flow for the B-D-F tube was from 0.15 to 0.70 gpm at 60 inches of header pressure; 0.30 to 0.70 gpm for the K tube and 0.03 to 0.18 gpm for the C assembly. A report on the results of the test has been issued as HW-80897.

The use of 32 OIIIN I&E and 46 KIIIE fuel elements has been proposed for the C ribbed aluminum tube. Several tests were run to determine the annular and hole flow to be expected with the above fuel charges at water temperatures of 20 C and 50 C. Annular flow for the OIIIN elements was 80% of the total at 45 gpm, the hole flow accounting for the remaining 20%. Flows obtained with the KIIIE elements were 61.5% annular and 38.5% hole flow at 45 gpm. The relative hole-annular flow was approximately constant regardless of water temperature or total flow rate.

Forty venturis which have been reamed to 0.344-inch ID have been measured and tested - the purpose being to (1) calibrate the modified venturis, and (2) define acceptable throat diameter tolerances. A preliminary review of the results obtained to date indicates that a tolerance limit of 0.344- +0.002- -0.000-inch throughout the throat will give a maximum deviation of $\pm \frac{1}{2}$ gpm at a flow rate of 45 gpm. Eighty additional venturis will be examined as they are received and the flow characteristics determined.

In an effort to reduce Van Stone seal insert damage during the charge-discharge operation and to improve the sealing characteristics of the present insert, a modification involving thicker insert walls and an additional rubber seal gland in the upstream portion of the assembly, has been proposed. Several tests involving flow and pressure-drop characteristics through the modified insert (Mo III) have shown that the additional pressure drop which can be expected will be 4 to 5 psi greater than that obtained with a standard insert. By increasing the ID from 3/4-inch to the standard size of 7/8-inch (decreasing the wall thickness), it is expected that the hydraulic characteristics will approach those of the present Mo II assembly. Further tests are contemplated in an effort to improve the flow characteristics of the modified thick wall seal insert.

A modified version of the present H-Reactor spline cap insert (H-1-5986) has been tested and the results reported in HW-80917.

[redacted]

The modification consisted of replacing the 3/4-inch OD portion of the solid shank with a 3/4-inch OD, 5/8-inch ID hollow tube. Radial holes were drilled through both walls to enable the coolant to flow through the hollow portion of the insert as well as through the annular region between the insert and the front nozzle barrel. Compared to the spline cap insert presently in use, the new model showed a decrease in pressure drop of about 5 psi. The corresponding flow increase is of course applicable only for those tubes which will contain poison spline assemblies.

5. Shielding Studies

N-Reactor Shield Evaluation. Since the monthly report of January, the major N-Reactor shield evaluation accomplishments are: completion of the zero power full core cold shield plug irradiation, partial analysis of the small core experiment, preparation for the hot zero power test and the N-2 test.

Small core results are not yet completely analyzed. Thermal and epithermal foil activation occurred at the inner foil locations and spectra were obtained with the He^3 and Li^6 detectors. Counting rates were low with the Li^6 detector; hence, the spectrum has poor resolution due to poor counting statistics. Efforts are under way to program the problem in MAC to check the measured spectrum against a MAC calculated spectrum. Neutron spectra taken with the He^3 and Li^6 detectors were plotted and the gamma fluxes measured by the ion chambers were plotted as a function of distance through the reflector. The calculations for the gold foils and sulfur pellets are yet to be completed.

6. Graphite Studies

N-Reactor Graphite Irradiations. The long term irradiations of N-Reactor graphite continue to progress satisfactorily. The second third-generation capsule, H-5-3, was removed from the GETR after successfully completing five cycles, 115.2 effective full power days, of irradiation. The capsule was disassembled and found to be in good condition. The 24 samples and 64 of the flux monitors were recovered and returned to Hanford.

The first of the fourth generation capsules, H-4-4, was installed in the GETR and has completed one cycle of reactor operation with all thermocouples functioning properly.

The second of the fourth generation capsules, H-5-4, is under construction. This capsule contains 24 previously irradiated

[redacted]

samples, seven of which have been irradiated in three previous capsules, 14 in two previous capsules, and three in one previous capsule.

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

PRTR Fuel Fabrication. Three Vipac and one swage compacted PRTR fuel elements were fabricated.

Tests of 800 PRTR fuel rods containing high energy impacted ("Nupac") UO_2-1% PuO_2 revealed essentially uniform PuO_2 distribution; gamma scanning of fuel rods containing Nupac material therefore will be discontinued.

A new crusher for the impacted UO_2-PuO_2 was installed; increased fuel throughput and improved control of particle size distribution were obtained. Oil contamination problems also appear to be solved; and outgassing of fuel processed through the new crusher can now be discontinued.

Closer control of Nupac process variables has been achieved. The preheat time and temperature are being controlled to 80 ± 5 minutes and 1180 ± 25 C, respectively. The fire pressure and tooling of the Dynapak machine have been controlled more closely, although some density variations of the material from lot to lot still occur. One important variable is the weight of material in the can, which affects the final impacted length and has a significant effect on the impact pressure. A weight control to fire pressure nomograph is being prepared to guide the machine operator. In the meantime, the greater fire pressures are being used.

Cladding components and core materials are being assembled for the fabrication of 3-foot long, 0.505-inch ID, Zircaloy clad fuel rods for the Plutonium Recycle Critical Facility. The UO_2-2 wt% PuO_2 core material is being prepared by the pneumatic impaction, and the fuel rods will be vibrationally compacted.

PRTR Fuel Element Performance. A significant milestone in the development of swage compacted UO_2 power reactor fuel elements was reached as exposures in PRTR satisfactorily passed 5000 MWD/T fuel. Satisfactory exposures in excess of 10,000 MWD/T are anticipated. PRTR exposure of swaged and vibrationally compacted UO_2-PuO_2 elements also exceeded 5000 MWD/T at month-end.

The PRTR loading now comprises seven UO_2 elements, 18 vibrationally compacted UO -PuO elements, and 60 swage compacted UO_2-PuO_2 elements. Among these are 14 swaged or vibrationally compacted

UO_2-1 wt% PuO_2 elements containing impacted fuel. Exposure of one of these elements is in excess of 1000 MWD/T.

A purposely defected, swaged UO_2, PRTR fuel element is operating remarkably well in the PRTR rupture loop. The element, previously irradiated to 1140 MWD/T_U, was defected by drilling a 1/16-inch diameter hole through the cladding of one rod. The element was examined in the basin after a short irradiation time (about 2 MWD), but after two thermal cycles and two pressurizations, no noticeable changes had occurred. Since the start of the defect test the element has been subjected to three pressurizations and eight power generating periods. The power rating has varied between 400 and 500 kw.

A vibrationally compacted UO_2-1% PuO_2 element failed after approximately 325 MWD/T. It has not been completly examined, but preliminary evidence indicates that hydriding has occurred at the end cap. Previous hydriding failures of vibrationally compacted elements have occurred at the same point, indicating that the crevice between end cap and cladding must be eliminated. One such element is now under irradiation. End caps now being fabricated have no crevice.

Swaged UO_2 fuel element 1026, previously reported to have split during approximately 18 months of storage in the PRTR storage basin water, was destructively examined. X-ray diffraction analysis of the fuel in the area of the split indicated the presence of UO_2 and UO_3. Analysis of the samples from a cross section revealed UO_2, UO_3, U_3O_8, and U_4O_9.

Irradiation Testing Program for EBWR Prototypic Fuel Rods. Twelve capsules are under irradiation in the MTR. Thirteen Vipac capsules, assembled with near-EBWR size, Zircaloy tubing (secondary aluminum sheath added later), contain impacted UO_2-1.5 wt% PuO_2 fuel (average bulk density 82% TD). The fuel was vacuum outgassed at 600-700 C prior to vibrational compaction. One capsule was discharged after one MTR cycle and returned to HAPO for ceramographic examination. The discharged capsule reached an estimated exposure of 2.8×10^{20} fissions/cm^3 (970 MWD/T of UO_2) and generated an estimated surface heat flux on the Zircaloy (1.07 cm or 0.42 in OD) of 158 w/cm^2 (500,000 Btu/hr-ft^2).

Irradiation of two Vipac capsules containing impacted UO_2-2.5 wt% PuO_2 and physically mixed UO_2 and PuO_2 continues in the ETR. Goal exposure of each will exceed 1.43×10^{20} fissions/cm^3 (5000 MWD/T of UO_2).

[redacted]

Six capsules with fission gas plenums duplicating the upper 25 cm (10") of an EBWR, UO_2-PuO_2 fuel rod (which includes 10 cm of fuel, a ZrO_2 insulator pellet, and a type 302 stainless steel spring) were fabricated for irradiation in the MTR-ETR. The Vipac specimens contain impacted UO_2-1.5 wt% PuO_2 fuel (average bulk density 80% TD) which had been vacuum outgassed at 600-700 C before vibrational compaction.

The specimens are planned to operate with Zircaloy surface heat fluxes of 158 and 316 w/cm^2 (500,000 and 1,000,000 Btu/hr-ft^2, respectively) to exposures of approximately 0.3 x 10^{20} to 1.8 x 10^{20} fissions/cm^3 (1000 to 6000 MWD/T of UO_2, respectively).

Cladding Procurement. The remainder of the 2800 EBWR fuel element cladding tubes has been received. Of these, 961 remain to be tested with ultrasonic equipment at HAPO. Reconditioning internally scratched tubes has been moderately successful, increasing yields by approximately 25%. Rejects identified as cracks or inclusions will be returned to the vendor for replacement. Testing of the EBWR tubing is proceeding at a rate of 160-180 tubes a day.

Examination of typical defects was made on the 0.690" ID Zr-2, PRTR fuel rod cladding received in January and small laps or tears 0.002-0.003" deep were observed. The rejection rate on this tubing was approximately 5%. The rejected tubes will be sent back to the vendor for replacement or credit.

PRTR Fuel Element Design Studies. End caps were designed for PRTR 19-rod cluster fuel elements to allow all rods except the center one to be rejuvenated. The end caps provide for complete bundle disassembly and allow individual rods to be opened, rejuvenated, and reclosed without breaking individual rod wire wraps. Existing, remote banding methods will be used to reband the rejuvenated cluster. A PRTR fuel element will be fabricated using these end caps and cored, sintered, fuel pellets.

Fuel Element Development. A PRTR fuel element with conventional top and bottom end hangers misaligned 60 degrees was tested in the TF-7 loop at PRTR operating conditions and with an imposed vibration of 25-30 cps. Examination of the wear pads after one week and five weeks revealed only small amounts of wear. No apparent difference was observed between the behavior of this element and a conventional element, indicating little benefit was obtained in rotating the hangers.

An instrumented (eddy current) fuel element is performing satisfactorily in EDEL-1 loop. Preliminary data indicate the fuel element is behaving as a pendulum with a period of about 0.8 sec and an amplitude of 0.001".

ThO_2-PuO_2 Irradiations. Preliminary examination of an irradiated ThO_2-8.8 wt% PuO_2 pellet specimen revealed structural features and radial fission product distributions similar to those observed in irradiated UO_2, UO_2-PuO_2, and ZrO_2-PuO_2. Recrystallization and columnar grain growth occurred over approximately 56% of the cross section and a center void formed in the top half of the specimen. The expected central fuel temperature was only 1800 C, suggesting that the thermal conductivity of ThO_2-8.8 wt% PuO_2 is less than that of UO_2. Plutonium self-shielding and the high absorption cross section of ThO_2 caused a relatively flat radial temperature profile in the specimen.

Unidentified inclusions associated with high fission product concentrations were observed in the microstructure. The inclusions, which are spherical when dispersed in the fuel matrix and irregular when located in cracks, have a transparent crystalline appearance. Cracks with deposits of this crystalline material on their surfaces show high fission product concentration on the autoradiographs. Other cracks without crystalline deposits, are not visible on the autoradiographs.

The central plastic region of the ThO_2-PuO_2 fuel slumped into the dished ThO_2 pellet at the bottom of the fuel column. Diffusion bonding occurred at the interface, but no PuO_2 or fission product migration occurred.

Recycle Fuel Element. Special connecting end caps and components for a special end hanger for this fuel element were fabricated.

Fuel Element Refurbishing. Two UO_2 fuel elements (1034 and 1099) with broken rod wire wraps were repaired in the PRTR basin.

Pneumatically Impacted Cluster Fuel Elements. A stainless steel clad, 80 vol% UO_2-stainless steel, four-rod cluster, with stainless steel end caps and spacing brackets, was fabricated by a "one-shot" pneumatic impaction process. Five micron UO_2 and 30-40 micron stainless steel pwoders were mixed and cold pressed into stainless steel tubing. The tubes were inserted in a mild steel die and stainless steel powder was poured into and around the ends of the tubes forming end caps and spacing brackets. After impaction, the mild steel die was removed in nitric acid.

Corrosion and Water Quality Studies

Corrosion of Iron and Nickel Base Alloys in Superheated Steam. Penetration results have been obtained on various stainless steel and nickel base alloys exposed 100 days in 550 C (1022 F), 5000 psi deoxygenated steam and descaled. There appears to be no general pressure effect on the uniform corrosion rate of the alloys in the 3000-5000 psi pressure range. The various alloys may be grouped in the following order of increasing uniform corrosion penetration: Group I (0.012 to 0.019 mils) - Incoloy, Hastelloy X, Inconel X; Group II (0.033 to 0.072 mils) - Hastelloy N, PDRL-102, R-235 Nichrome V, R-27, 446 SS; Group III (0.107 to 0.139 mils) - an experimental 25% Cr, 3% Al, 0.5Y, Fe alloy, 17-4 pH SS, 304L SS; and Group IV (0.200 to 0.331 mils)- 316L SS, two heats of 406 SS, 430 SS.

Penetrations from the previous 3000 psi test ranged from 0.010 to 0.340 mils as compared to a range of 0.012 to 0.331 mils for the present test. Differences in corrosion rates observed for 446 and 304L SS are attributed to different heats rather than a pressure effect.

Reactor Components Development

Shim Rod Development. Assembly of the driving head for the PRTR second generation shim rod was completed and the unit has been installed and operated satisfactorily with the aluminum lead screw assembly in the environmental test facility. Some difficulty with sticking in the aluminum lead screw assembly was encountered. This is probably due to foreign particles in the water because the occurrence and location of the sticking is random. The aluminum lead screws will be replaced with zirconium lead screws prior to reactor installation.

Fretting Corrosion Investigations. The instrumented 19-rod bundle fuel element and associated instrumentation for the PRTR fretting corrosion study have been completed. Calibration drift in the instrumentation following heatup of the fuel element in EDEL-I precluded satisfactory measurement of the exact location of the fuel element within the pressure tube. Efforts are being directed to correct this fault.

Some preliminary observations of fuel element-pressure tube vibration characteristics can be made. The specific assembly under test shows movement of only about one mil at the bottom of the fuel element. This movement is of comparatively low frequency requiring

about 0.8 second to complete the peak-to-peak travel. Simultaneously, vibration measurements of the pressure tube show 0.2 to 0.5 mil amplitude at frequencies of about 20-25 cps. Generally, the 0.2 mil amplitude of the pressure tube corresponds to one extreme of the fuel element movement and the 0.5 mil amplitude to the opposite extreme. While relative position of the fuel element with respect to the pressure tube is unknown (due to the aforementioned instrument drift), it is conjectured that the minimum pressure tube amplitude coincides with the close proximity of the fuel element to the tube wall and, conversely, the maximum reading with a distant location of the fuel element.

Data on the effect of coolant flow rate and temperature on vibration are also inconclusive due to instrument drift.

The vibration counter and alarm system installed on the inlet connection to PRTR Tube 1348 has been modified to provide a continuous strip chart recording of the vibration level at the inlet jumper. The calibration of the recorder, which measures 10 mils peak-to-peak amplitude full scale at 30 cps frequency, was checked with a Raydata vibration analyzer. Vibration levels from the strip chart recorder for a shutdown and startup were determined. From the data it appears that tube vibration increases with increases in power level, coolant temperature, moderator level, and flow through the tube.

The vibration counter and alarm system previously employed in the reactor contained a capacitor-amplification circuit which attenuated the predominate 30 cps frequency by a factor of 10. Hence, that instrument showed only the high frequency vibration which was about 1.2 mils amplitude at full reactor power level. The present unit measures the 30 cps frequency, which yields a normal full power amplitude of about 8 mils peak-to-peak.

The high frequency unit appeared more sensitive to reactor disturbances. Another such unit connected to a strip chart recorder will be built to record these high frequency vibrations (above 60 cycles per second) to obtain a more complete vibration picture of the reactor pressure tube.

Nozzle Closure Seal Test. The nozzle closure seal testing has been limited because of EDEL-I loop shutdown during calibration and installation of the instrumented vibration fuel element. Each of the three assemblies under test is arranged to provide leakage collection data from prototype PRTR inlet flange, nozzle-

to-tube seal, and nozzle cap. Spare parts materials as used for the PRTR have been installed in all three assemblies except that one assembly contains a tubular stainless steel nozzle cap gasket. Only 75 ft-lbs torque is being used on the cap set screws, and the molycote lubricant as used at PRTR is used on the threads. Typical leak rates at 375 F, 1050 psig and 120 gpm, range from 0.2 to 8.3 ml/hr.

Design Analysis

Nuclear Safety of Mixed PuO_2-UO_2 Fuels. Analytic expressions were derived for peak power, time to peak power and energy release for a nuclear excursion with time delay in Doppler feedback. An exponential burst model was assumed in which the feedback is expressed by a nonlinear term whose effectiveness is governed by the time constant for heat transfer to the uranium. The problem is somewhat analagous to the charging of a capacitor by an exponentially increasing voltage in which the time constant of the charging voltage is a nonlinear function of the energy stored in the capacitor. Analog calculations have been completed for a wide range of time constants and step reactivity inputs. Detailed comparisons with the analytic model have not yet been completed.

PRTR Specific Power Increase Study. A preliminary study of increasing the specific power in the PRTR was completed. Five different cases involving a decrease in number and/or length of fuel elements were investigated. A range of increase in enrichment of from 1.3% (55 tubes, 88" fuel) to 2.1% (43 tubes, 48" fuel) was estimated. Although the moderator level coefficient and individual shim rod strength would be increased, and may require some control system modification, the inherent safety of the reactor would not be appreciably reduced. Details of this study are reported in HW-80926.

PRTR Flux Trap Study. A brief investigation of maximum attainable neutron fluxes in the PRTR was completed. By constructing a flux trap in the center of the core and surrounding this with Pu-Al fuel, unperturbed thermal fluxes near 10^{15} n/cm^2-sec can be achieved. Insertion of the gas loop pressure tube and fuel samples were found to reduce the unperturbed flux of the order of a factor of two, depending on the blackness of the sample. It is concluded that an unperturbed flux near 10^{15} n/cm^2-sec may be achieved, but increases beyond this are limited by heat flux with the present fuel.

Fuel Re-use. The burnup calculations for the current study of fuel re-use in four types of thermal reactors are essentially complete. Presently the economics code is being run on these burnup tapes to generate the fuel cost information. Preliminary analysis of results on the U-235/U-238 control cases show only a 0.2-0.3 mill fuel cycle cost variation among the four reactor models, but about a 0.5 mill difference between zirconium and stainless steel jacketing in a given reactor type.

PRTR Pressure Tubes. Investigation of a flaw in PRTR Tube 6115 has revealed a local area under the flaw containing a hydride concentration estimated from photomicrograph to be 400 ppm. On bursting, the strength of this tube was similar to that of an irradiated defect free tube even though the failure did initiate at the flaw in the tube. Apparently the sensitivity of the PRTR pressure tubes to notches is low.

Pressure Tube Monitoring. Fifteen process tubes were visually inspected this month. The process tube in the FERTF was examined twice. Prior to the first of these two inspections, the FERTF operated with an Al-Pu fuel element. With this fuel element a 0.01 mm (0.004-inch) fretting mark was formed by the small pads at the upper end bracket. Prior to the second of these two inspections the FERTF operated with a UO_2 fuel element and during this period there were no new fretting corrosion marks.

In the Zr-2 tube in process channel 1354, a 0.01 mm (0.004-inch) fretting mark was formed by a large (6 mm x 12 mm or $\frac{1}{4}$-inch by $\frac{1}{2}$-inch) spacer pad at the lower end of the fuel element. Prior to this, no such marks with depths greater than 0.025 mm (0.001-inch) have been produced by wide pads.

The tube in process channel 2053 was found to have a new 0.25 mm (0.010-inch) deep fretting mark associated with the spiral wire bundle warp of a UO_2 fuel element. There are now at least five such fretted areas in this tube.

2. Plutonium Ceramic Fuels Research

Plutonium Ceramics Irradiations. Two irradiation capsules containing PuO_2 and PuN were returned to Hanford from the MTR. Radiometallurgical examination of the PuO_2 is proceeding. The oxide sample appeared to be extensively cracked but was not broken. Calculated burnups on these two samples are: PuO_2 - 22.0 x 10^{20} fissions/cm^3, 13.5% Pu burned, or 84,600 MWD/T Pu. PuN - 19.9 x 10^{20} fissions/cm^3, 9.22% Pu burned, or 57,700 MWD/T Pu.

Irradiation of UN-20 wt% PuN. Three capsules containing UN-20 wt% PuN solid solution pellets were charged into the Materials Test Reactor. The calculated fuel surface heat flux is 572 watts/cm^2 (1,810,000 Btu/hr-ft^2) with a fuel surface temperature of about 1100 C. The three specimens are to be irradiated to planned burnups of: 2.8, 5.4 and 13.3 x 10^{20} fissions/cm^3 (40,000; 77,000; and 190,000 MWD/T Pu).

Mixed Nitride Systems. Thermal stability studies show that PuN, UN, and a PuN-50 mol% UN solid solution are highly volatile in a 10^{-5} Torr vacuum at 1800 C. In contrast, ThN, TiN, and ZrN are very stable under these conditions. Experiments were therefore begun to obtain solid solutions of PuN in ThN, TiN, and ZrN in an effort to stabilize PuN. A four-hour soak at 2000 C under 500 mm of nitrogen resulted in a partial solution of ZrN in PuN, but no solubility was detected between PuN-ThN or PuN-TiN.

3. Ceramic (Uranium) Fuels Research

ThO_2 Irradiation Studies. No catastrophic corrosion of the aluminum cladding was found on a deliberately defected thoria element that had been irradiated in the MTR GEH-4 facility. Further metallographic examination will be conducted to determine the amount of internal cladding corrosion.

Two other irradiated thoria elements are being destructively examined. Analysis of approximately 150 ml of gas extracted from one element (free volume 41.8 ml) revealed 98.8 mol% hydrogen. Air contained in the element after welding was sorbed or reduced and H_2 was released during irradiation. There was no apparent sintering of the thoria during irradiation. Chemical dissolution studies are in progress, and isotopic analyses will be conducted.

Irradiation of UO_2 Single Crystals. A zone of porosity approximately 10 microns wide was observed at the exposed surfaces of a UO_2 single crystal irradiated to an estimated exposure of 5 x 10^{20} fissions/cm^3. The porosity appeared as small bubbles approximately 0.1 to 0.5 micron in diameter.

High Temperature Microscopy of UO_2. High temperature transmission (optical) microscopy of single crystal UO_2 with metallic uranium inclusions showed the formation of uranium hydride after two hours at 500 C in 8% H_2-A. The hydriding of the uranium increased the volume of the inclusions and resulted in the fracture of the UO_2 thin sections.

Fabrication of Cermets. Core-mandrel bonding problems were eliminated in the fabrication of cermet platelets by autoclaving an oxide coating on the steel mandrel prior to impaction. An aluminum oxide coating also was used successfully. To obtain this type of protective coating, the stainless steel mandrels were vapor plated with aluminum, which was then oxidized in an autoclave.

Metal Clad Cermets. Loading of a stainless steel clad, 40 wt% UO_2-stainless clad, 40% UO_2-stainless steel tubular fuel element, with stainless steel end caps, was started. The fuel element is a preliminary design to evaluate the pneumatic impaction process relative to tubular sections.

4. Basic Swelling Studies

Irradiation Program. Two irradiation test capsules reached their irradiation goal and were discharged from the reactor. One operated isothermally at 625 C (1157 F), independent of reactor power level, while the other operated at about 425 C (797 F) when the reactor was up but dropped to ambient when the reactor was down. Both capsules contained split tubular specimens of high-purity uranium, uranium with 100 ppm of Fe and Si, and uranium with 500 ppm of Fe and Al.

An irradiation test capsule containing high-purity uranium specimens of various geometries is currently undergoing final bench testing prior to transport to the reactor for irradiation.

Preparations are under way for the irradiation of uranium specimens in a high pressure environment. A data package, summarizing the capsule design, has been submitted for approval. In addition, bids have been requested on the fabrication of components for three high pressure capsules and an order is being placed for (1) high pressure fittings and tubing for the manifold and (2) high pressure control system for use at the reactor.

A solid state temperature controller, with a silicon controlled rectifier power supply, was evaluated for dynamic response characteristics on a normal bench test of a completely assembled irradiation test capsule. The controller-power supply combination performed satisfactorily and the purchase of two of these units is being considered to replace obsolete equipment currently in use at the reactor.

Zr-2 clad rods of uranium and U-2 wt% Zr alloy are being irradiated in NaK-filled capsules to provide information on fuel performance under various conditions of cladding restraint. From additional experimental data a recalculation of the temperature distribution in the U-2 wt% Zr fuel rods was made. The results show a maximum irradiation temperature approximately 200 C (392 F) higher than was previously reported. The earlier data indicated that the fission gas induced swelling of U-2 wt% Zr was two to three times that of unalloyed uranium at an equivalent irradiation temperature and burnup. The refined data now show the swelling to be about the same for the alloyed and unalloyed fuels at equal temperatures. In addition, the microstructure observed in the various U-2 wt% Zr fuel rods now corresponds to that expected from considerations of the U-Zr equilibrium phase diagram. A second phase observed in the microstructure of U-2 wt% Zr fuel which was previously thought to be an extensive agglomeration of the delta phase is now shown to be gamma-1. These new results will require a re-evaluation of the influence and effects of zirconium additions on uranium swelling.

Recent metallographic examination of a U-2 wt% Zr rod has revealed extensive grain boundary tearing. The tearing along grain boundaries and possibly twin or deformation boundaries occurred in a band approximately 0.060-inch wide at the outer surface of the fuel rod. Extending approximately 0.175-inch from the center of the rod is a region in which large, irregular-shaped pores are aligned perpendicular to the major axis of many of the twins. The appearance of this microstructural feature is very similar to that observed in unrestrained, thin uranium cylinders described previously. This fuel rod was irradiated at a maximum fuel temperature of 550 C (1022 F) to a total atom burnup estimated to be 0.25 a/o. This is the first evidence of grain boundary tearing and extensive aligned porosity in Zr-2 restrained fuel rods. Further metallographic examination of this sample is planned, as are density and accurate burnup measurements.

5. Irradiation Damage to Reactor Metals

Alloy Selection

Twelve refractory metal alloy tensile specimens irradiated at 50 C (122 F) in the ETR were examined and tested at room temperature. Eight of these specimens showed signs of corrosion after irradiation. Upon measuring the diameters of these specimens, a 0.040 - 0.050 inch decrease in diameter was observed.

Corrosion and oxidation tests of 10 nickel base alloys are continuing as part of a program to evaluate the applicability of these alloys as nuclear reactor structural materials. One series of tests were completed using an environment of 15 Torr water vapor in helium at 927 C (1700 F). The present series of tests are being run at 1038 C (1900 F) in the same environment. The alloys are also being tested in 650 C (1200 F) steam at 3000 psi. Initial results from these tests show that Hastelloy R-235 is severely attacked by intergranular oxidation by the 927 C (1700 F) environment.

In-Reactor Measurement of Mechanical Properties

Ex-reactor creep testing of annealed 304 stainless steel has been started. The first tests will be conducted under a stress of 30,000 psi at various temperatures between 400 C (752 F) and 600 C (1112 F). The primary objective of these first tests is to determine typical creep behavior of program material and to establish any structural changes which might affect creep rates.

Two creep capsules containing annealed 304 stainless steel have been shipped to the reactor and are awaiting charging.

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.

Work has progressed in the development of two new methods of evaluating material properties - low cycle fatigue and cleavage. The compatibility of the low cycle fatigue specimen design and loading arrangement was established. Tests were also conducted to evaluate a cleavage specimen and loading method. The cleavage technique is a new method of evaluating the fracture properties of materials.

A dynamic loading system capable of applying 45,000 pounds load at a rate of five million pounds per second was installed during the month. This system will play a role in the determination of the effect of shock loading rates on the yielding, flow, and fracture properties of materials. The performance of this machine is being studied.

Eight button-head specimens of Inconel X-750 (four representatives each of two test heat treatments) are being irradiated during

cycle 61 in the G-6 position of the ETR. Two wafers, each representing one of these same two heat treatments, are undergoing metallographic examination to determine the comparative effects of the heat treatments upon microstructure and hardness. Treatment #1 is the standard heat treatment for this alloy: 2150 F (1177 C)/2 hrs, air cool; 1550 F (843 C)/2 hrs, air cool; 1300 F (704 C)/20 hrs, air cool. Treatment #2 incorporates a lower aging temperature: 2150 F (1177 C)/2 hrs, air cool; 1200 F (649 C)/50 hrs, air cool. While optical microscopy shows a general precipitate for treatment #1 and none for treatment #2, essentially no difference in hardness was found. Diamond pyramid hardness using a 1 kg load was 351 and 359, respectively. Four more specimens of Inconel X-750, representing test heat treatment #3 /2150 F (1177 C)/2 hrs, air cool; 1000 F (538 C)/50 hrs, air cool/ are being prepared for shipment to the ETR for irradiation during cycle 62.

A three-element furnace for 700 C (1292 F) tensile and stress-to-rupture teating has been built and is being fitted with control thermocouples. The furnace control has been given a preliminary check. Work on the design of a 700 C (1292 F) inert environment irradiation capsule has been initiated. Apparatus for room temperature low cycle fatigue studies has been designed and fabricated.

Damage Mechanisms

Electron transmission microscopy studies of titanium doped iron show many small precipitate particles about 0.1 micron in size with dislocations associated with them. Upon straining this material five percent, the precipitates appear to act as sources of dislocations, thus providing another explanation of the lack of a yield point in this material.

The determination of the functional dependence of strain rate and the effective stress (the rate controlling stress) and the quantitative calculation of effective stress is the most important single obstacle to the formulation of a rate theory of deformation. Considerable work has been devoted to the solution of this problem in the past few months both by experimental and theoretical techniques. The analysis of stress relaxation experiments indicates this technique is capable of providing information about the functional relation between strain rate and effective stress over a range of five or six orders of magnitude for strain rate. Estimates of the effective stress at specific strain rates agree well with estimates made by independent methods.

ATR Gas Loop Studies

Insulation Fabrication. Insulation between the shroud and primary piping of the model loop must have high temperature stability, low coefficient of thermal conductivity, strength, minimum outgassing characteristics and fabricability into special shapes. After extensive evaluation of commercially available insulations, castable Fiberfrax was selected as the material which best met the available criteria.

Because vendors have declined to provide Fiberfrax in the special shapes required, it has been necessary to develop the techniques for fabrication of the insulation section required for the model loop. The insulation fabrication work is about 70% complete.

Design Test 1172. A report on thermal and bending tests of Grayloc seals was issued at the completion of Design Test 1172. This report will be incorporated in a final report on the Grayloc fitting and to be issued in March 1964.

Corrosion Studies. Samples of Haynes 25 and Hastelloy X were exposed to pure nitrogen at 100 mm pressure and 2192 F(1200 C) for four days. During this period the alloys lost weight slowly due to evaporation. The rate of evaporation was reduced by the nitrogen overpressure. The evaporation rate for Haynes 25, for example, was a factor of 20 less than for the same temperature in vacuo.

Metallographic examination revealed no nitride phase. The samples were bent into a U-shape and smashed flat with no observable cracking. It is concluded that nitrogen present as an impurity in the helium will not adversely affect Haynes 25 and Hastelloy X in ATR applications.

Gas Chromatograph Valves. It has been confirmed that the valves in the gas chromatograph began to leak after use. This makes O_2 and N_2 analysis impossible. The valve design is a poor one for trace analysis work where O_2 and N_2 contents of 1 ppm are important. Replacement of the teflon seals with Buna-N O-rings substantially reduced contamination; however, for analysis of ultra-pure gases, this valve is not adequate. Plans are under way to purchase a welded valve with no sliding seals.

Fast Neutron Spectrometry. Fast neutron spectra were determined instrumentally during the N-Reactor startup tests in an experiment which represents the first successful attempt to correlate theoretically and experimentally determined spectra in detail. The

measurement was made with the detector suspended in Ball Channel 60 at the reactor center line.

Measurements were attempted using He^3 and Li^6 detectors positioned within the core of reactor. Data from the He^3 detector could not be resolved, apparently because of the sensitivity of the detector to gamma radiation. However, a spectrum determined from tests with the Li^6 detector correlated well with the spectrum calculated for that point in the reactor using transport theory.

Further tests will be made during the second phase of the physics startup tests during which there will be full core fuel loading. It is expected that shifts in fast spectra from the center of the reactor to the boundaries should be measurable.

6. Gas Cooled Reactor Studies

Gas-Graphite Reactions. Irradiations of hydrogen, methane, carbon monoxide and nitrogen in sealed quartz capsules were conducted in a Hanford reactor at about 550 C and in a Co-60 facility at 30 C over a range of pressures. Changes in gas composition and graphite weight were measured, and the nature of the deposited materials was examined.

There was no measurable formation of CH_4 in capsules containing graphite and H_2 in the pressure range from 3 to 30 atm. However, the results of these experiments were not quantitative because of the probable loss of H_2 from the permeable quartz capsules.

The radiation-induced decomposition of CH_4 to form H_2 and carbon in reactor at 550 C was measured in capsules filled with H_2-CH_4 mixtures. In some of the capsules with this mixture, trace amounts of C_2H_6 and C_3H_8 were formed. Gamma irradiations of H_2-CH_4 mixtures to a dose of 9.68×10^8 R produced very small changes in the gas composition.

In-reactor capsules containing H_2-CO mixtures at total pressures ranging from 1 to 25 atm produced a good yield ($\sim$ 40%) of CH_4 and CO_2.

In only one of the four capsules filled with N_2 was any C_2N_2 detected in the gas phase; however, small amounts of a solid, believed to be paracyanogen, were found on the walls of the four capsules.

A report on this work has been issued as HW-78948.

Graphite-Water Vapor Reaction under Gamma Irradiation. It has been previously reported that the measured oxidation rates in the water-graphite oxidation studies may possibly be attributed to the presence of trace quantities of oxygen in the helium stream. During the report period, experiments were done which have established the concentration of oxygen necessary to cause the observed effects at about 6 ppm. This study is currently being extended to determine the effect of the water-vapor concentration on the graphite-oxygen reaction.

EGCR Graphite Irradiations. The seventh capsule, H-3-7, in the series of long-term irradiations of EGCR graphite, was discharged after four cycles of operation in the GETR.

Effect of Tensile Load on the Radiation-Induced Contraction of Graphite. Investigation of the effect of a constant tensile load applied to graphite specimens irradiated in a helium atmosphere in the core of the ETR has shed new light on the proposition that radiation-induced creep is of sufficient megnitude to relieve all the differential contraction stress generated in large graphite blocks.

Several tensile specimens of NC-8 graphite cut parallel to the extrusion axis have been simultaneously stressed to 600 or 800 psi and irradiated at a temperature of 475 C in the ETR for different total exposures. Over the exposure range of 3 to 8 x 10^{20} nvt, E > 0.18 Mev, the tensile-loaded specimens always contract, although not as much as the unloaded controls that are exposed simultaneously. The amount of restraint seems to be dependent on load.

The rupture of several tensile samples at approximately the same strain implies the possible existence of a strain-rupture limit. However, there is as yet insufficient evidence to establish unequivocally whether the ruptures were coincidental or related to a specific strain value.

One interpretation of radiation-induced contraction in graphite suggests that it is a function of the internal stresses. If this is so, then these results indicate that contraction can be partially restrained by an opposing external tensile stress, and the amount or restraint is dependent upon the magnitude of the external stress.

An explanation is proposed for the mechanism of failure of the two broken samples. During neutron irradiation at these temperatures,

graphite contracts. Some of this contraction can be restrained by a tensile load. The rate of restraint could be dependent on the radiation-induced creep observed by others at lower exposures.(1) Because the creep rate is low, the differential stress in the specimen increases faster than the creep strain can relieve it. A complex internal stress pattern develops because the graphite used in reactors is an anisotropic mixture of crystalline grains and amorphous or cryptocrystalline binder, has a porous structure, and contains both micro- and macrocracks. Stress will not be uniform over the entire cross section and eventually a rupture stress will be produced at some weak link, and the process of failure begins. This pattern will be repeated and accelerated at successive weak links until a point is reached where the accumulated stress exceeds the strength of the remaining material, and brittle fracture occurs.

Thermal Conductivity of HLM-85 and TSX Graphite. Thermal conductivity measurements were completed on a series of irradiated and unirradiated HLM-85 and TSX (N-Reactor) graphite samples. The TSX samples were measured perpendicular to the extrusion axis, whereas the HLM-85 samples were cut from a 0.9-inch diameter tube and measured perepndicular to the axis of the tube. Data were obtained by the thermal pulse method in the temperature range of -195 C to 525 C. The maximum thermal conductivity of the unirradiated samples was measured at -10 C. For HLM-85 graphite the maximum was 0.42 cal/cm/sec/°C and for TSX graphite, 0.30 cal/cm/sec/°C. Samples were irradiated by General Atomic at two temperatures, 650 C and 1000 C, to an estimated exposure of 7×10^{21} nvt (E > 0.18 Mev). In the range 200 to 525 C, the thermal conductivity of the HLM-85 samples irradiated at 1000 and 650 C was 0.12 and 0.085 cal/cm/sec/°C, respectively. Similarly, the conductivity of the TSX graphite samples irradiated at 1000 and 650 C was 0.075 and 0.060 cal/cm/sec/°C. The temperature coefficient of thermal conductivity for the irradiated samples was zero in this temperature range. Below about 200 C the thermal conductivity of all the irradiated samples decreased with decreasing temperature. Typical values for the irradiated HLM-85 and TSX graphite samples at -195 C were 0.02 and 0.015 cal/cm/sec/°C.

(1) A. J. Perks, J.H.W. Simmons, "Radiation-Induced Creep in Graphite," AERE-R-4372.

7. Boronated-Graphite Studies

Boronated-Graphite Irradiations. The long term irradiation capsule continues to operate satisfactorily, well within preset specifications for all three sections. The gradual drift to higher temperatures reported last month continued and is attributed to slight changes in thermal conductivity of the materials of the inner capsule construction.

A replacement capsule is being prepared which will include irradiated samples in one of the sections from the first long-term irradiation. New samples will make up the inventory in the second and third sections. Two of the new replacement samples have been specially marked and replicas of the surfaces have been made. This will permit pre-post-examination of irradiation effects on identical areas by electron microscopy.

A series of three short-term (10^{19}, E > 0.18 Mev) capsules is planned to augment the long term irradiations. Included in one of the short term tests are two shielded samples to determine the relative effects of thermal and fast flux. Two of the capsules are planned to operate in the temperature range of 200 to 450 F.

8. Aluminum Corrosion and Alloy Development

C-1 Loop. The first corrosion test of aluminum-clad fuel elements in C-1 Loop was conducted at pH 7 with 290 C outlet temperature. The core temperature of the upstream fuel element (which operated at lowest power) was measured by a thermocouple. This temperature remained fairly constant for 50 hours after the reactor was brought to full power and the loop brought to full operating temperature. It then suddenly started to rise at a rate of about 2°C per hour. At the same time the flow started to drop, and the pressure drop across the test section started to rise. After about 15 hours the loop temperature was reduced, to avoid excessive corrosion. The core temperature continued to increase at the same rate when the loop was held at 205 C outlet temperature but held constant when the outlet temperature was further reduced to 130 C. Examination after discharge showed heavy deposition of red-brown crud on the thermocouple (upstream) element, less on the second and third elements, and only small amounts on the others.

The first test was discharged and replaced by a similar charge. This is also being exposed at pH 7, but at a lower temperature of 260 C outlet. The core temperature of the thermocouple element

rose about 5°C per day for 10 days and has since seemed to level out.

The C-1 Loop has operated well and its components have functioned properly, though maximum flow has been somewhat less than expected.

9. Metallic Fuel Development

Irradiation of Thorium-Uranium Fuel Elements. The irradiation of three tubular Zircaloy-2 clad thorium - 2.5 wt% uranium fuel elements continued in the ETR-P7 loop. The fuel elements have achieved an integrated exposure of 1.8×10^{20} fissions/cm^3 (5000 MWD/T). The fuel elements are operating at a maximum temperature of 540 C, a surface heat flux of 65 cal/sec-cm^2 (8.7×10^5 Btu/hr-ft^2), and a specific power of 60 watts/gm (183 kw/ft). Weight measurements made at an integrated exposure of 1.6×10^{20} fissions/cm^3 (4600 MWD/T) showed a maximum fuel volume increase of 9.5%, essentially no change from that measured at an exposure of 1.3×10^{20} fissions/cm^3 (3600 MWD/T).

Thorium Defect Testing. Behavior of defected Zircaloy-2 clad Th - 2.5 wt% U - 1 wt% Zr rods in 300 C water was observed by testing in a windowed autoclave. Previous tests had shown that some as-extruded specimens behaved well while others developed a longitudinal split in the cladding with an accompanying increase in corrosion rate due to the increasing area of core material being exposed. In later runs, specimens were vacuum heat treated at 750 C for one hour and 800 C for three hours to investigate the effect of thickening of the diffusion zone between the clad and the core. These heat treatments increase the diffusion zone from about 0.01 mil in the as-extruded material to approximately 0.7 mil and 1.9 mil, respectively. Both heat treated specimens behaved well in that external deformation was restricted to a single blister around the 25-mil hole through the cladding. Formation of the blister was apparent after about one hour and grew slowly until it was approximately 0.43" x 0.70" x 0.100" high when the test was terminated at the end of the tenth hour. Commencing within the first half-hour and continuing throughout the test, a fine corrosion product was expelled out through the defect. This continuing loss of thorium oxide could be utilized as a signal for early detection and location of a fuel element failure during reactor operation.

Peel Test of the Thorium-Zircaloy-2 Bond. Specimens from Zircaloy-2 clad Th - 2.5 wt% U - 1 wt% Zr core, coextruded rod stock were

prepared by machining circumferential grooves through the cladding, splitting the resulting quarter-inch bands longitudinally, and bending up on end to provide a tab. The specimens were mounted in a special holder that permitted rotation as the tabs were being pulled. A 60,000-pound Baldwin Universal Testing Machine was used to pull the tabs. The force acting upon the band was neither a true tensile nor shear stress but a combination of the two. Results of the peel tests are:

1. The force required to peel the clad from one of the as-extruded rods is about 310 lbs/inch at tab width and is about 330 lbs/inch for the other rod.

2. No significant change in peel force was observed from specimens taken from the front, middle, and rear of either extrusion.

3. Steam autoclaving at 400 C for 72 hours had no appreciable effect on peel force.

4. Vacuum heat treatment at 750 C for one hour increased the peel force from 310 to 340 lbs/inch, an increase of approximately 10%.

5. Vacuum heat treating at 800 C for three hours resulted in peel force in excess of 340 lbs/inch. Exact values could not be obtained due to parting of the tabs at the bond line. Tabs failed at forces from 340 to 560 lbs which is equivalent to 13,000-22,500 lbs/in^2 in the Zircaloy.

6. Cursory examination at low magnification indicates that in all cases peeling occurred at the diffusion zone. Failure of the tabs on the heat treated specimens gave indication of some grain growth in the Zircaloy.

Corrosion of Thorium Contaminated Braze. Corrosion data are being taken on a series of thorium-contaminated braze coupons with thorium content ranging up to 3.1% Th. The coupons have completed a total of 60 hours of autoclaving in pH 7 water. Color changes on the highest thorium content brazes are noticeable. All corrosion data are being handled by a special computer program for more rapid and more accurate evaluations.

Irradiation of Fine Carbide Uranium Fuel. In a test to investigate the ability of a submicron dispersion of uranium carbides to reduce uranium swelling, three NaK-filled capsules will be irradiated in

the ETR to evaluate the performance of fuel rods produced from chill-cast uranium shot. Two of these capsules each contain two fuel rods identical in uranium composition, but with one having a uranium carbide size of 2-5 microns produced from arc-melted uranium and the other a carbide size of less than 0.5 microns produced from the uranium shot. The third capsule contains two fuel rods with the fine carbide.

One capsule, GEH-14-609, which is presently in its second cycle of irradiation, has accumulated an exposure of 0.2 at% burnup. Goal exposure for this capsule is 0.3 at% burnup. The capsule is operating at a maximum core temperature of about 535 C. The capsule will be moved to another location in the reactor in order to attain the test design fuel temperature of 600-625 C.

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

Two-Phase Pressure Drop Through Pipe Fittings. A final report, HW-80970, was prepared concerning the study of pressure-drop experiments with typical reactor piping. The results are summarized as follows:

Pressure losses resulting from the flow of both liquid and steam-water mixtures were determined for horizontal two-inch pipe and piping components which included 90-degree bends of two-inch, three-inch, and ten-inch radius of curvature, a branch tee, and valves of both gate and globe design. Additionally, the study included expansions of two- and three-inch and $1\frac{1}{2}$ by 2-inch pipe; contractions of 3 by 2-inch, 2 by $1\frac{1}{2}$-inch, and 2 by 1-inch pipe; and an orifice with an opening to pipe diameter ratio of 0.80. Data were obtained at mass velocities from 1,000,000 to 4,000,000 $lb_m/hr\text{-}ft^2$ in the two-inch pipe and at steam qualities up to 24% by weight. Most data were obtained at 1200 psia, while a selected number of experiments were performed at 800 psia and 1600 psia.

The two-phase to single-phase pressure-drop ratios for the straight pipe were correlated favorably by the Levy momentum exchange model.(1) The model coincided with data at 1600 psia, agreed within ± 12% at 1200 psia and was least satisfactory at 800 psia;

(1) S. Levy, "Steam Slip - Theoretical Prediction from Momentum Model," Trans. of the ASME, Series C, Journal of Heat Transfer, pp 113-124, May 1960.

being a maximum of 20% low but only 8% low at 21% quality. Effect of the flow parameter on the pressure-drop ratios was shown to be negligible which is inconsistent with the results of other investigators. It is suggested that the flow-parameter influence diminishes with larger diameter pipe although high Reynolds numbers and small changes in the liquid friction factor may contribute to this result.

The single-phase pressure-drop data for the bends agreed quite well with values in the literature; however, the three-inch radius bend proved to be very sensitive to upstream disturbances. Ratios of the two-phase to single-phase pressure drop were different for the various bends; the maximum being for the three-inch radius bend with ratios $2\frac{1}{2}$ times the straight-pipe ratios. A correlation was written relating the bend ratios to the straight-pipe ratios over the pressure range of 800 psia to 1600 psia.

The data showed that if a homogeneous specific volume was assumed in calculating the kinetic energy change across a diameter change, then the two-phase to single-phase pressure-drop ratios were similar for both the expansions and contractions. Furthermore, these ratios conveniently agreed with the straight-pipe ratios at 1200 psia, the only pressure for which these data were taken.

The orifice had a single-phase pressure loss which corresponded to a resistance coefficient of 1.50, this being in agreement with reference works. The pressure-drop ratios for the orifice were lower than those of any other component and fell about 23% below the ratios for the homogeneous model, i.e., the ratio of the two-phase to single-phase homogeneous specific volume as a function of quality.

Single -phase pressure losses for the gate valve compared well with those cited in the literature, while the Y-pattern glove valve gave losses 30% below the generally accepted values. Steam-water flow through the gate valve produced pressure-drop ratios that increased linearly with quality and ran about 36% above the homogeneous model. The globe valve had pressure-drop ratios not consistent with the other components in that influence of the flow parameter was evident. Beyond a certain "critical" quality, which was dependent on the flow rate, a "choking" action occurred across the valve, resulting in a rapid increase in the two-phase to single-phase pressure-drop ratio. Critical discharge criteria were examined but were found to be a remote explanation for the phenomenon.

11. Advanced Reactor Concept Studies

Fast Supercritical Pressure Power Reactor. The report was revised to include fuel cycle costs for private ownership of fuel and has been submitted for publication.

Segmented Fast Reactor (SFR). The Segmented Fast Reactor (SFR) is a sodium-cooled, PuO_2-UO_2 fueled, 1000 Mw(e), fast power reactor. Two preliminary core design cases are being considered and studies embody the following major assumptions:

1. The hot spot factor of 1.5 assumed in the calculations is not a statistically determined factor, but it is taken from a fast reactor design having approximately the same fuel and coolant geometry.

2. The overall peak-to-average heat flux ratio for the core was assumed equal to two.

3. The axial power distribution was assumed to be a cosine with a 17-cm attenuation.

4. Heat transfer coefficients are assumed to be given by:

$$Nu = 0.625 \ (N^{Re})^{0.4} \ (Pr)^{0.4}$$

as in GEAP-3721.

Sodium physical and thermodynamic properties were taken from WADD-TR-61-96.

A very brief study is being made on different fuel pin sizes and coolant fractions to give an indication of a more satisfactory core arrangement from the standpoint of fuel and cladding temperature.

The plutonium isotopic composition is assumed to be 6.15 at% Pu-239, 30 at% Pu-240, 5.8 at% Pu-241, and 2.7 at% Pu-242.

The critical mass without moderator segmentation of Core I is about 2600 Kg of Pu (total) and of Core II is about 2900 Kg of Pu (total).

Some difficulty has been experienced calculating in the reactivity effect of sodium voiding. The axial extrapolation distance appears to be a function of neutron energy and sodium density. This requires a different axial leakage term for each energy group for each sodium density. At present there is some uncertainty as to

the magnitude and importance of the extrapolation distances for lower neutron energies.

Military Compact Reactor Plutonium Studies. Document HW-79449, Supplement 1, was issued, detailing additional evaluations made on the use of plutonium fuel in the Military Compact Reactor. Results of this study were summarized in last month's report. They indicate considerable potential for the use of plutonium fuel in reducing the size and weight of the reactor system, increasing reactor power, or extending the core endurance. The study pointed out several problems which would require solution in the development of a plutonium-fueled core for this type of reactor. These include the higher control strength requirements of these smaller and more reactive cores, and uncertainties in fuel design resulting from the scarcity of data on the properties of plutonium compounds. This study indicated that Hanford-developed advanced fuel designs could be used to advantage in a plutonium-fueled MCR core.

Issuance of this document completes the planned studies on the MCR evaluations.

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.

Quenching studies are being conducted on nickel specimens representing three levels of purity (99.997%, 99.98%, and 99.4%) in order to determine the nature of vacancy-impurity atom interactions. Qualitative measurement of quenching rates was conducted during the month, employing oscilloscope recording of the decay of visible radiation from the specimen during cooling. The indicated quenching rate for the first 100 millisecond was 2500 C/sec. There is believed to be an inherent time lag in the circuit used, so that the first 10-20 millisecond of the cooling curve is not recorded. The initial quenching rate is therefore thought to be at least 2500 C/sec, if not greater.

Tensile tests conducted on nickel foils which were aged after quenching showed considerable increases in the yield stress above the as-quenched value. Defining the percent increase in yield stress as the quantity

$$\frac{\sigma - \sigma_o}{\sigma_o} \times 100,$$

where σ_o is the 0.2% offset yield stress of a specimen held two minutes at 1675 K and slowly cooled, and σ is the yield stress after a particular quench-aging treatment, the percent increase in yield stress was found to vary from 1 to 100%. The pattern of behavior appears to be well established; the hardening reaction seems to involve the continual release of vacancies from clusters or traps of varying binding energies. Experimental verification of this hypothesis is expected when the transmission electron microscopy specimens which have been prepared are examined.

Molybdenum single crystal specimens for combined length-change and lattice parameter measurements have been annealed at 1330 C (2426 F) for 18 hours, and lengths remeasured. Most crystals showed no change after annealing, although a few apparently changed slightly. Lattice parameters are currently being measured by the symmetrical back-reflection technique. Large subgrains are not evident in the annealed crystals although the peaks are rather broad. This effect is ascribed to polygonization. Lattice parameters of the annealed crystals are reproducible to ± 0.00003A, or about 0.001%. This is about the same precision as the length measurements.

Preparation of seeded molybdenum crystals for subsequent purification and addition of known amounts of carbon has begun. The crystals are being grown by application of an electron beam zone-melting technique. The starting material is sintered 3 mm diameter polycrystalline molybdenum rod containing 15 ppm carbon, 10 ppm oxygen, and less than 150 ppm total impurities. Two seed orientations have been employed: $< 144 >$ and $< 149 >$. These axial orientations locate the $\{211\} < 111 >$ and $\{101\} < 111 >$ glide systems, respectively, in orientations of maximum resolved shear stress when the crystal is tested in tension.

Transmission electron microscopy studies of wafers cut from fractured single crystal tensile specimens of molybdenum containing 450 ppm carbon and irradiated to 10^{19} nvt has continued. The crystals were oriented to that the surface of the cut wafer was

parallel to the (111) plane, which is perpendicular to the slip direction commonly observed in molybdenum. Defect structures in the form of spots and loops were observed in all sections of the sample. Scattered dislocations were observed in the grip end of the sample. In the deformed region of the sample, channels were observed to intersect with each other at an angle of 60 degrees. One micrograph and accompanying diffraction pattern have been analyzed to date. In this case the channels correspond to traces of the (110) plane on the plane of the foil. This result is consistent with observations made on irradiated and subsequently deformed polycrystalline molybdenum foils. Occasional twins were also observed in the deformed region of the sample. Twins can be distinguished from channels by their change in contrast as the sample is tilted in the microscope; channel contrast does not change during tilting. The twins, like the channels, have been swept free of defect clusters. Some dislocations are present inside the twin. Many dislocations are present in the matrix surrounding the twins, and small channels were observed to emanate from the twin boundary. Unfortunately, the twins are narrow and electron diffraction patterns for deriving their orientation could not be obtained.

2. Plutonium Physical Metallurgy

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

Experimental studies of the creep characteristics of the stable phases of plutonium and the deformation during phase transformations under constant stress have continued. Typical true strain-time creep curves of each phase were obtained. The steady state creep rates of the beta, gamma, and delta phases at the alpha to beta, beta to gamma, and gamma to delta transition temperatures have been firmly established. The creep rate of alpha plutonium is quite small, about 5×10^{-5}%/hr at 115 C (239 F) and 7500 psi. Thus, there now exists a basis for comparing the creep rates of alpha and beta, beta and gamma, and gamma and delta at the respective temperatures of mutual stability. This affords the first opportunity to relate the creep rate of a pure metal directly with crystal structure. Additional data on the stress and temperature dependency of the steady state strain rate for the various phases has been obtained.

Further improvements have been made on the bright field etching of alpha plutonium using a nitric acid-alcohol-water etchant. This etchant reveals grain boundaries but does not attack the inclusions and microcracks in alpha plutonium as readily as the tetraphosphoric acid-2 ethoxy ethanol-water etchant. There appears to be much promise for use of this etchant to distinguish retained beta from the alpha matrix.

The study of transformation kinetics of electrorefined plutonium received from LASL was continued. The incubation time for transformation of the electrorefined plutonium was two hours at 90-100 C (194-212 F) whereas it is weeks at 95C (203 F) for vacuum cast virgin material.

In order to obtain the electrorefined plutonium received from LASL in a form suitable for rolling studies it was necessary to melt and recast the as-received round bar stock into thin slabs. Since it was essential to preserve the initial purity, the conventional crucible melting and pouring into a mold was considered to be less desirable than in-place casting. Consequently, the bar stock charge was placed in the hot top of an integral high density magnesium oxide mold and the entire assembly heated in a vacuum induction furnace using a graphite susceptor. The charge, when melted, flowed directly into the mold cavity producing a casting four inches by one inch by 0.125 inch.

Although the density of the as-received material, 19.5013 g/cm^3, increased to 19.5983 g/cm^3, metallographic examination revealed microcracking too extensive to afford optimum rolling conditions. It is apparent that the casting must be further treated to raise the density. This can best be done by beta pressing and preparations for such treatment are under way.

E. CUSTOMER WORK

1. Radiometallurgy Laboratory

Examinations and Measurements

Routine examinations and measurements are or will be reported as part of the sponsoring research and development programs.

Metallography

Samples Processed	182
Photomosaics	3
Autoradiographs	10

Chemistry

Burnup Dissolutions	29
Decladding Dissolutions	1
Fission Gas Collections	15
Vacuum Induction Furnace Runs	2

Physical and Mechanical Properties Testing

Tensile Tests	44
Density Measurements	4
X-ray Analyses	1
Crushing Tests (Ceramics)	30
Modulus of Rupture Tests(Ceramics)	18

General

Negatives Processed	507

Equipment

Dilatometer and Annealing Furnace. Shipment of the annealing furnace has been delayed again. The vendor was contacted. Replacement thermocouples were installed in the dilatometer.

"E" Cell Metallography Facility. The south half of "E" cell has been decontaminated and construction forces have started to erect the metallograph shielding. Design and fabrication of different equipment components are in progress. A design and construction work schedule was prepared which shows that installation of the metallographic sample preparation equipment should be completed by April 30, 1964. Installation of the metallograph is scheduled to be completed by August 15, 1964.

Microhardness Blister for "I" Cell. A purchase order was issued for a basic Kentron microhardness tester without optics or rectilinear stage. Fabrication of a table to support the shielding around the tester was completed. Fabrication of a sample transfer conveyor and shielding window frame is in progress. The optical relay system from the obsolete microhardness tester was decontaminated and sent to the optical shop for modification.

Metallograph Camera. Drawings are being prepared for a system to automate a number of the steps involved in preparation of micrographs which are used for construction of photomosaics.

Stereo Zoom Hot Cell Microscope. The stereo zoom hot cell microscope was received from the vendor.

Induction Heating Equipment for "N" Reactor Elements. A plug-mounted induction heating coil and quench tank was designed and is being fabricated for heat-quench tests on irradiated "N" reactor fuel elements. Installation of a 50 KW induction heating unit for use at "A" cell is in progress.

"B" Cell Modification. An Appropriation Request was written requesting authorization of funds to purchase two large lead glass viewing windows for "B" cell to improve visual control of remote operations.

High Temperature Tensile Testing Machine. All parts of the high temperature tensile testing machine have been received from the vendor. Temporary electrical connections were completed for out-of-cell testing and personnel training.

2. Metallography Laboratories

During the report month 399 samples were processed, a total of 787 macrographs and micrographs taken, 1853 negatives printed, and 7422 prints processed.

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

A request for uranium metal quality evaluation from NRD prompted an investigation of a procedure for rating the uranium according to the length and population of the stringers within the metal. It is known that the greater the number of stringers in a metal of a given type, the poorer will be the mechanical properties of that metal. The samples for study had to be prepared so that the normal inclusions could be distinguished from those which were aligned into stringers. To establish a stringer value rating system for uranium, submitted samples were prepared as follows: A longitudinal sample was cut to a convenient size and mounted in resin with a copper wire attached to each for conducting electric current. Samples were ground through 400 grit silicon carbide paper, then electropolished. The electropolishing solution contained one part by volume of 60% perchloric acid in eight parts of glacial acetic acid. Electropolishing required 0.4 amp per square inch which was obtained at about 20 volts. As the solution aged and as the

temperature changed, the necessary voltage would change slightly. Two minutes were required to remove the scratches of the 400 paper, and at this point those inclusions which occurred in stringers were uniformly over-etched so they could be distinguished easily and photographed at 5X with dark field illumination. After a sufficient number of photographs of varying stringer content were available, they were carefully judged as to which represented successive steps in metal quality, using only the stringers in this evaluation. A chart was made from these first photographs with arbitrary values assigned to each step in the sequence. The balance of photographs taken was then judged by visual comparison with the standard rating chart. It appears that this method can be used to characterize metal of similar chemical composition according different inclusion distribution patterns. Study of the results of such tests must be made in relation to uranium performance as fuel before the usefulness of the test can be fully determined.

A procedure is being evaluated for dissolving polyvinyl alcohol replicas away from the vapor deposited carbon film without having swelling occur which spoils the carbon replica. In this procedure the polyvinyl alcohol-carbon replica is mounted on a cold-finger condenser system and water vapor is gently condensed on the surface. Other investigators have reported this technique eliminates the swelling problem.

3. High Temperature Lattice Test Reactor

HTLTR Heating Element Mockup. The 4-foot long, 1-5/8-inch diameter graphite element installed last month has been in operation at temperatures up to 1700 C during the month. The cumulative time of operation at 900 C or above is 504 hours.

Periodic arcing occurred within the containment box during the run. An attempt to determine the cause of the arcing will be made during the next shutdown.

Exit gas samples have shown the presence of H_2, O_2, CO_2, CO and H_2O. From these sample results, it appears there is a water-graphite reaction occurring within the box.

HTLTR Vault Calculations. The MAC calculations were completed for the HTLTR shield representing the final core design. One calculation included a Boral and steel outer reactor skin, the other calculation omitted the Boral. The Boral does provide appreciable shielding savings; however, the decision to use Boral will be made on economic considerations.

Compatibility of Structural Materials with the HTLTR Environment. In a test to determine the compatibility of candidate structural materials with the HTLTR environment, decomposition products of some of the test materials caused corrosion failures in the system necessitating a shutdown of the test. The test materials included alumina refractory, boron carbide, boron nitride, and aluminum nitride in addition to several candidate alloys. The test involved exposure to a nitrogen gas-graphite environment at 1200 C. The test, intended to run for 1000 hours, was stopped after 75 hours when a corrosion failure of a nickel gas-effluent line occurred. A possible source of the corrodent is thought to be the refractory brick. The boron nitride and boron carbide specimens also showed some evidence of instability at this temperature. When the unstable materials are positively identified, they will be eliminated and the test re-run.

4. EBWR Fuel Elements

EBWR Plutonium Fuel Element Fabrication. More than 150 EBWR fuel rods containing Nupac UO_2-1.5 wt% PuO_2 were fabricated. To date, more than 600 such rods have been fabricated by either the resonant beam or resonant plate vibratory compaction techniques.

Characterization of (EBWR)-UO_2. O/U analyses of pulverized UO_2 samples after low temperature roasting to increase the oxygen content (as a step in the pneumatic impaction process) revealed the larger mesh particles were only slightly oxidized, and the -325 mesh particles contained about 90% of the excess oxygen. It thus appears possible to control the O/U ratio by roasting only the -325 mesh material and mixing it with the coarser fraction.

5. NASA Fuel Development

Tungsten-Uranium Coprecipitation. In the development of submicron UO_2 particles in tungsten matrices, uranium and tungsten were coprecipitated from an ammonium meta tungstate solution to produce a product having approximately 10% by volume UO_2 in the as-reduced product. The calcined precipitate was reduced in hydrogen at 900 C for four hours, 900 C for nine hours, and 1000 C for five hours to determine the effect of time and temperature upon product quality. Product quality was damaged initially by water in the furnace atmosphere. The problem was corrected by installation of a Deoxo unit and a Driente dryer in the hydrogen supply line. Exit dew points of -50 F or lower are now being achieved. The reduced powders are being densified by high energy impaction. Evaluation of the pellets is covered in another section of this report.

A mixture of UO_2-W powder (nominal 5 wt% UO_2) which was prepared by a simultaneous precipitation process followed by a reduction step was densified by a high energy rate process (Dynapak) at 1200 C. The bulk density of the resulting compact was 19.0 g/cm^3. The dispersion of the UO_2 (size and morphology) in the W matrix is presently being evaluated by metallographic techniques.

Tungsten Honeycomb Fuel. Metallography on the tungsten plate in shell demonstration extrusion previously reported has been completed. A good diffusion bond with some oxide inclusions exists between the plates and shell with no cracks evident in a transverse section at the middle of the extrusion. Larger grains are evident in both the wrought tungsten and molybdenum plates (molybdenum used as filler between the tungsten plates), and the pressed and sintered tungsten and molybdenum shells. The interface between the sintered molybdenum and tungsten is much smoother than that between the sintered molybdenum and either the wrought molybdenum or the wrought tungsten. The large grains are, therefore, considered the cause of interface roughness and indicate the best extruded honeycomb will result from use of all sintered powder components. Inclusions in the W-W bond are probably oxides and are due to a leak in the billet assembly during preheat. These can be prevented by improved welding techniques.

Cermet Fuel Development. A tungsten honeycomb element, 3-3/4-inch diameter and 3½-inch long, with hexagonal 1/8-inch holes, was fabricated by pneumatic impaction. Fabrication techniques similar to those used for 2½-inch diameter grids were used. Good control of dimensions was obtained.

An experimental cylindrical fuel shape was pneumatically impacted. The shape consists of eight concentric cylinders held in place by connecting webs. Good grid dimensions were maintained in the outer five rings. However, the center of the grid was distorted due to failure of the punch during impaction. The 2¼-inch diameter, 80 vol% tungsten-UO_2 cermet grid was fabricated using mandrels machined from mild steel cylinders, vibratory compaction, pneumatic impaction, and mandrel removal by chemical techniques.

Several tungsten-clad and unclad, 80 vol% tungsten-UO_2 cermet plates were fabricated by pneumatic impaction. Particle sizes of both the tungsten and UO_2 were approximately one micron. The plates will be used in basic studies of UO_2 fuel loss during thermal cycling at high temperatures.

Cycling Behavior of NASA Tungsten-20 Vol% UO_2 Cermets. Further study on the cycling behavior of NASA tungsten-20 vol% UO_2 cermets has shown additional characteristics affecting the UO_2 fuel loss rates.

1. A UO_2 phase develops in virtually all W grain boundaries on cycling between 2000 C or higher, and 1000 C or lower, regardless of starting tungsten or UO_2 grain size. A metallic phase appears in many UO_2 grains during early high temperature cycles but is not apparent on further cycling; no metallic phase has been detected in the grain boundaries.

2. The presence of small particle UO_2 (less than one micron diameter) appears to inhibit tungsten grain growth, whether or not large UO_2 grains are present. The smaller W grains in turn reduce the rate of UO_2 loss by vaporization.

3. The logarithm of fuel loss rate is inversely proportional to 1/T above 1800 C.

4. A clad specimen with nonclad edges exhibits a decreasing rate of loss up to 25 cycles at 2600 C; this behavior may be caused primarily by UO_2 losses out the nonclad edges. Further cycling results in a rapidly increased UO_2 loss rate. The loss rate decreases with decreasing cycling temperatures; calculation of activation energies after 40 cycles indicates the loss at that stage is largely by UO_2 vaporization.

Work currently in progress is concentrated on studying behavior during early thermal cycles at 2400 C or higher. At that stage actual opening up of the cermet grain boundaries appears to be the critical phenomenon, leading to subsequent gross fuel loss by vaporization.

Thermal Stresses in Cermets. Mechanical stresses in a W-20 vol% UO_2 cermet grid under extreme temperature gradients are being investigated by thermal cycling in an arc plasma.

No breakdown of a honeycomb grid structure was apparent after ten 10-second cycles, although cracks visible in the reject grid before cycling widened.

The web thinned and partially melted in the area of plasma impingement. A temperature difference greater than 2000 C across the grid face was inferred from the tungsten oxide formation.

6. Other Customer Work

Irradiation Specimens for Phillips Petroleum. These elements consist of aluminum-clad, gold-aluminum and lithium-aluminum alloys for reactor physics experiments. Five gold alloys are required which contain up to 25 wt% gold. Six lithium alloys are required which contain less than 4 wt% lithium. Similar samples were fabricated from different aluminum alloys in 1962. A coextrusion process is used to fabricate the tubular shaped elements. Ten aluminum-2 wt% silicon alloy billets were cast, from which the cladding components will be machined. Preparation of the gold alloys was begun.

Crud Detector Elements. Fabrication of crud detector elements for Coolant Systems Development was completed. The elements are Mo-UO_2 cermet pellets in stainless steel cans. Two thermocouple elements and two heater elements were fabricated.

Calcination of Sol-Gel ThO_2. In cooperation with Process Engineering Development, the effect of heating rate and temperature during the calcination of sol-gel ThO_2 was investigated. Before calcination, the material contains approximately 4 wt% water and some nitrate. It was found possible to obtain dense (97% TD), unshattered specimens by using an initial heating rate of 5 C/min and a peak temperature of 1150 C. Rapid initial heating caused fragmentation, and specimens calcined at temperatures below 1150 C have subsequently disintegrated when exposed to either water or the atmosphere.

Vibration Testing. A program was initiated for vibration testing of cermets at temperatures to 2600 C. A high temperature vacuum furnace and a 50-lb thrust electrodynamic vibrator are being used for the experimental work. The vacuum seal for the vibrating actuator rod is a problem.

Thoria Development. The second ton of thoria fuel elements was completed. Irradiation experience is satisfactory to date after 25 days of exposure on one ton of fuel in 100-F Reactor.

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[redacted]

PHYSICS AND INSTRUMENTS LABORATORY

MONTHLY REPORT

FEBRUARY 1964

FISSIONABLE MATERIALS - 02 PROGRAM

N-Reactor Lattice Parameter and Spectral Measurements Tests at Startup

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

Relative effective mass measurements are under way on the Eu, Lu, Pu-Al, and U-235-Al pins which were irradiated in the cold N pile test. The pins will be irradiated with a constant neutron flux in the PCTR, and relative masses will be deduced from the subsequent activities. Rotation of the pins during irradiation will assure identical flux exposure for all pins of each type. Normalized Cu pins will be used as a standard in the mass measurement and to measure any lateral flux gradient. The counting of the residual activity in the pins is near completion. The modifications for positioning and rotating the pins in the PCTR are in progress.

Both the test and spare fuel elements for the hot test have successfully survived a 72-hour autoclave test.

PCTR Measurements of Δk_∞ Between N-Lattice and PCTR Mockup

Experimental measurements were begun and completed in the PCTR to determine the change in k_∞ between the PCTR N-mockup core, and the N-core with NPR Zircaloy-2 process tubing and fuel cladding. A report, HW-80731, "Measured Differences in k_∞ Between the N-Lattice and the PCTR Mockup," has been written.

N-Reactor Flux Ratios

Measurements have been performed in the PCTR N-mockup core concerning flux ratios to be expected in special fuel and poison columns as compared to the regular .947 w/o fuel columns. Bare and cadmium covered copper foils were irradiated with several different loading configurations. The measured gamma activities were processed by computer code APDAC-I and the results

have been made available to N-Reactor Department.

The flux ratio data and reactivity measurements on the worth of the special columns are being documented.

NPR Utilization Studies

An extensive series of calculations for the NPR lattice with standard U-235 enriched fuel is under way. The calculations employ the GAM(HRG)-TEMPEST codes for cross section averaging and the HFN Diffusion Code for the calculations of criticality. A major part of the calculations involve the determination of an effective fuel surface area for resonance absorption. For this purpose a Monte Carlo procedure for the calculation of first flight collision probabilities is used in conjunction with a recipe by C. R. Richey and L. L. Carter (HW-77871, pp. 15-16). This effective surface area is used to properly account for the spatial self-shielding in the determination of the effective U-238 resonance integral. Detailed temperature coefficient calculations for the lattice have been made and are being compared with presently accepted results. The purpose of the calculations is to develop and test a set of physics methods which can be used in the present NPR utilization studies.

Instrumentation

In preparation for scheduled N Reactor physics tests, low noise coaxial instrument cables to replace marginally acceptable cables originally purchased were investigated, and the temperature recording system for the nuclear detector thimble cooling water was completed. Connections will be made to the thermistor temperature detectors as the thimbles are installed.

Modifications were designed and incorporated in an N Reactor process monitor instrument to provide a sensitive neutron flux monitor. The instrument is planned for use during N Reactor startup tests which are to follow the physics tests.

Development and test work was continued on the N Reactor low-level neutron monitoring system. The four fission-counter-cable assemblies were extensively tested and were found to be acceptable. Work is in progress to improve the linearity of the pulse amplifier system.

The size of radioactive sources to be employed during operational testing of the gamma spectrometer portions of N Reactor's fuel rupture monitoring system has been determined. In addition, a review was made of the instrument calibration procedures as prepared for the spectrometer system. The prototype spectrometer was actually calibrated for use in KE Reactor loops test.

Engineering assistance was rendered to Reactor Design, IPD, regarding shielding and collimation requirements for measuring the I-131 content of irradiated fuel on a systematic basis.

System Studies

The N Reactor secondary systems simulation which consists of one steam generator, one dump condenser, and one surge tank model and associated piping has been made operational on the new computer. Controller circuits are now being debugged and connected to the system simulation.

Several additions were also made to the primary system simulation. They include the heat exchanger bypass valve dynamics, hydraulics, and heat balance relationships; the new digital-type transport delay simulators; and the primary coolant volume contraction and expansion calculations. In addition, the primary coolant temperature control circuits were designed, including the temperature measurement lags and controller output limiters. Construction of the control system simulator to be used with the N Reactor studies is approximately 50% complete. Trunk lines are being installed between one of the two control system simulator racks and the new analog computer. The other rack is yet to be wired. Of the planned 35 plug-in control circuit boxes, eight are completed and checked operationally; 12 more are approximately 80% complete.

A decision was made to drop further efforts on MIDAS programming for N Reactor systems after it became apparent that the computing time required to make useful runs would be prohibitive. The major systems have been programmed and partially tested.

Analysis of the N Reactor neutron flux signals recorded in January shows that the useful signal to noise ratio is so low that meaningful kinetics data cannot be obtained. This result was not entirely unexpected, however, in view of earlier theoretical studies pertaining to small, space-independent reactors. One additional test is planned during the low-power physics tests to confirm these conclusions.

One of the two transport lag systems for the N Reactor simulation studies was checked out. The transport lag systems are to be used in the N Reactor simulation where temperature of the primary coolant in the region between reactor and heat exchanger is to be studied. The system provides variable time delay in order to simulate the effects of changing coolant flow.

SEPARATIONS

Critical Experiments with PuO_2-Plastic Mixtures

Criticality experiments performed during the month with PuO_2-polystyrene compacts and the Remote Split-Table Machine were designed to provide further information on the effect of core shape on criticality. Critical core configurations studied ranged from near cubes to thin slabs. The plutonium concentration of the core material was in each case 1.12 g/cc (2.2% Pu^{240}), with an H/Pu atomic ratio of ~ 15. From the data obtained, the critical dimension of a Lucite reflected cube is 8.5 in., and the critical mass contained therein is 11.1 Kg Pu. (The critical dimension for a bare cube is 12.4 in with a critical mass of 34.6 Kg Pu.)

Information pertaining to the critical thicknesses of "infinite" slabs is of interest for nuclear safety applications, both locally and off-site. From an analysis of the data, a value for the critical thickness of an "infinite" slab, composed of the above core material and fully reflected with Lucite, is 2.6 ± 0.2 in. The critical thickness of an unreflected infinite slab is calculated to be 6.4 in. from buckling conversions.

The critical thickness of the reflected PuO_2-plastic slab differs appreciably from the value estimated in LAMS-2415 for a Pu-H_2O mixture. The current data would indicate a critical thickness for a PuO_2-plastic slab which is > 50% larger than that for a Pu-water mixture with the same H/X ratio and Pu density.

The effect of placing a 0.03 in. cadmium sheet between the core and Lucite reflector was determined for a rectangular prism reflected on two sides by eight inches of Lucite. The effect of the cadmium sheet was to decrease the reflector savings of the Lucite by ~ 2.6 cm. The effective reflector savings for the combination of cadmium sheet and Lucite is equivalent to that for a Lucite reflector 0.9 in. thick.

Theory and Methods of Computing Criticality Parameters

A problem for which little data exists, and on which few calculations have been made, concerns the criticality of enriched uranium rods in uranium solution. Problems of this kind are encountered during the dissolution of uranium. Attempts to compute k_∞ for such systems, by means of diffusion theory, have not been overly successful due to the difficulty in computing the resonance capture in the combined uranium solution--uranium rod system.

In this case, it appears that a satisfactory solution can be more easily obtained by means of a Monte Carlo type calculation. A geometry routine is

presently being incorporated in the HISMC (Homogeneous Monte Carlo Code) which will permit its use on the desired lattice cell type calculation.

Critical Mass Laboratory Instrumentation

To determine k_{eff} directly from pulsed neutron source experimental data, without a knowledge of the neutron lifetime, the delayed neutron buildup above the background must be precisely measured. This delayed neutron contribution, which results from the pulsed neutron source, can be obtained by pulsing the neutron source at a high repetition rate. In principle, it can also be obtained by pulsing at a low repetition rate and allowing the time-analyzer to make a multitude of complete sweeps after each pulse. This would result in an "effective" pulse rate equal to the time-analyzer sweep rate times the actual pulse repetition rate. To investigate this latter technique, a system using three signal generators has been developed, and is now being tested, which causes the time-analyzer to make five complete sweeps each time the neutron source is pulsed.

Consulting Services on Nuclear Safety--Criticality Hazards

1. Nuclear Safety in HL

The following nuclear safety specifications were issued in HL:

J-7 - Special Rules for Handling Plutonium in Metal Powder Line (Metallurgy Development)

M-4 - Rules for the Storage and Handling of 1.5 w/o PuO_2-UO_2 EBWR Rods (PRCF)

2. Nuclear Safety in CPD

As per request by CPD, a committee has been formed for an independent nuclear safety review of CPD operations. Plans are to complete the review during March.

Two additional problems have been received which are currently under study; one of these concerns the nuclear safety of a lead shipping cask, and the other, the nuclear safety of processing PRTR fuel elements at Redox and Purex.

3. Nuclear Safety in NRD

A detailed nuclear safety review of N Fuels Processing in the 333 Building for NFD was completed and two documents issued detailing the results

of the study. Document HW-80312 contains the technical bases for the nuclear safety specifications, and Document HW-80313 contains the nuclear safety specifications derived therefrom.

4. Nuclear Safety of Off-Site Shipments

The special shipment of 90 g Pu on a loaded Redwood car was approved (letter from PF Gast to AT Gifford dated 1-17-64). Also approved was the routine shipment of 7.35 w/o Pu-Al alloy fuel rods on the Redwood car (HW-68452 ADD 2, Addendum 2 to Redwood Car Specifications, PF Gast, February 12, 1964).

Separations Instrumentation and System Studies

Fabrication was completed on the solid state portable gamma spectrometer instrument which will be used to provide direct measurement of plutonium in hoods and glove boxes. The scintillation detection instrument, which will monitor the 384 keV photons from plutonium, is now undergoing extensive laboratory testing prior to field application.

Development effort was accelerated on temperature and radiation resistant preamplifiers. The units employ General Electric "TIMM" circuits and are designed for operation at 600°C. A special resistance welder was obtained for assembling the circuits.

Plutonium samples were used with a multichannel analyzer and various multiplier phototubes to determine which phototube and scintillation crystal combination would provide optimum performance for use in the plutonium X-ray sample counter being developed. The counter will be used to measure plutonium content in liquid samples.

NEUTRON CROSS SECTION PROGRAM

Scattering Law Measurements for H_2O at 95°C

The triple-axis spectrometer at 105-KE was put back into operation following an extensive outage and concurrent modification of the spectrometer shielding. Measurements are in progress to determine the basic alignment of the spectrometer. Preliminary measurements have been made at an incident energy of 0.452 eV using a series of filters to study order contamination. These measurements indicate that systematic effects which occurred in the measurement of the scattering law for 95°C H_2O may have been due to order effects. Further measurements will be required to verify this conclusion and provide quantitative corrections to the scattering law data.

Time-of-Flight Spectroscopy for Slow Neutrons

Development of components for the measurement of slow-neutron inelastic scattering by time-of-flight has continued. Modification of the shielding of the triple-axis spectrometer at 105-KE to allow access to the 4-B hole was completed. Additional internal shielding was inserted in the beam-shutter plug in the 4-B hole which resulted in significantly decreased radiation levels. Further measurements were made of electronic drift speeds in He-Co_2 counters and a paper on time-jitter in slow-neutron detectors was submitted for the April Meeting of the American Physical Society. A paper on shielding of slow-neutron detectors was given to the Tucson Meeting of the American Physical Society.

Fast-Neutron Cross Sections

A series of 3-15 MeV total cross section measurements was completed during the month. In this series measurements were made for the second time on samples of 18 elements and separated isotopes. The new 400-channel analyzer designed and built by Instrument Research and Development was put into service for the first time in these measurements. The data obtained in these measurements were obtained for the first time on punched paper tape. Results of previous measurements on seven additional elements were transferred to compilation centers at BNL and LRL - Livermore. Other results were sent on request to Germany, South Africa, and KAPL.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Phoenix Fuel Program

1. Comparison of Thermal Spectrum Models for a Typical Phoenix Core

In a 1959 NSE paper by P. Greebler, et al., Wilkins and Wigner-Wilkins spectra for low enrichment reactors (with and without Pu-239) are compared. It is stated that the Wilkins equation, rather than the Wigner-Wilkins equation, may provide a more suitable model for neutron thermalization in water-moderated assemblies. The use of the Wilkins equation is rationalized on the basis that below the binding energy of the water molecule, the effective scattering mass is much larger than unity. The use of the Wilkins equation for Pu-H_2O media is also recommended by P. G. Aline and J. McWhorter, based on an analysis of some Pu-Al-H_2O approach-to-critical experiments. Recently, Lindenmeier has called attention to the difficulties that can arise with this approach if very large amounts of Pu-239 are present.

Three types of thermal spectra (Wilkins, Wigner-Wilkins, and Nelkin) were obtained for a typical Mark-I Phoenix core containing 80/20 type fuel. If the Nelkin spectrum is assumed to be "correct", the W-W spectrum is a much better approximation than the W spectrum. Some cross section averages are compared in the following table.

AVERAGE THERMAL CROSS SECTION COMPARISON
MARK-I, 70 KG 80/20 TYPE FUEL

Spectrum Model	"Nelkin"	"Wilkins"	"Wigner-Wilkins"
σ_a^{239}	993.26	1230.0	994.4
σ_f^{239}	647.24	775.4	650.9
α^{239}	0.5346	0.5863	0.5277
σ_a^{240}	206.06	202.9	204.9
$\sigma_a/^{240}/\sigma_a^{239}$	0.596	0.446	0.597
σ_a^{241}	814.92	723.0	852.5
σ_f^{241}	615.30	535.4	644.8
α^{241}	0.3244	0.3504	0.3221
σ_a^{242}	9.463	7.837	9.822

The large α^{239} for the Wilkins spectrum results in a low k_{eff} of 1.06218, compared to a k_{eff} of 1.07719 for the Wigner-Wilkins spectrum.

2. Dependence of Phoenix Core Burnup Results on Thermal Flux Spectrum Model

To test the effect of flux spectrum models on the burnup characteristics of Phoenix cores, a series of burnup calculations were carried out for a typical 70 kg, Mark-I core. A comparison was made between Wilkins and Wigner-Wilkins calculations for three different plutonium fuel composites. Since the cores are quite epithermal, the effect of changes in the thermal spectrum are de-emphasized, and the effect on the flux averaged absorption and fission cross sections (over the entire reactor spectrum) is not too pronounced.

The results of the burnup calculations, the required shim control, and the time dependent isotope concentrations are surprisingly insensitive to the thermal spectrum model.

3. Calculation for the Proposed VSTR-Phoenix Experiments

Calculations for a homogenized VSTR core ("thin" plate) have been carried out using the PHYSICS CHAIN code. Three plutonium compositions have been studied in four fuel-to-moderator ratios. The k_∞ values for CH_2-moderated, 20 w/o Pu in Al plates are listed in the following table:

V_{fuel}/V_{mod}	Pu-239/Pu-240 95/5	80/20	70/30
0.25	1.712	1.483	1.374
0.50	1.653	1.425	1.322
1.0	1.594	1.378	1.275
2.0	1.576	1.375	1.269

The preceding values were calculated using the Wigner-Wilkins equation to obtain the thermal cross sections. The same set of cases was run using the Wilkins spectrum approximation. The resulting $\Delta k/k$ cases are tabulated below:

V_{fuel}/V_{mod}	Pu-239/Pu-240 95/5	80/20	70/30
0.25	0.0175	0.0097	0.0045
0.50	0.0279	0.0185	0.0115
1.00	0.0265	0.0205	0.0152
2.00	0.0144	0.0146	0.0142

Critical and Critical Approach Experiments

Low Exposure PuO_2-UO_2 Lattice Studies in the PCTR

The fuel for these experiments is being fabricated by the Ceramics Research and Development Operation. The work has been progressing very slowly due to conflicts with other higher priority jobs. Fuel for the EBWR and PRTR require the use of the same equipment. The core material for this fuel has been densified and now must be ground in preparation for vibratory compaction.

The Critical Facility of the PRP

Process Specifications for the H_2O-moderated PRCF have been reviewed and comments forwarded to the author.

A subroutine for the Generalized Least Squares Program GLEX has been written in order to obtain the parameter β/ℓ which best fits data from a reactor noise spectrum. The value of 6.3 ± 0.6 sec^{-1} has been obtained from the reactor noise spectrum in the D_2O-moderated PRCF using the program.

A summary of a paper has been written and submitted for presentation at the 1964 Annual Meeting of the American Nuclear Society. The paper is entitled, "Noise Measurements and Irradiated Fuel Measurements in a Plutonium Enriched D_2O Reactor." The paper describes experiments in the PRCF using D_2O moderator, and includes startup measurements, kinetic studies, and reactivity measurements of irradiated fuel elements.

In order to interpret the results of the moderator level coefficient measurements more exactly, the effective height of the reactor at various moderator levels is being determined. The coefficient is proportional to H^{-3}, where H is the effective height of the reactor. Data obtained from vertical flux traverses have been analyzed to estimate the points at which the flux extrapolates to zero. These points are also being estimated by calculating the flux distribution using three-group diffusion theory (SWAP code).

Approach-to-Critical Experiments Using EBWR Fuel

Sufficient fuel was available to complete the 0.71-inch lattice that is now in the approach-to-critical tank in the TTR reactor room. All the experimental data have been taken on the approach-to-critical and data are now being taken as the fuel is being unloaded.

Summary of D_2O Systems

The work which has been done at Hanford Laboratories in support of D_2O-moderated reactors since 1958 is being summarized for inclusion in a paper, "Lattice Studies and Critical Experiments in D_2O Moderated Systems," which is to be given at the 1964 Geneva Conference. The paper, to be co-authored with persons from MIT, NDA, WAPD, and SRL, is being co-ordinated by SRL.

Code Development

Transport-Transmutation Analysis

Theoretical Physics development work on nuclear fuel cycles progressed

through formulation and programming of an isotopic transmutation analysis based upon one-dimensional multi-energy transport theory. Debug runs have been completed, and adequacy-evaluation studies with the new deck (GE-HL Program S-XIII) have begun on the Fast Test Reactor design.

Analysis versatility spans high exposure spectral-shift breeders, with unrestricted inhomogeneity, with unrestricted exposure time, and with unrestricted transmutation intricacy (spontaneous and induced, endoergic and exoergic, decaying and multiplying, branching and cyclic) including energy-variant induced-yield fractions.

Program TEMPEST

TEMPEST is being revised to include routines for solving the modified heavy gas equation instead of the usual heavy gas equation. Inclusion of these routines in TEMPEST will allow the calculation of more accurate neutron flux distributions with a corresponding increase in the accuracy of the computed average cross sections. In addition, improvement in the accuracy of the routine which solves the Wigner-Wilkins equation is being attempted.

Computer Program TRANS

Program TRANS, a 7090 computer program for determination of the amplitude and phase shift of the zero power reactor transfer function, has been coded for use in HTLTR studies. A document has been written, HW-80957, and will be available to those who might be interested. The code calculates values of amplitude and phase shift for each period desired by the user and gives output in a tabulated form. This code also utilizes the Benson-Lehner plotter to obtain plots of phase shift vs. period and amplitude vs. period.

PHYSICS CHAIN

Cross sections for sodium, as obtained from the Hanford Basic Library Tape, have been added as nuclide 132 to the GAM file of the Composite Library Tape (CLT) in the PHYSICS CHAIN system. The necessary minor changes have been made in the GAM program used on the PHYSICS CHAIN to insure that the CLT is properly positioned for its next file.

TRIP205

A specialized version of the reactor kinetics code TRIP was prepared for HTLTR studies. TRIP205 computes reactivity by either square or sawtoothed functions. Brief documentation of formulation and input requirements was given limited distribution.

ZODIAC

A ZODIAC Chain Tape with two new features was released: (1) After one full iteration, TEMPEST and GAM may be omitted or selected on all subsequent iterations in order to save machine time, and (2) restart procedures are greatly simplified and improved. Document revisions concerning the new tape were issued to holders of the ZODIAC User's Manual coincident with release of the new tape.

Two further steps in the planned development of ZODIAC are partially completed. The first of these concerns the elimination of many of the previous limitations; this version has responded satisfactorily to testing, but has not yet been released. The second development, now being written, will cut machine time by elimination of the unnecessary retracking through TEMPEST and GAM for nonburnable regions.

RBU Basic Library

The RBU Basic Library is currently being updated to allow ln σ to vary linearly with ln E. Changes to the first ten isotopes of the RBU Basic Library have been completed and plots of cross section data for 53 isotopes have been obtained. The plots of the cross section data are obtained from the Benson-Lehner plotter which uses output from BARNS and a plotting code. The current library tape does not include the changes that have been made.

RBU

With completion of document, "RBU Vol. III: Operation and Procedures," to be issued as HW-80003, efforts shifted to Volume II, and in part to the formulation document, Volume I. The RBU contribution to the PRP status report was coordinated with a general recasting of the preliminary formulation remarks to be included in Volume I.

During initial attempts to execute the three Monte Carlo system representation of the PRTR 19-rod cluster, a mathematically divergent situation was uncovered. The problem concerned the specular reflection of neutrons from the concave side of a curvilinear boundary with an albedo greater than unity. At reflection angles greater than a critical angle, depending on the geometry involved, divergent multiple-splitting is possible, both in neutrons and their supposedly attenuated first-flight beam contributions. Coincidentally, similar difficulties were reported by the British in transport calculations. An adequate solution to the problem for RBU purposes in running the 19-rod cluster has been incorporated and the cluster simulation is now back on the machine.

Theory-Experiment Correlation

Analysis of Pu-Al-H_2O Criticals

The results, as quoted in the January monthly report, for the calculated multiplications of the Pu-Al criticals with the Nelkin kernel, are in error. The kernel used was that for H_2 and not for $H_2/2$. This error became apparent when the total scattering cross section calculated by the Nelkin kernel was compared with measured cross sections for H_2O. All of the Pu-Al loadings were recalculated with the correct kernel (by reducing the bound atom scatter cross section by 2), and the results now compare closely with those calculated when using the Brown-St. John scattering model. In addition, the fast group constants were recalculated using HRG instead of GAM I and used in HFN. The effect was nearly negligible on the calculated k_{eff}'s (at most, 0.5% for any case).

Burnup Experiments

Analysis of the experimental data from fuel element 5111 has been initiated. This is the second of three H_x Pu-Al elements which will be analyzed. The preliminary results obtained from this element, along with the results obtained from element 5108, indicates that the initial enrichment of these elements was 2.62 w/o plutonium rather than 2.57 w/o which is what they were thought to be.

Rods from elements 5109 and 5103 were processed in the PRTR Gamma Scan Facility. The design changes have improved the ratio of primary count to background count rate, but some difficulty still exists in positioning. Also a drift in count rate has been noticed which might be due to instrument instabilities due to insufficient warmup time. Some additional work is needed to make the results useful in burnup analysis.

The burnup analysis of a 1 w/o PuO_2-UO_2 fuel element has been initiated by the obtaining of fuel element 5214 for the burnup program. This element developed a leak at about 50 Mwd and, therefore, is applicable as the first burnup step in the analysis of a 1 w/o element.

A contribution to the Geneva Conference paper, entitled: "Calculation of Criticality, Power Distribution, and Fuel Burnup in Thermal Reactors," has been forwarded to G. H. Minton. The contribution was a summary of the results obtained from the burnup of three low exposure Pu-Al alloy fuel elements exposed in the PRTR.

Isotopic Analysis of PRTR Samples

Isotopic analyses were provided on 32 plutonium samples of PRTR-irradiated fuel elements in support of the Plutonium Recycle Program. Of these, 11 were burnup and 13 macrodrill samples from Al-Ni-Pu fuel element No. 5111, 5 were macrodrill samples from UO_2 element No. 1101, and 3 were macrodrill rechecks from UO_2 element No. 1041.

Instrumentation and System Studies

Plans were formulated for general improvements to the PRTR fuel rupture monitor. Specific investigation and development work will include detectors, scanners, and general instrumentation. A new scanning system will be developed as a demonstration prototype. It has been decided that the 85 sample lines will be divided into 17 headers and that it will be necessary to obtain a gas monitoring sample from each header. The new detectors and scanner will be used with this sampling system.

The detailed instruction and maintenance manual for the PRTR liquid effluent gamma monitors was completed and copies were distributed. In addition, a descriptive report was prepared for the aural signaling monitor which was developed for use with the PRTR automatic reactor controller.

During the latest shutdown of the PRTR, the automatic controller was checked out and returned to operation. Two precision-matched resistors were replaced. Electronic equipment is expected to be available near month end which will permit complete automatic operation of the PRTR from startup to shutdown, with the exception of initial startup ranges.

Additional analog computer runs which simulated a prompt excursion for a uranium plutonium mixture were requested. Using previously furnished equations and parameters, the simulation runs were made and the results forwarded.

EBWR PROGRAM

An analysis of the hazards associated with including special fuel elements (e.g., Pu-Al and UO_2) in the EBWR core has been completed, and the results communicated to personnel at Argonne National Laboratory for inclusion in the over-all hazards report.

HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

The fourth in a series of high temperature tests of materials in nitrogen was shut down after 75 hours of operation at 1200 C, because of a leak which developed in the gas outlet line from the containment vessel. The

major additions to this test were some large samples of brick insulating material that is planned to be used in the large mock-up assembly for testing full scale heater rods, control rods, and other reactor components. The brick appeared to be the source of the corrosives causing not only the leak in the line but some severe changes in some of the metal samples. The severe effects were confined to the metal samples actually in contact with the brick. Others, held in contact with graphite, were not affected. A more detailed examination of the samples is in progress.

A second heater element, consisting of a 4-1/2 foot long graphite rod, 1-5/8 inches in diameter, was inserted in the small HTLTR mock-up test assembly. This heater operated for about 500 hours in the temperature range 900 to 1600 C in the nitrogen atmosphere. This is about the length of time that the heaters in the HTLTR are expected to be used over a period of six months. The test assembly has been shut down so that the element can be examined at this stage.

There are two mechanical oscillators included in the HTLTR design--one for moving large samples, up to the size of a complete lattice cell, into and out of the core region; and the other for moving smaller amounts of material. The oscillators will thus be an important tool for the measurement of reactivity changes in the course of experimental work done with the HTLTR. For the purpose of initiating work on the final design of these devices, some of the mechanical operations that will be expected of them are now being worked out in greater detail than it was possible to give in the design criteria.

One method of measuring the relative reactivity coefficients of materials in the HTLTR would be to oscillate a sample of the material in the reactor. The neutron response of the reactor to sinusoidal variations in reactivity may be estimated from the reactor transfer function. Computer Program TRANS, HW-80957, has been written to aid in these calculations. TRANS calculates the proportionality constant and phase shift between the reactivity oscillation and the neutron density response as a function of frequency of oscillation.

The proposed design of the neutron chopper has been described in a document, "The HTLTR Time-of-Flight Neutron Spectrometer," HW-80540, issued during the month.

NEUTRON FLUX MONITORS

Analysis of the mass spectrometer data obtained for the plutonium regenerating detectors indicates that Westcott cross sections are of limited value in describing the behavior of such detectors. Using the experimental values

of r and T, the measured sensitivity (fission cross section) of the regenerating detectors deviated from the calculated values by as much as 5%. Although values of r and T could be postulated which would more closely fit the data, a skewness exists which makes it unlikely that a consistent set of parameters can be found which will describe the data to within the experimental uncertainty. The following tables summarize the results obtained to date:

TABLE I

INTEGRATED NEUTRON EXPOSURE BASED ON URANIUM COMPOSITION

Sample	Ratio 236/235	Integrated Exposure, ϕt ($n/cm^2 \times 10^{20}$) Experimental r,T	Arbitrary r,T to fit Pu data
Original	0.00217 ± 0.00005	Zero	Zero
1	0.03175 ± 0.00015	2.45 ± 0.025	2.48 ± 0.025
2	0.05535 ± 0.0003	4.155 ± 0.03	4.21 ± 0.03
3	0.1050 ± 0.0005	7.22 ± 0.05	7.32 ± 0.05
4	0.15165 ± 0.0008	9.635 ± 0.065	9.75 ± 0.07

TABLE II

PLUTONIUM REGENERATING DETECTOR SENSITIVITY AS A FUNCTION OF EXPOSURE

Sample	Ratio 241/239	Fission Sensitivity, Relative: Experimental r,T Measured	Experimental r,T Calculated	Arbitrary r,T to fit data Measured	Arbitrary r,T to fit data Calculated
Original	0.0055 ± 0.00005	0.850*	0.850*	0.891*	0.891*
1	0.3896 ± 0.0020	0.936	0.870	0.980	0.980
2	0.6835 ± 0.0035	0.949	0.998	0.992	1.000
3	1.3035 ± 0.006	0.922	0.972	0.962	0.963
4	1.872 ± 0.009	0.871	0.913	0.907	0.903

* Normalized

The uncertainty limits in the exposure are approximately one-third due to the uncertainty in the mass spectrometer ratios and two-thirds due to uncertainties in cross sections, primarily for U-236. The uncertainty in the measured relative fission sensitivity is estimated to be less than 0.5% in each case.

It is concluded that the experimental-calculational technique developed for optimizing regenerating detector composition for any reactor environment, although adequate, is limited by the spectral assumptions made in the utilization of Westcott cross sections. A report on this work will be prepared as soon as the final two data points are obtained and included in the analysis.

The offsite bid for fabrication of two neutron chambers using uranium enriched U-234 as the regenerating coating material was reviewed and it was concluded that the bidder had adequate fabrication techniques at hand. Approval was given to proceed.

Developmental progress was achieved on the boron-11 beta-current generator neutron flux monitor concept. Two experimental detectors were irradiated during the month. The major change incorporated in the first detector was the evacuation of the entire assembly using a vacuum pump during the test. For this detector, the boron-11 element and an extra lead wire were used to measure both the element current and the residual lead wire current. The detector was charged in the facility in KW Reactor and for the first time in this series of tests, the measured currents were stable, repeatable, and analytically explainable. The cable "noise" current was two orders of magnitude below the measured "noise" currents of the previous tests; however, this test was unexpectedly terminated when the boron-11 element shorted due to the vibrations of the device in the facility water stream.

An effort was made to fabricate a more rigid second prototype device by centering the electrical conductors in the tube by using steel spring brackets to hold the insulators. However, the shape of the end plug and the high velocity water (40 gallons per minute) resulted in such turbulence and vibrations that lead shorting again took place. As a result of these two tests, however, the uncertainties about the detector have essentially been eliminated. The only remaining problem is one of development and fabrication of a mechanically rigid device.

For the microwave neutron flux monitoring experiments, design was completed and fabrication started on the specific high voltage power supply controls for the reflex klystron. Design modifications incorporated include a solid state over-current protective relay circuit and appropriate battery supply sources for the klystron reflector and control grid. Design was completed on the mounting bracket with cooling fan for the klystron. Because of retubing work, the reactor tests were rescheduled.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

Development of the multiparameter tube tester is proceeding. A laboratory test arrangement is being assembled using a portion of the previously developed multiparameter tester and two Hanford Model 1004 tube testers. The first laboratory demonstration of the multiparameter tube test will be made using an internal test coil of the size appropriate for testing N Reactor process tubes. This test will have four parameter capability.

Parts for four new multidimensional (three degrees of freedom) calibration adjustment devices were received from the shop and are being assembled for incorporation in the multiparameter tube test equipment. An alternate, more compact design is also being evaluated, and the machine work on one unit of this type is completed.

Electrical noise measurements are being made on a Model 1004 tube tester for future design use.

Small "point contact" coils (plane of coil normal to test surface) are being evaluated for use in tube tests. A 0.004 inch deep by 0.125 inch long I.D. notch was detected in a 0.34 inch wall Incoloy tube using two 0.2 inch diameter by .016 inch wide differentially connected point contact coils on the outer tube surface. This same notch was not detected with an encircling test coil assembly due to background noise caused by the tube material.

Zircaloy-2 Hydride Detection

The problem of scanning the full inner circumference of each N Reactor process tube is expected to be a serious limitation to the existing eddy current hydride detector. In order to facilitate more rapid scanning, a novel test coil is under development which inductively couples the test signal from a stationary primary winding to a rotating secondary winding. This system was experimentally demonstrated in principle with laboratory tests in which a 455 kc signal was coupled through the transformer while the secondary was rotated by hand. With this arrangement hydride signals from standard samples were equivalent to those obtained with conventional test probes.

Efforts are being made to improve the circuitry of the eddy current tester. Compact battery powered transistor oscillators to drive both the 455 and 44 kc bridges have been developed. Used in conjunction with the ultra-stable circuits, the system will provide a portable instrument that would be convenient for field use. The fact that a battery powered instrument is independent of power line transients and ground loops is also an advantage.

Fundamental Ultrasonic Studies

Ultrasonic attenuation measurements were continued. Longitudinal wave attenuation coefficients for an aluminum sample were measured at ultrasonic frequencies ranging from about 3 Mc to 25 Mc by using the Fresnel field method discussed in the previous monthly report. The coefficients were found to vary approximately as a linear function of frequency over the measured range. The 3 Mc frequency was found to be the lowest frequency having a measurable attenuation in this aluminum sample. This represents a low frequency cutoff which may be difficult to include in the analytical studies on wave behavior. The analysis will become more difficult for ultrasonic pulses which have significant frequency components near the cutoff. For pulses having less low frequency components the presence of the cutoff will not be a problem.

The analysis of ultrasonic wave behavior in the presence of attenuation and Fresnel field diffraction includes studies of broad frequency band pulses as well as the narrow band pulses used for the aforementioned attenuation studies. The frequency spectra of broadband pulses must be measured to determine how attenuation affects the pulse shape. It is also of interest to study how the pulse propagates in the absence of attenuation in order to make clear any shape changes due to diffraction effects.

An exceptionally broadband pulse was generated in water to examine the influence of diffraction. This was accomplished by driving a thick, one-inch diameter barium titantate transducer with a broadband electrical pulse. The pulse had a duration short compared with the wave transit time between the faces of the transducer, thereby suppressing resonant frequency oscillations. Though the frequency spectrum of resulting ultrasound wave was not measured, the waveform received with a small probe was qualitatively observed to contain an exceptionally wide band of frequencies.

To observe pulse changes which were largely a function of the Fresnel field effects, the receiver probe was kept at a constant distance from the transmitter and traversed parallel to the wavefront. The measured wavefront contour was observed to have a concave shape in the center and a convex shape at the edges. Also, as the probe was moved to the beam edge, the higher frequencies became more dominant. Continued lateral movement resulted in a frequency reduction until the pulse amplitude decreased to an undetectable level.

Since the distance between the transmitter and receiver was kept constant the measured effects were assumed to be a function of Fresnel field diffraction. A qualitative explanation was worked out which apparently verifies that the observed changes were in fact due to wave diffraction effects.

In essence, the premise holds that the diffraction causes spreading of the ultrasonic beam with low frequency components spreading more rapidly than high frequency. The center of the beam contains a higher percentage of low frequencies than does the beam edge. As the waves propagate further, the low frequency components continue to leave the beam due to spreading and thus the pulse becomes richer in higher frequencies. A competing attenuation effect exists, however, since the higher frequencies attenuate more rapidly. The net result of these two frequency dependent effects cause the pulse shape to change according to the relative magnitudes of the two competing processes.

The algebraic form of the frequency equation for Lamb waves in a hollow cylinder was derived and one calculation was made of a practical case. From this it was concluded that, for a given frequency, the wave length of Lamb waves in a cylinder does differ appreciably in some cases from the length of waves in a flat plate for the sizes of cylinders of most interest, and that the equation with just a first order correction term gives accuracies of one percent or better for most practical cases. To state these conclusions in terms of specific numbers will require many more numerical calculations, which can only be done economically on a computer, such as the 7090. It may be possible to tackle the problem analytically, but this will require further study.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

The prototype sheath tubing tester was set up to test 65 mil wall, 5/8 inch diameter sintered aluminum tubing received from AECL. Two sets of notches, three O.D. notches 1.5, 2.5, and 3.5 mil deep by 15 mil length and one I.D. notch 2.5 mils deep by 15 mils length, served as notch standards for each test. Both the circumferential test for detecting circumferentially oriented defects and the axial test for detecting axially oriented defects were set up to detect the 2.5 mil notches on outer and inner surfaces with equal amplitude. The 1.5 mil notches were then detected at 1/2 amplitude while the 3.5 mil notches were detected at 1.5 of the 2.5 mil amplitude. Each tube tested had several defect indications at or below the two mil level. Two tubes had indications well above the 3.5 mil level. These defective areas were located on the inner surface and had been detected by a previous test performed by AECL. The only correlation observed between the AECL and Hanford tests was obtained on those defects which had lengths greater than 50 mils.

Experience gained while setting up the prototype tester to test various thicknesses of tubing was also factored into the final report data and

equipment. Checkout of the final electronics is nearing completion while fabrication of the mechanical system is continuing. Methods of providing easy set up or alignment procedures were developed and incorporated into the electronics and mechanical system.

Acquisition of data for the final report is nearing completion. One mil depth by 15 mil length notches can be detected in Zircaloy tubing from 10 mils to 65 mils in wall thickness. The data are complete for the 17 through 65 mil wall tubing. By adjusting the focal point of the 3/16 inch diameter, spherically-focused transducer to different points inside the tube wall, both I.D. and O.D. notches are detected with equal amplitude. The one mil notch amplitude in the 65 mil wall is down 25% from its amplitude in the 17 mil wall; however, these notches are detected well above the tubing noise level. Almost identical results are obtained on axial and circumferential oriented notches. The axial notches are detected with the transducer aligned to generated shear waves propagating around the circumference of the tube, while circumferential notches are detected by shear waves propagating down the axis of the tube. The best amplitudes are obtained when the ultrasound entry angle is adjusted to generate shear waves at 45° within the tube wall. Due to the curved surface on which the sound must enter to detect axial flaws, the entry angle is slightly under (about 3°) the entry angle calculated to generate 45° shear waves. However, the entry angle for the circumferential flaws is 24°, the calculated angle.

Experiments with a 15 mil wide beam on the 17 mil wall tubing and an 8 mil beam on 10 mil wall tubing has shown that if the beam width is one-half the wall thickness, a minimum of interference is present and unambiguous signals are obtained from even the small notches. When the beam width approaches the wall thickness, the notch depth vs. amplitude curves degenerate and it becomes progressively harder to electronically gate these distorted defect signals. Small non-symmetrical conditions present in the ultrasonic beam when the beam width is equal to the wall thickness further distorts the return signal. The 3/16 inch diameter focused transducer provides an 8 mil diameter beam which is sufficient for testing tubing down to 15 mils in wall thickness. Since efforts to obtain a beam smaller than 8 mils have been unsuccessful, a dual lens which provides a symmetrical beam was machined onto the focused beam transducer. This dual beam transducer, which was described earlier in an invention disclosure (HW-77778), compensates for edge effects and lens misalignment in the transducer. Using the dual lens transducer, one mil notches are detected in tubing down to 10 mils in wall thickness. Although the received signals are somewhat distorted, they can be gated electronically and then reliably detected.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Further investigation of the relationship between the area enclosed within a dosage isopleth and the value of the dosage isopleth progressed during the month. All data from ground source dispersion experiments conducted between 1959 and 1963 are included in the study. By dividing the data into three atmospheric stability classes according to the local Richardson number, the data stratified well with stability. Least square regression lines of area enclosed versus exposure level were calculated and compared with those found for Green Glow and Prairie Grass data. It was found that the normalizing wind speed used had an important effect on the regression analysis, and may be the cause of discrepancies between the earlier and current analyses. Plotting and analysis were started on the data collected at Vandenberg Air Force Base in which the Hanford technique was employed. These data will be examined for effects of topography, vegetative cover, and climatic regime in relation to the area problem.

Using the real time and bulk samplers for fluorescent particles, and diazo paper techniques for liquid droplets, a field test was conducted to ascertain the extent to which rain-out of formulation from our aerosol generators may affect results of the diffusion studies. Smoke and zinc sulfide particulates were released simultaneously from the 83-foot level of the Meteorology Tower into an atmosphere at 50 percent relative humidity. Some droplets and large particles were found at the ground level sampling locations. However, comparison of all sample methods leads to the conclusion that the rain-out effect is at least of second order when one considers the current state of knowledge of the dispersion problem. For more detailed field studies than are presently scheduled, more complete knowledge of the droplet rain-out will be needed.

In the detailed atmospheric turbulence studies, investigation into the applicability of the analog computer program for noise analysis to analysis of wind component meter data was started. Preliminary indications are that the program can be used for studying the change of turbulence power density spectra with time, as well as for determination of the spectra and cross-spectra. Adaptation for the noise program for turbulence analysis would greatly expedite the turbulence studies.

Micrometeorological instrumentation on both the portable mast and the tri-level tower required additional trouble-shooting. Temperature difference sensors on the tri-level system showed a consistent error at one height interval, the cause of which has not been determined. The defective digital voltmeter from the portable mast system was returned to the vendor for repair.

Radiological Physics

The results of the whole body counting at Anaktuvuk Pass last month became available. Seventy-one Eskimos were counted. Their average body burden of Cs^{137} was about 30% lower than it was the summer before. There is a shortage of caribou at the village this winter, but we do not know if this shortage or if decreased amounts of Cs^{137} in the caribou caused the decrease in the Eskimos. The amount of Cs^{134} in the Eskimos is still decreasing.

An opportunity arose to make a new observation of the effect of eating moose meat on the body burden of Cs^{137}. We counted a visitor from Alaska who eats more moose than any other meat. His body burden was very nearly the same as that of the Indians at Fort Yukon. In his case there is no caribou in the diet that might have been the source of the Cs^{137}.

Twelve of the males counted at Anaktuvuk Pass also provided urine samples. These were counted at Hanford. The relation between the amounts of Cs^{137} in the body and in the urine was the same as found in previous tests. Some previously unobserved photopeaks appeared in the gamma-ray spectrum of the urine. We identify them as from Na^{22}. Radiological Chemistry has observed even smaller amounts of Na^{22} in urine of Richlanders. Investigation is continuing to see what information about fallout can be obtained from this new isotope.

The beta-ray coincidence counter developed for P^{32} in-vivo used one scintillation counter and one proportional counter. In an attempt to improve it still further two proportional counters were used. This did not help; the background actually became much worse.

Studies with the plutonium counter were aimed at determining how much of the background came from high energy gamma rays emitted from the subject's body. Different concentrations of K^{40} were used in a plastic phantom. Apparently contamination of the counter occurred at one time. Background increased by a factor of about three. A 33 KeV photopeak appeared. The increase in background disappeared with a half life of slightly more than nine hours. The contamination may have been due to Xe^{135} resulting from a fuel rupture at PRTR.

A new accelerator tube made of glass with higher voltage-breakdown strength appears to have solved most of the difficulties that have been appearing in Van de Graaff operation for almost two years. The Van de Graaff was opened for maintenance three times during the month, but over-all performance was good.

When filled with Xe-CO_2, the neutron spectrometer proportional counter had good resolution. Addition of hydrogen made higher operating voltage necessary and increased the noise level. The added noise is believed to be coming from a voltage breakdown. Apparatus for the pulse shape discrimination of gamma rays was completed. Fabrication of the He^3 counter was completed.

Calorimetric measurement of one of the Pm^{147} sources was completed.

Tests of a new (borrowed) control system for the new calorimeter showed it to be slightly better than the old system and very much more reliable. An improved heat shield was fabricated and is being tested.

The resident University of Washington graduate student designed a shield and exposure facility for their neutron generator, specified and obtained the materials for it, had the necessary structures fabricated, and is now assembling it.

Instrumentation and System Studies

Field tests were conducted on the animal physiological function telemetry system with the temperature monitor portion of the system calibrated against thermometers provided by Biology Laboratory personnel. The telemetered data of sheep rectal temperatures were determined to be accurate to ± 0.25°C. Minor difficulties in data reception occurred due to the confinement of the animal in a metallic cage; this affected the signal transmission and changes were initiated to alleviate the difficulty. Steps were taken to separate the temperature-measuring thermistor from the oscillator section to provide a smaller probe. Experimental work was continued on the respiration rate and blood pressure transducers and it was requested that a fourth channel be added to the system to provide a measurement of dose to the thyroid.

Successful experiments were carried out on the special sliding valve and associated instrumentation to be used in inhalation studies at the Biology Laboratory. The equivalent of three inhalation experiments were conducted with the noise level being held at an acceptable level. In addition, a new tidal volume monitor was completed, calibrated, and installed for field testing.

All commercial components were received and the detector head assembly was completed in fabrication for the alpha energy analysis air monitor, which will be employed during radionuclide inhalation experiments. Tests were carried out on the detectors to be used.

General design was completed on all circuits for the special radiological spectrometer monitor to be employed in biology field experiments. Fabrica-

tion was initiated on remaining portions of the instrument.

The first stage circuit for the peak pulse reading instrument, which will be employed in dosimetry studies, was completed and tested satisfactorily. Development of the other required circuitry continued.

Significant progress was achieved on a pulse rise time discriminator which will be utilized with proportional counters. Level comparison and delay detection circuits were developed and general fabrication was partially completed on a prototype instrument. Test results on the portions completed were satisfactory.

Development and fabrication work was essentially completed on the wind component meter system to be utilized in atmospheric physics studies. All developed circuitry performed as required and field test results appeared to be satisfactory. In addition, fabrication was completed on the special test function instrument which will be employed in testing the remote data stations of the HAPO Radiotelemetry System. The test function instrument performed correctly in laboratory tests.

Promising results were obtained in tests of several commercial multiplier phototubes which are scheduled for use in a single electron counting system to be used in dosimetry studies. The required high frequency components for the system circuitry were received.

Prototype fabrication was approximately 75% completed on the experimental six-decade logarithmic response radiation monitor. The monitor employs a scintillation detector and all solid state circuits.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples at a reduced rate during the month due to malfunctions in the mass spectrometer ion-detection system.

Studies of the operational characteristics of the mass spectrometer equipped with the new scintillation-type ion detector were curtailed when it was determined that the output pulse of the detector did not have the proper form to satisfactorily operate the 10-megacycle scaler. Work is in progress to develop a suitable output pulse. The vendor has still been unable to deliver the basic parts of the vacuum-lock sample-changer. A new ion source to be used with the sample-changer was assembled.

EXPERIMENTAL REACTOR PHYSICS FACILITIES

PCTR Operation

Operation was on a limited basis due to an extended maintenance outage. There were no unscheduled shutdowns. Two sets of mica detectors were irradiated to measure uranium penetration into canned fuel welds. One set of foils was irradiated for normalization purposes.

TTR Operation

The TTR was operated on a two nights a week basis for the University of Washington Graduate Center. There were no unscheduled shutdowns.

Subcritical Facility

The critical approach experiment, with water-moderated EBWR fuel rods in a 0.71 inch lattice, was partially completed during the month. The amount for criticality is indicated to be 462 rods, based on incomplete data. Exponential data were obtained at a loading of 253 fuel rods.

HTLTR Status

Design work on the HTLTR was continued by Vitro Engineering Company. A preliminary design electric heater assembly was operated successfully by Mechanical Equipment Development Operation. Work was started on the Reactor Safeguards Report.

VSTR Status

A construction project data sheet for the Variable Spectrum Test Reactor was prepared by Facilities Engineering and was submitted to the Atomic Energy Commission. The requirement of complete containment has increased the cost of the building significantly.

COMPUTER FACILITIES

The first tentative schedule for connecting the G.E. 412 digital process control computer to the PRTR was drawn up. Most likely, modifications will be required, but at the present time the schedule takes into account the procurement of additional equipment for both the computer and for the reactor. For the computer, the additional required equipment includes a 4,000 word core memory and a 16,000 word drum memory, cold junctions for thermocouples, three scanning amplifiers for increasing the scanning speed to

approximately 75 points per second, and an input-output register to allow the two typewriters to simultaneously log input information to the memory at the ame time output data is being logged. Some forty-three pneumatic-electric converters are needed to condition available PRTR signals before they are sent to the computer. The schedule also takes into account time for laying out and the actual wiring of interconnections between the computer and the control room, and time for programming the computer.

This effort has been materially assisted by a two-week training session at the G. E. Process Control Computer Plant in Phoenix, Arizona. The session covered programming a sample problem and then debugging the program on the computer. Time was spent at the computer console in debugging the problem and becoming familiar with the operation of the computer.

Application of the iterative capabilities of the new analog computer were tested in the development of a demonstration problem in which two parameters were automatically optimized at high speed. A model curve was produced by solution of a first-order differential equation which had two variable parameters. A similar first-order differential equation was solved in which it was desirable to automatically adjust the two parameters to give the best fit to the model curve. The tests were completely successful.

Preliminary investigation of the use of the iterative techniques on the new analog computer was studied to determine the feasibility of solving multiple integrations. It was found that with the existing equipment, multiple integrations can be easily determined for three variables.

The analog computer utilization was as follows:

EASE 1132 (Old Computer)	EASE 2133 (New Computer)	
150	16	Hours Up Time
22	0	Hours Scheduled Downtime
4	0	Hours Unscheduled Downtime
176	16	

Problems considered during the month were:

1. N Reactor Primary Flow Simulation.
2. N Reactor Secondary Loop.
3. PRTR Prompt Excursion.
4. N Reactor Cold Water Injection.
5. PRTR Hazards Analysis.
6. Two Parameter Optimization.
7. Multiple Integrations.

The new analog computer has been placed in limited service as of February 10. The acceptance testing is approximately 90% complete. The remainder of the tests will be completed on swing shift.

CUSTOMER WORK

Weather Forecasting and Meteorological Services

Meteorological and climatological consultation services included 1) review of the 233-S ventilation system release limits for RPO, 2) continued work on the N-reactor accident calculations, 3) a study of minimum temperature expectancies for application to heating problems of on-site shipments of sensitive materials, 4) compilation of air density extremes at Hanford and other nuclear sites for inclusion in a Plutonium handbook, and 5) compilation of selected Hanford climatological data for use in relation to the Hanford diversification program.

River flow and temperature data for the Columbia River for 1963 were compiled and a diagram prepared for publication. All 1963 climatic data for the Hanford area were compiled for the statistical report. At the Commission's request, wind data for February 28, 1961, were sent to a Kennewick law firm for use in a wind damage hearing.

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

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	87	82.9
24-Hour General	58	85.5
Special	147	82.3

February marked a continuation of the warm and dry conditions which have prevailed since last July. During this six-month period, temperatures have averaged substantially above normal in every month except December and precipitation has totaled substantially below normal in every month except November and December. Total January-February precipitation this year was only 0.38 inch, the lowest of record for the two months.

Mass Spectrometry

Isotopic analyses were provided on 9 samples of uranium in support of HAPO U^{233}-production studies.

Instrumentation and System Studies

Laboratory test results were satisfactory on the solid state count-rate integrator developed for use with plant portable neutron dose rate instruments. The integrator, which provides a register indication of accumulated dose, was developed for Radiological Development and Calibrations, HL.

All fabrication and system assembly was completed on the U-235 fuel enrichment measuring instrument designed for Metal Fabrication Development, HL. Considerable laboratory testing was carried out and system operation appeared to be satisfactory.

Engineering assistance was rendered to Waste Calcination Demonstrations, HL, regarding a general testing program for ion chambers and picoammeters to be used in the 324 Building.

Engineering improvements were incorporated in the timer of a resistance welder for Advanced Fuels Development, HL. Modifications were made to provide a preheat cycle before welding was initiated. Further work is contemplated.

Successful field testing was achieved with the neutron detection soil moisture measuring probe developed for Chemical Effluents Technology, HL. Two wells were logged repeatedly and the results appeared to be fully satisfactory.

A bid by Control Data Corporation for a complete system of temperature comparators to be used as out-of-limits detectors at the 100-KW test facility was reviewed and accepted.

The General Electric GEMAC temperature control system, purchased on a rental basis for evaluation and testing with option to buy, has been obtained and checked out. The system is of interest because of a unique "reset-unwinding" circuit which helps prevent extreme overshoots during automatic startup or return to setpoint after a long deviation. Temperature overshoots cannot be tolerated in the uranium swelling experiments, yet they have occurred on occasion because of large system perturbations (reactor scram, heater trip and reset). The GEMAC system is being evaluated in this respect.

Costs and methods of encoding the position of a shaft located in a high temperature, high radiation environment were scoped for Reactor Metals Research. This was part of a study of the creep capsules being prepared to determine their adaptability to various reactors.

The creep data logger was installed at the 100-KW test facility and has operated a total of 192 hours to date at the facility with no failures. The system is operating off-line interrogating a dummy capsule at a log interval of 0.1 hours. Plans are to place the system on-line at the next capsule charging.

Some thought was given to the remote displacement measuring system in regard to improving resolution and accuracy. The diffraction pattern produced by the opaque flags mounted on the specimen provide maxima and minima whose locations do not change with intensity. By detecting these maxima and minima and gating at the same maximum, intensity changes and threshold detector jitter would have little effect on accuracy or resolution. These theoretical considerations will be further investigated in actual experiments.

Several prototype isotope heat source containment cells were nondestructively tested for wall thickness. The right cylindrical shape's wall thickness was measured on both the ends and around the circumference. The measurements were performed with a focused ultrasonic transducer at several points along inspection paths which were plotted or mapped. Both electrical and mechanical equipment is being designed so that these thickness measurements can be performed and recorded on the entire surface.

Since barium titanate piezo-electric crystal are capable of operating at higher temperatures than other piezo-electric crystals, they become the most eligible candidates for performing the required nondestructive tests on hot fuel cells. These ceramic crystals also are quite versatile in that they can be molded into any shape desired. A series of these units has been purchased. Studies on the effect which ionizing radiation may have on the crystal, epoxy backing number, and connecting cables will be performed.

Optics

Work continued on the design of a traverse mechanism for measuring distortion in N Reactor process tubes. The design is complete except for minor modifications and additions which will be made as necessary. Fabrication is nearly complete; assembly will require an additional two weeks.

A device employing a lamp, two lenses, and a photocell is being developed to monitor 0.3 inch displacements in rapidly heated (50 μsec) metal strips. Tests indicate the device will be accurate to 0.005 inch for displacements from zero to 0.1 inch, with the accuracy decreasing to 0.050 for maximum range measurements.

A study was made of possible methods of measuring surface temperatures of a rotating stainless steel surface in an intense gamma field. Conclusions documented in Physical Measurements Memorandum 64-1 suggested that: (1) surface temperature could be monitored with an infrared radiometer arranged so that the radiometer is outside the cell and shielded from the gamma field but views the surface through a periscope, or (2) the surface temperature could be monitored with an evaporated thermocouple placed close to, but not in contact with, the surface.

Tests were run to compare radiometer performance with the performance of standard thermocouples. Preparations were also made to fabricate evaporated thermocouples. As the radiometer would be sensitive to changes in emittance of the surface, a means of decreasing this emittance dependence was tried in which a spherical reflector was placed over the surface. The reflector contained a small hole through which the surface can be viewed. Tests showed that the emittance was greatly increased by this method, but the correction is not complete for surfaces with low emittance.

During the four-week period (January 19 to February 16) included in this report, 480 man hours shop work was performed. This work included:

1. Repair of one tank periscope for CPD.
2. Fabrication of one displacement measuring device for Plutonium Metallurgy.
3. Aluminizing 10 mirrors for various customers.
4. Service four crane periscopes for Purex.
5. Fabricate laser polishing jib.
6. Fabricate four three-way rheostat yokes for Physical Measurements.
7. Service one underwater viewer at 105-F.
8. Ream one four-inch diameter pyrex tube.
9. Fabricate components for the NPR traverse mechanism.
10. Adapt a 16 mm motion picture camera for use with a Questar telescope.
11. Install one underwater viewer at 105-KE.
12. Aluminize 12 steel plates.
13. Repair one camera shutter.
14. Evaporate boron on accelerator targets for Radiological Instrument Development.
15. Machine six alundum cylinders.

Physical Testing

The final shipment of Zircaloy-2 tubes for retubing the K reactors has been made. Inspection of approximately fifty tubes, scheduled for inventory, will complete the program. All tubes have received the special controlled

grit-blasting surface conditioning process.

Welds on the recently installed lifting lugs of Steam Generator 4-A were inspected by the magnetic particle method as quality assurance for the generator retubing program. Magnetic particle inspection was also performed on hoist equipment in 100-B, and at 100-H to determine the safety condition of the critical components. Magnetic particle testing was also used as a check on the quality of the weld bevels on two sixteen-inch pipes at 109-N as well as for determining incipient cracking of the Vanstoning dies used to install process tubes in the reactors.

Radiography was utilized as the inspection technique for the requirements of code compliance on welds on process water lines, inner and outer shells of tanks, steam lines and condensate lines throughout the plant, on both new construction and during maintenance.

Detection of the uranium contamination of fuel element welds and brazes continues to be a problem; laboratory methods being investigated include neutron tracks in mica or plastic, alpha counting, and autoradiography. Of a series of 14 fuel elements tested in conjunction with known standards, good correlation is obtained between the three methods. Neutron track determinations apparently offer the most quantitative technique, but are the most difficult to apply. Autoradiography is the least sensitive.

Equipment for the mechanical punching of standard notch defects in the walls of tubing has been developed to an acceptable degree. By this method, notches can be quickly and accurately punched in either the outside or inside wall of tubing. Tolerances of plus or minus 0.0002 inches have been routinely obtained. Benefits of the new method are a thousand to one decrease in fabrication costs and greater notch reproducibility.

Scoping of the examination facility in support of the Waste Solidification Engineering Development Program (WSED) was accelerated this month. A considerable amount of exploratory work was conducted in an effort to determine the applicability of potential testing methods to such measurements as wall thickness, surface temperature, content homogeneity, internal pressure, and content-to-pot bond. As a by-product of tests which were conducted with X-rays and ultrasound, on various samples of a proposed glass storage media, considerable information about the samples has been provided the customer. Assistance was also rendered in the preparation of a testing specification to be used for the procurement of the storage pots. An interim status report is being prepared detailing the results of the exploratory work.

The Eddy Current Motion Analyzer (ECMA-1) has been turned over to the customer for their use in studying the fuel vibrations which have been encountered in the PRTR. The instrument is presently being used for ex-reactor studies in the EDEL-1 loop in conjunction with the special nine-teen rod, sensor equipped, fuel cluster which has been completed. Work remaining on this project includes the writing of an instruction manual and a formal report.

Performance tests are now under way on the capacitor discharge equipment for rapidly heating metallic specimens by discharging electrical energy through them. Tests on the insulating materials were made to determine their breakdown potentials. Of the various materials suitable for use, sheets of teflon offered the best insulation at the high kilovoltages required for the system. Inductors, with a low value of resistance, are being designed and used as the method of control for the rate of discharge.

Ultrasonic determinations of the elastic properties of metal at high temperatures have been limited by the shear waves present in the metallic rods used to transfer the energy between transducer and specimen. From other investigators, it was learned that serrating, or threading, the rod eliminated the shear waves. This has been confirmed by using the aluminum rod. The titanium rod necessary for higher temperature measurements is also being modified.

Ultrasonic testing of Zircaloy-2 fuel sheath tubes has continued on a two shift schedule. A thousand tubes, varying in size from .372 to .690 inches in diameter, have been tested this month. A defect level of 0.001 inches in the tube wall is the basis for rejection.

Development of point contact eddy current coils continued. Point contact coils offer the advantage of substantially increasing the testing sensitivity in Incoloy when compared with encircling coils. The most promising results are achieved with coils consisting of 100 turns of 0.002 inch wire wound to a diameter of 0.200 inch and a width of 0.016 inch. Sensitivity for defect detection in Incoloy is increased by a factor of five with this coil.

INSTRUMENT EVALUATION

An instruction and maintenance report was prepared and distributed for the scintillation alpha-beta-gamma hand and shoe counters of which seven are ready for plant service and three are being tested and calibrated. Efforts continued regarding the training of personnel on the maintenance methods to be employed with the instruments.

Evaluation tests of hand and shoe counter detection probes with Lucite light pipes showed a beta counting efficiency of 6% versus about 3% obtained with the present air light pipe approach. The results corresponded to experimental results obtained several years ago; at that time, economic considerations outweighed the performance improvements achieved using machined Lucite light pipes.

Continued successful field testing was achieved on the experimental gamma background compensated beta-gamma hand and shoe counter now being used in the 321 Building.

Evaluation testing was initiated on a commercial portable radiation survey instrument just received. The instrument uses an air proportional probe for alpha detection, a G.M. tube for beta-gamma, and a BF_3 proportional counter for neutrons. Initial test results were promising.

Evaluation tests were completed on a commercial radiation detection instrument which employed solid state detectors. The results indicated that the instrument would be unacceptable for use at Hanford.

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CHEMICAL LABORATORY

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Overheated N-Reactor Fuel Study

Study of macro-photographs and the moving pictures taken during the circumferential localized heating test performed in January supported the direct observations made during heating. Sections at the heated zones showed swelling and complete separation from the uranium of the cladding with no melting, confirming the validity of the heating test.

Preparation neared completion for the fuel-element heating test. Some delay is resulting from difficulties in getting the new 50 KW induction furnace in operation.

Disposal to Ground

A two-dimensional, conductance-paper analog study was completed defining the minimum ground-water travel time from the 1301-N crib to the Columbia River. The purpose of the study was to evaluate the effect that surface springs, which will likely develop along the riverbank at 100-N Area, will have on the minimum travel time. Results indicate a calculated minimum travel time of about 12 days; however, the actual minimum travel time, based on a safety factor derived from a study of a similar flow system at 100-H Area, will probably be about 96 days.

Currently in progress is an evaluation of the rate that fission products (particularly I-131), which might be contained in primary coolant diverted to the 1301-N crib, will reach riverbank locations in consideration of the flow paths for the complete system. The minimum travel time is applicable to only a very small fraction of the total outflow from the crib, and progressively longer flow paths (and travel times) will result in the fission products, which are discharged to the crib over a quite short period, entering the river over a much longer period. Therefore, radioactive decay will play an important role in reducing the amount of short-lived radionuclides which reach the river; and river dilution, due to the relatively long entry period, will be of major consideration in reducing the concentration of both short and long-lived radionuclides. Initial results indicate that about a five-fold reduction in the rate that radionuclides may enter the river is attributable solely to flow system geometry (in comparison to that calculated, based on minimum travel time).

Uranium Metal Surface Studies

EMF measurements as a function of time have been made on uranium metal (vs. a reference electrode) in $NaClO_4$ solution, for specimens of cold rolled ingot, beta ingot heat treated to 730 C, cold-rolled dingot, and beta dingot heat treated to 730 C, and revealed surface differences between ingot and dingot uranium. The tests show that heat treatment considerably alters the surface of dingot material, whereas ingot is unaltered. The significance of this observation in terms of surface bonding in the Al-Si process is being explored.

Electrodeposition of Nickel on Uranium

Nickel plating of uranium from a nickel sulfamate plating bath (just recently received from the Hanson Van Winkle Mining Company) was found to improve the plated nickel surface over that obtained in a Watts bath. Pore area was found to be one-fifth the pore area generated by a Watts bath, which is known to produce microscopic unplated areas.

SEPARATIONS PROCESSES

Purex Diluent Studies

A new sample of the hydrocarbon from South Hampton Refineries was examined to determine its potentialities as a possible Purex diluent. This material, the first straight chain hydrocarbon to be separated from petroleum on a large scale, can be produced at a cost low enough to compete seriously with the kerosene-type diluents normally used. The excellence of this product was shown by the following laboratory tests:

1. Gas chromatographic analysis showed it to consist of 99 percent straight-chain hydrocarbons from C_{10} to C_{14} inclusive, the other one percent being mainly C_8 and C_9 hydrocarbons, with small amounts of branched chain material.

2. The hydrocarbon showed a high degree of resistance to attack by sulfuric acid.

3. Passage of the hydrocarbon through a silica gel column caused no coloration of the column, indicating the absence of significant reactive, fission product-retaining compounds.

4. It performed about 100 times better than Soltrol 170 in thermal degradation and fission product pick-up and retention tests.

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Trilaurylamine Extraction of Neptunium and Plutonium from Purex 1WW

Hydroxylamine sulfate solutions strip neptunium and plutonium from 0.3 M TLA-Soltrol solutions very efficiently. However, there is reluctance to use this reagent in the Purex plant because of the, as yet unknown, potential for generating ammonia. Large amounts of ammonia cannot be tolerated in the Purex vessel vent system.

Laboratory tests show Na_2SO_4 - $Fe(HN_2SO_3)_2$ solutions are a satisfactory alternate to hydroxylamine sulfate. Contact of the solvent for one hour at 50 C with an equal volume of 0.05 M Na_2SO_4 - 0.05 M $Fe(HN_2SO_3)_2$ solution removed over 99 percent of the plutonium and over 97 percent of the neptunium present. The use of the Na_2SO_4 - $Fe(HN_2SO_3)_2$ solution has the disadvantage of adding more sodium to the 1WW stream. It does not result in more iron in the 1WW since ferrous sulfamate addition to a hydroxylamine sulfate strip would be needed to prepare a satisfactory feed for the 3A column.

Alternate Plutonium Reductant for Purex

Sodium formaldehyde sulfoxalate (SFS) is being studied as a plutonium reducing agent alternate to ferrous sulfamate (FS) in the Purex plant to eliminate addition of ferrous ion to the process. A plant test of this reagent in the 2D column is being considered. Laboratory tests showed that SFS is considerably less stable in nitric acid than is FS. Half-lives for SFS in 4 M HNO_3 were no more than one-seventh those for FS. The presence of holding reductants such as hydrazine or sulfamic acid did not increase the half-life of SFS in nitric acid. However, the sporadic rapid destruction of SFS, previously reported, has never been observed when hydrazine or sulfamic acid was present. The half-life of SFS in nitric acid at 45 C is quite low when the aqueous is in contact with Purex solvent (less than 1 hour at 1-3 M HNO_3). When uranyl nitrate is present in the aqueous phase, SFS reacts with it rapidly to produce U(IV). Thus, the presence of U(IV) must be considered in practical applications of SFS.

Four mini-mixer-settler runs simulating the Purex 2D column were made with plant 2DF as feed. The plant feed was sub-normal in uranium. As a consequence, uranium concentration in the solvent was low and uranium losses, zirconium-niobium decontamination and plutonium decontamination were all lower than is normal in the plant. However, plutonium decontamination obtained with SFS as reductant compared favorably with that obtained with ferrous sulfamate as reductant.

[redacted]

[redacted]

Thorium Processing

Study of a modified Sol-Gel process for the production of high density thoria continued with an electron microscope study of the steam denitrated thoria powder obtained from Mallinckrodt Chemical Works. The particle size ranged from 200 Angstroms to 200 microns and appeared largely as agglomerates of 100 Angstroms particles. This finding in part explains the 50 percent loss of thoria to the off-gas experienced at MCW.

In other work, sol drying-firing studies indicated that for sol depths in drying trays greater than 1 inch, thoria gel particles on the order of 0.5 to 0.75 inch size can be obtained. The preliminary firing studies show that the gel must be fired slowly (< 300 C/hr) up to 500 C to achieve large particle sizes. Above 500 C, the firing rate can be accelerated but must reach approximately 1150 C if subsequent shattering (due to a hydration mechanism) is to be avoided. For example, inert atmosphere fired (1040 C) and cooled thoria, shattered on exposure to ambient humidity air.

The applicability of sol techniques to the preparation of cermets is being investigated as a result of Ceramics Research interest. Thoria sols blended with 5 micron tungsten have been prepared and are awaiting pneumatic impaction.

Isolation of High-Isotopic-Purity Uranium-233

Isolation was successfully completed on the first sample of Hanford-produced high-isotopic-purity U-233. The product, slightly less than 0.5 gram of U-233 oxide (UO_3), contains only 1.3 ppm of U-232 and gives a radiation dosage much less than 1 mr/hr. Overall decontamination from radioactive impurities was > 10^8; gamma spectrometry showed fission products to be below detection limits and protactinium to be equivalent to less than 10 percent of the feeble U-233 X-ray count. The significance of this IPD-HL test is that it verifies Hanford's capability to manufacture U-233 very low (ca. 1 ppm) in U-232.

To perform the test, thorium metal slugs were irradiated by IPD in F-Reactor to ca. 500-600 g U-233/T Th. The slugs were dissolved in a one-slug dissolver installed in B-Cell of the High Level Radiochemistry Facility and the U-233 concentrated and partially purified (from Pa-233, fission products, thorium, and other impurities) by these sequential steps: MnO_2 scavenging of Pa; solvent extraction (with TBP or DBBP); and thorium oxalate by-product precipitation. The uranium product concentrate was then removed from the cell, and purification was completed by small-scale anion exchange from a chloride medium. The purified U-233 was precipitated (as UO_4) with hydrogen peroxide, ignited to UO_3, and sealed in an ampule for display purposes.

[redacted]

Remaining thorium dissolver solution (equivalent to a gram or more of U-233) and two or more (at least one high density and one low) irradiated thoria (ThO_2) target elements will be similarly processed. Principal objectives of total dissolution of the thoria slugs are: (1) to confirm nuclear calculations on isotopic content, and (2) to ascertain whether there is any problem in the dissolution of irradiated thoria. The U-233 will be recovered as a by-product of these operations. In addition, a small sample of ultra-pure U-233 (< 1 ppm U-232) will be obtained from decay of Pa-233 isolated by MnO_2 scavenging.

Solvent Extraction of Polonium

A survey of potential polonium extractants (all in xylene diluent) was made. Promise was shown by 1.14 M DBBP, 0.19 M Aliquot-336 (NO_3^-), 0.44 M TOA, and 5.9 M 1,4 diethoxybutane, which gave E_a^o values of 3.3, 11, 40, and 3.0, respectively, for extraction of polonium (probably tetravalent) from a 3 M HNO_3 aqueous phase. E_a^o values less than 1 were measured for 0.3 M D2EHPA, 0.1 M EHPA-0.1 M DBBP, 0.1 M D2EHPA-TBP, 0.1 M TTA and 0.1 M TOPO.

Ion Exchange Contactor Development

The Cold Semiworks folded loop ion exchange column was modified to conform closely with anticipated redesign of the Purex contactor. This modification was made to demonstrate the feasibility of "pulsed" resin pumping with the proposed Purex configuration. The possibility of using more of the contactor length, thereby increasing scrubbing efficiency, will also be explored. The effects of flow rate, acidity and temperature upon the ability to move resin will be evaluated. Preliminary studies indicate that the Purex configuration may be expected to perform like the Redox configuration, both units having comparable total lengths and resin pump placement.

Exchange Reactions of Transuranic Elements on Zeolites

An investigation into the possible use of zeolites in neutron irradiation applications was begun. Since the ability of zeolites to enter into exchange reactions with cations, including UO_2^{++}, PuO_2^{++} and NpO_2^{+} was not known the initial part of the study was concerned with sodium-uranyl ion equilibria. Five zeolites (Norton Zeolon, Linde 4AXW, 13X, AW-400, AW-500 and Hector clinoptilolite) were equilibrated with three solutions containing 0.24 g U/l in common, along with 0.01 M $NaNO_3$, 0.1 M $NaNO_3$ and 1.0 M $NaNO_3$, respectively. The large diameter of the UO_2^{++} ion (approximately 4.2 Å) indicated that the spatial requirements would be a factor in cation exchange reactions. The results showed that spatial considerations were in fact a primary consideration. Only the 13X (pore size opening 9 Å) removed a substantial amount of UO_2^{++} (230 µg U/g zeolite) from the solution containing 1.0 M $NaNO_3$ plus 0.24 g U/l.

[redacted] [redacted]

WASTE MANAGEMENT AND FISSION PRODUCT RECOVERY

Kilogram-Scale Technetium-99 Purification

Preparations continued for purification of the kilogram quantity of CPD-recovered Tc-99. The run plan is nearly complete, and equipment is being set up in the contamination-control zone of the 325-A Building. Start of the purification run is targeted for mid-March. Because a shielded cell is not currently available, a shielded cask will be used as feed tank, another cask as receiver, and the lightly shielded ion exchange column will be located in the stainless steel, cask-loading station. Zr-Nb will be removed by passing the feed through a silica gel bed contained in a concrete shielded drum, which will also serve as its burial container.

Supporting laboratory work included completion of the definition of optimum conditions for the preparation of technetium dioxide (by quantitative precipitation of $TcO_2 \cdot 2H_2O$ with hydrazine at pH > 8 followed by dehydration at 100 C) and further scouting experiments on the elution of technetium from anion exchange resins. Although either nitric acid or thiocyanate elution is satisfactory for the forthcoming final purification run, a more efficient reagent would be desirable for use in the primary recovery process to reduce the large volumes currently required. Various reducing agents and complexants are being tried. Thus far, neither ammonium salicylate, hydroxylamine nor hydrazine have suitable kinetics.

Direct Extraction of Cerium from Purex Plant "Carbonate Product"

A series of experiments was run to determine whether direct persulfate oxidation-D2EHPA extraction (for cerium-trivalent rare earth separation) could be applied to "carbonate product," a lead-containing rare earth concentrate produced in Purex Head-End. Cerium extraction (from a 2 M HNO_3, 0.2 M $S_2O_8^=$, 0.02 M Ag solution of carbonate product) was found to be quantitative, and promethium losses to the lead sulfate precipitate were only 2 percent. Thus, use of the simplified process appears feasible.

Cesium Removal from High-Level Waste

The use of lithium nitrate in place of ammonium carbonate as a scrub solution for removing sodium from Linde AW-500 ion exchanger was investigated. An alternate scrub solution would be desirable since disposal of the spent ammonium carbonate scrub solution presents a potential problem if the ammonia concentration is high enough to prevent blending with other waste streams. Lithium ion was selected because of its low competitive effect with cesium ion in the packaging process. Unfortunately,

the solubility of the lithium carbonate is quite low so that a significant amount of this salt will crystallize out during evaporation of the ammonium carbonate. An estimated minimum of 6 percent by weight lithium carbonate will be present in the evaporated eluate for a ten-fold reduction in eluate volume. Studies on the use of lithium in a scrub solution were cancelled. Other methods of reducing the amount of ammonia in the spent scrub solution will be studied.

Strontium-Lead Sulfate Carrier Precipitation

Further pilot plant investigations of the lead sulfate carrier precipitation process for removing strontium from acidified stored alkaline waste were made. These studies demonstrated that high temperature digestion, followed by cooling, of a sulfate-containing feed had no effect on subsequent strontium recovery by the normal lead sulfate strike. It was also demonstrated that several successive lead sulfate strikes with intermediate decanting could be made in the same tank with no increase in strontium loss. The cumulative loss after six successive strikes was only 2.7 percent. The procedure simulated a flowsheet modification in which precipitate from several batches of feed would be allowed to settle and accumulate in the precipitation vessel with considerably shortened time cycle.

Attempts to metathesize (caustic-carbonate) the lead sulfate cakes were only moderately successful (only 68 percent recovery was obtained in one carefully analyzed run). The poor recoveries were attributed to inadequate agitation and contact with wash and metathesis solutions, but the possibility exists that the digestion techniques described above produced refractory precipitate.

Strontium Semiworks Solvent Wash Studies

Difficulty with flooding in the HA column, excessive strontium losses in the HC column, and poor decontamination from calcium were noted in a recent Strontium Semiworks (SSW) Sr-90 purification run. Near the end of the run the solvent was replaced with new solvent; all abnormal behavior disappeared.

Laboratory studies showed that a combination of caustic (2.5 M NaOH) and acid (5 M HNO_3) washes decreased the disengaging time of the SSW solvent but had little effect on the strontium extraction and strip behavior. Addition of citrate or tartrate to the caustic wash prevented precipitation of iron. The presence of citrate aided in removal of yttrium, cerium and zirconium from the solvent.

Alkaline and acid washing of the SSW solvent was started at month's end, and samples of the washed solvent should be available soon for further laboratory study.

[redacted] [redacted]

In-Tank Solidification Studies

Technology for a second generation in-tank solidification unit is being developed in which evaporation is effected by immersed electric heaters. Prototype heaters from four vendors have been obtained for testing and evaluation. Installation of a pilot scale demonstration unit was completed and operations using non-radioactive coating wastes are underway. Included in this phase of the program is an investigation of methods and equipment for pumping hot concentrated wastes between tanks. Successful development of this procedure will result in substantial savings and improved flexibility of the overall Waste Management Program.

In a related study, arrangements have been completed for gamma irradiation of a heater operating in refluxed concentrated coating waste. By comparison to a similar system operating in the laboratory, the effects of radiation on scaling may be determined. Installation of the test equipment in the 3730 gamma facility is essentially complete.

Waste Packaging

In the zeolite sorption process for packaging Sr-90 and Cs-137, helium is injected into the container to improve the thermal conductivity in the final package. Helium is introduced at the end of the drying step through lines used to transfer the drying gas. The lines are then disconnected and the container is capped. An experiment was conducted at room temperatures with a full scale model of the container to determine the extent of helium loss while the container is disconnected. The experiment showed that helium concentration measured at the bottom of the container 60 minutes after the lines were disconnected, was 78 percent of the starting concentration. Concentrations at 5, 15 and 30 minutes were 93, 92 and 84 percent of original. The losses as measured in the model are probably high due to leaks and because the measurements were made at the bottom of the container.

EQUIPMENT AND MATERIALS

Electrical Discharge in Radiation Shielding Windows

Two samples of Penberthy Co. 3.8 density shielding glass were exposed to radiation from calcined Purex 1WW. The samples are 5 x 5 x 6 inches blocks mounted in steel frames with plywood shims touching and supporting the glass. One sample received 0.76×10^6 R at a rate of 4560 R/hr. The other received 9×10^6 R in a total of 44 days; part of the exposure was at 6600 R/hr and the remainder at 8900 R/hr. A third sample of glass was exposed to gamma radiation from a gold target under electron bombardment from a Van de Graaff generator. The sample received a total of 10^6 R during 97 hours; exposure was intermittent (day shift only) at about 5×10^4 R/hr. No electrical discharge has occurred in any of the samples.

Non-Metallic Materials

Samples of Viton A sponge (duPont Co.) were immersed for 28 days in carbon tetrachloride and Pydraul Fg (Monsanto Co.). No change was observed in either sample.

Corrosion of 304-L and 316 Stainless Steels in 234-5 Incinerator Scrubber Solution

Test coupons of 304-L and 316 stainless steel which have been in the 234-5 incinerator scrubber for several months were removed and inspected. Apparent general corrosion rates were very low, probably less than 0.1 mil/mo. Some pitting attack was visible on the 304-L sample; the deepest pit was 27 mils, corresponding to a pitting rate of 1-2 mils/mo. No pits were visible on the 316 sample.

Corrosion of 304-L Stainless Steel in HNO_3-HF Systems

A study of the inhibiting effects of thorium and aluminum on the corrosion of 304-L stainless steel by HNO_3-HF systems has been completed and is being documented as HW-80658. The inhibiting effects of boron and silicon are currently under study.

Linear Polarization as a Test for Susceptibility to Intergranular Attack

Linear polarization of 304-L and 304 stainless steels in 6 M H_2SO_4 - 0.01 M Fe(III) was investigated as a potential test for susceptibility to intergranular attack. The results are not reproducible and showed marked variation as a function of specimen history. Further tests with electropolished samples will be made in an effort to improve reproducibility.

PROCESS CONTROL AND DEVELOPMENT

Plutonium Reclamation Facility Support

A new experimental air-pulsed column has been placed in operation in 321 Building and is currently being utilized as a test unit in support of certain features of the Plutonium Reclamation Facility. Life testing of modified solenoid valves used in the pulsing control system is underway to determine reliability and service life. A symmetric pulse controller is being evaluated as a back up to the complex Cypak pulse timer initially provided for the project. Several months of satisfactory operation have been logged for the new controller. A pulse velocity control system comprised of a differential pressure cell, an AC to DC converter with a frequency response to 0.2 cps, and associated recording and controller instrumentation is under development and testing.

Operation of the column and associated control equipment has been very satisfactory to date. In other developments for the Plutonium Reclamation Facility, further measurements on a specific gravity meter for plutonium streams using a strain gauge load cell indicated a long term (one month) drift of about 4 grams, corresponding to ± 0.01 gram per cc.

In-Line Plutonium Monitor

Based on results reported last month, agreement has been reached on installation of a dual cell plutonium monitor to monitor plutonium losses for Purex N-Cell ion exchange contactor. The monitor consists of a Mark II (3-in. diameter) scintillating glass detector, with a 17 kev X-ray detector in series. Detailed design of the installation is now in progress. Concern about fluoride attack on the scintillating glass prompted a test in 0.01 M sodium fluoride, complexed with aluminum nitrate. During two weeks' exposure no apparent attack occurred.

Pressure Sensor Development

A device using a pressure sensitive resistive paint was tested to determine its applicability to pressure transducers, load cells and similar sensors. The paint is reportedly a mixture of zirconium chloride and rare earth salts suspended in an air-drying binder. In initial, exploratory tests a load cell device using this material was not stable to better than 1 percent. The resistance was found to be somewhat voltage sensitive. The current through the devices appears to follow the equation $I = kE^n$. Use of the devices as potentiometers and variable function generators appears feasible when high accuracy is not required.

Advanced Process Control Development

The A-Column flow control systems were completed with the installation of turbine flow meters. Incorporation of these systems into the automatic data logger is now in progress. Operation of the facility has been temporarily delayed by lack of accurate analyses of the TBP content of recently delivered Soltrol. Use of the gamma absorptiometer for this analysis has been explored with encouraging results. Rapid results of equal or greater accuracy than those obtained by standard analytical laboratory measurements appear to be obtainable.

Numerous revisions to the GE-412 master program package are necessary to adapt it to the particular machine on order and to the specific process being monitored and controlled. Most of the changes related to the subprogram which controls the scanner. Since the entire program depends on communicating with the process through the scanner, it is important to specify the scanner output at an early stage. This work was undertaken this month and about 40 percent of the needed changes have been completed.

Photometer Development

A single beam, filter photometer has been devised that separates the yellow color of an organic uranyl nitrate solution into that amount that is contributed by the solvent and that amount that is due to the solute, permitting an indication of the uranyl nitrate concentration of a sample to be made regardless of the current degraded condition of its solvent. This system is unique in that the information necessary to effect this separation of the two color contributors is obtained from two simultaneous measurements of the absorption characteristics of the sample, no reference solvent sample being required. Measurement and analysis of absorption spectra of uranyl nitrate solutions have shown that at any particular wave length the absorbance of the organic colored uranyl nitrate solution is the sum of the absorbance of the colored solvent and the absorbance of the uranyl nitrate. Further studies demonstrated that absorbance measurements at wave lengths of 392 mμ and 425 mμ provide the greatest resolution between uranium and colored organic constituents.

ANALYTICAL AND INSTRUMENTAL CHEMISTRY

Burn-Up Analysis

Bias between burn-up analyses provided by coulometric and alpha-activity determinations has been further examined and seems to have been partly due to the extraction procedure followed in preparing samples for alpha-counting. It is still believed that the remaining 2 percent bias between methods is due to an erroneous half-life for the decay of Pu-240. Adequate methods are now available for burn-up analysis of plutonium-aluminum alloys.

Amine Extraction of Uranium from Irradiated Thorium

A solvent extraction method using tri-iso-octylamine (TiOA) has been established for the analytical separation of uranium from irradiated thorium for measuring U-233. After the amine separation, U-233 is determined by alpha counting methods. Uranium is extracted into 10 volume percent (TiOA) in xylene from 7-8 N HCl. The uranium bearing organic solution is scrubbed with 7 N HCl and 7 N HCl - 1 N HF to obtain further separation from thorium, protactinium, zirconium and niobium. The uranium is removed quantitatively from the organic solution by stripping with two volumes of 0.05 M HNO_3. The aqueous phase is evaluated for U-233 by alpha counting. Consistent uranium recoveries of about 97 percent are obtained for the complete separation procedure.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Uranium-233 - Thorium Fuel Cycle Studies - Dissolution of ThO_2 by HCl in molten LiCl-KCl was shown to be slow, even with calcined samples having large surface areas (3 M^2/g). Although quantitative measurements have not been made, addition of CsCl to the melt (LiCl-KCl-0.375 CsCl) seemed to increase both the rate and extent of dissolution. The thorium content of the melt reached 10 weight percent after 24 hours in contact with dry, bubbling HCl.

An attempt to separate UO_2 from Th(IV) in this melt through reduction of UO_2Cl_2 by 8 percent hydrogen in argon failed because ThO_2 also precipitated. Either a metathesis reaction took place, viz.,

$$UO_2 + Th(IV) \rightarrow ThO_2 + U(IV),$$

or sufficient water vapor entered the melt to cause precipitation of ThO_2. Separation factors as high as 50 have been measured for the separation of thorium from uranium by electrodepositing uranium dioxide. These experiments involved the LiCl-KCl-0.375 CsCl solvent and mixtures of O_2-Cl_2; however, HCl-O_2 mixtures produce about the same results.

EMF Measurements of M/M(II) Couples in the Molten Alkali Chlorides - These measurements have been finished for the KCl-NaCl system. The behavior of the Co(II) in this is similar to that of Ni(II). Although ΔG is more negative by about 9 kcal for 0.1 m Co(II) than for an equal concentration of Ni(II) in the same solvent, the slopes of the EMF-melt composition curves are very nearly equal.

After appropriate corrections for concentrations and thermal EMF were made, the data obtained for Ni(II) in the equimolar KCl-NaCl agreed within 2 mv with data obtained by D.C. Hamby at Oregon State. Analytical data are not yet available to make such a comparison for the Co(II) system.

Tungsten-Uranium Dioxide Cermet - An electrolysis at 500 C and 1.75 V (vs. Ag/AgCl) of a LiCl-KCl melt containing 5 weight percent tungsten added as WCl_6 and 5 weight percent uranium as UO_2Cl_2 produced a finely divided black deposit on a tungsten cathode which analyzed 37 mole percent tungsten metal in UO_2. Further work on the preparation of tungsten UO_2 cermets, in a fluoride melt system, is planned.

Engineering Development - In continuing evaluation of the UO_2-PuO_2 from the three electrodepositions reported last month, 0.5-inch water-washed chunks were crushed to minus 4 mesh and washed with water for 8 hours. The chloride content was reduced to the range 28 to 50 ppm, appreciably lower than the previous range of 40 to 140 ppm.

Operation of the Salt Cycle Process in the High Level Radiochemistry Facility was continued with fuel rods from two vibratory compacted mixed oxide fuel assemblies containing 0.45 percent Pu in natural uranium. One fuel assembly had been irradiated to 100 MWD/T in the Plutonium Recycle Test Reactor and cooled for 292 days, and the other had been irradiated to 360 MWD/T and cooled for 373 days.

The oxide was successfully removed from eight half-rods by the vibratory decladder and the UO_2 partially oxidized to U_3O_8 with a nominal 1.5 liter per minute air purge at 450 C for 24 hours. Apparently wet oxide (water in-leakage during underwater storage) reduced the air rate appreciably by plugging some of the oxidizer air inlet holes. Alternate passage of oxidizing air and reducing argon-hydrogen through the empty oxidizer at 700 C unplugged some of the air inlet holes. No volatile gamma activity was detected in the caustic scrubber solution as a result of the oxidation step.

Approximately 25 pounds of U_3O_8 + UO_2 and PuO_2 were introduced into the molten salt bath, and dissolution was begun with chlorine gas. The dissolution time was increased appreciably by incomplete oxidation of the feed, and the cerium and plutonium oxides began to dissolve only toward the end of the uranium oxide dissolution. Ruthenium was the major fission product found in the caustic solution, and its activity in the salt bath gradually decreased as the dissolution proceeded. Traces of zirconium were also found in the caustic, but the major portion remained in the salt bath. From off-gas analyses, an overall decontamination factor greater than 40,000 (limit of detectability) for ruthenium and zirconium was obtained in the off-gas treatment system (caustic scrubber, absolute filter, charcoal trap). Decontamination factors of 14 and 7 were obtained across the caustic scrubber, the remainder across the absolute filter.

At month's end, a UO_2 electrolysis (partition) was completed with the data as yet unanalyzed.

Dissolution of PRTR UO_2-PuO_2 Fuels

Laboratory studies were made on a one-step dissolution of PRTR UO_2-PuO_2 fuel cores as an alternate to the two-step (separate dissolution of UO_2 and PuO_2) procedure previously studied. In the one-step approach both PuO_2 and UO_2 are dissolved in HNO_3-NH_4F solution at 80 C. The dissolvent

is, initially, 11.1 M HNO_3 - 0.1 M NH_4F. Final dissolver solution is 10 M HNO_3 - 0.42 M $UO_2(NO_3)_2$ - 0.5 g/l Pu - 0.1 M total fluoride. Complete dissolution of UO_2 occurs before the PuO_2 is all dissolved. Laboratory tests showed that the rate of dissolution of PuO_2 in 10 M HNO_3 - 0.1 M NH_4F at 80 C is not affected by the presence of up to 0.8 M $UO_2(NO_3)_2$.

Glass Studies

Laboratory study continued on the effect of chemical composition and fission product content on the properties of glass systems proposed for waste storage. "Cold" fission products are being incorporated corresponding to (1) low burn-up uranium (current Purex), (2) 10,000 MWD/T plutonium recycle, and (3) 22,000 MWD/T uranium converter fuels. With these compositions, drip temperature and solubility are being measured for phosphate glasses as a function of sulfate to phosphate ratio. For a Purex composition, both the drip temperature and solubility were found to decrease as the sulfate/)sulfate + phosphate) ratio increased from 0.20 to 0.30. This phenomenon is in marked contrast to that observed earlier with higher fission product contents.

In other laboratory experiments, use of diammonium phosphate in place of phosphoric acid was found to raise the drip temperature and change the color and appearance of the solidified melts.

Preparations for the BNL-HL continuous glass making experiment continued. The Brookhaven equipment was received and will be initially set up in the hot cell mock-up area for cold testing and operator training. Extensive modification will be necessary prior to installation in the cell.

RADIOACTIVE RESIDUE PROCESSING DEVELOPMENT

Cold Semiworks Spray Calciner

The experimental development program in the 18-inch diameter spray calciner continued to explore the effects of varying feed stocks and internal geometries. Simulated Redox feed with added silica, borax and sugar to produce a low melting point powder was used in three runs. Plugging of the recirculation annulus occurred with both the 16-inch and 14-inch by 8.5-foot draft tubes. Heavy deposits also occurred on the bottom cone of the calciner. Plugging of the 6-inch diameter reactor outlet with large lumps was evident in two runs. Borax appears to be responsible for these difficulties. The product powders were found to agglomerate at about 300 C in contrast to the calciner wall temperature of 600 C. Adjustments of Redox feed compositions are being explored to obtain a desirable spread between the agglomeration temperature of a meltable powder and the operating temperature of the calciner wall.

Water and feed capacity tests at 700 C without a draft tube indicated capacities about 60 percent of those observed with a 14-inch draft tube. Clean-wall water capacities without the draft tube ranged from 18 to 24 gallons per hour with steam pressures to the atomizing nozzle from 30 to 60 psig. Corresponding capacities with simulated Purex 1WW feed were 12 to 14 gallons per hour. Residual equilibrium wall deposits from the 1WW runs reduced the water capacities to about 16 to 18 gallons per hour.

Intermediate-Level Waste Studies

The electrodialysis unit was successfully operated for extended periods with Purex acid condensate. Acid removal of 80-90 percent was maintained at a flow rate of 1 liter per minute and a volume reduction of 10. Radioruthenium removal on continuous operation at these conditions was 40-45 percent. Relative acid and radioruthenium decontamination factors obtained during a continuous recycle experiment are presented below:

	Relative DF
HNO_3	Ru
10	2.2
20	2.9
50	3.7
100	5.0
1000	11.5

COLUMBIA RIVER SEDIMENT STUDIES

Columbia River Travel Times

Columbia River travel time calculations were completed for the tracer test using I-131 discharged to the river following a fuel rupture. These travel times are tabulated below. The tracer was introduced at a time when the Columbia River flow was 45,000 cfs, but the flow was sharply regulated so that the mean daily discharge of the river was 64,700 cfs at Priest Rapids Dam. When tributary flows and reservoir heights are incorporated, these data will be useful to the U.S. Geological Survey, who participated in the test, and to the Hanford Radiation Protection Operation.

Sampling Point	Distance from Origin to Sampling Point (miles)	Travel Time (days)
Finley	43	1.5
McNary Dam	77	5.9
Biggs Junction	156	7.8
The Dalles	175	9.4
[redacted] River	196	10.6
Vancouver	259	14.6

[redacted]

During this travel time study, measurements were made of the Na-24 concentration at Finley, Washington, for a 24-hour period. During the corresponding 24-hour period, the Columbia River flow rate past the reactors varied a factor of three because of operation of the Priest Rapids Dam. In spite of this large fluctuation, the Na-24 concentration varied only about 40 percent, indicating that these flow rate changes do not as seriously affect the radionuclide concentrations at downstream locations as might be expected from the flow rate changes.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

A mathematical model of the Hanford ground-water flow system is to be built rather than the electrical analog previously planned. This change in approach resulted from evaluations of the relative magnitudes of costs, the broader applicability of the mathematical model (evaluated by digital methods), the success achieved in development of the Hanford "steady-state" flow program, and most significantly the recent advances made in devising techniques for solving large systems of simultaneous algebraic equations (diakoptics). The formulations and programs associated with the development and interpretation of the mathematical model will not be limited to the Hanford flow regime but will have application in conducting similar flow studies at other sites.

The Hanford-developed "Steady-State Flow" computer program is being modified also to permit calculating the flow of heat through soils. An analysis was applied to show that the program will be stable for such an application. The original program was copied and changes were made to incorporate the heat generation term. The first calculation attempt is being applied to evaluate the steady-state soil temperatures resulting from the generation of heat by a radioactive salt cake contained in a buried waste tank.

A peculiar shape of the capillary pressure-permeability curve measured in the laboratory for a coarse sand indicated either pore size discontinuity or wall effect. The ratio of column diameter to mean sand particle diameter eliminates the possibility of wall effect. These sand columns were packed in the soil column packer which was tested and found to give uniform density of pack along the column. The cross-section of the column showed that an annulus of larger particles existed adjacent to the wall of the cylinder containing the sand column. This annulus possibly explains the peculiarity observed in the capillary pressure-permeability curve.

The capillary pressure-permeability relationship of such a highly specialized type of nonhomogeneous material is of very little value as a characterization of the material, since what is measured is a combined effect of the parts rather than a measure of the sample as a whole. This material is produced through the method of packing and has not been noticed before. Methods of measuring the radial uniformity of material packed in the packer are being studied. It is desired to test for radial uniformity for various materials and to find ways of correcting the packer to produce uniformity in both the radial and longitudinal directions.

Fabrication of the neutron moisture probe was completed by the Nuclear Instrument Development Unit. The probe was specifically designed to assist in evaluating tank farm thermal conductivity problems by measuring soil moisture adjacent to boiling tanks; however, initial instrument tests indicate that the probe will be a valuable geophysical tool for geohydrologic research. The neutron well-logs correlate well with both gamma and litologic logs, and the probe exhibits a high sensitivity for detecting small changes (1-2 percent) in soil-moisture content.

RADIOLOGICAL AND HEALTH CHEMISTRY

Bioassay Studies

A comparison of the Cs-137, Cs-134, Na-22 and Zn-65 levels in urine samples of local origin with a sample from Alaskan Eskimos showed that the Eskimo sample contained 50 times as much Cs-137, Cs-134 and Na-22, but only one-half as much Zn-65. This reflects the fallout origin of Cs-137, Cs-134 and Na-22 and the higher levels in the diet of the Eskimos. Zn-65 occurs in fallout, but also occurs as a result of reactor operations in the water local residents drink. These data are being correlated with whole body counter results.

Uranium Inhalation Study

Arrangements were completed to perform uranium-daughter inhalation measurements on uranium miners at the Union Carbide mines in southwestern Colorado in April. In this joint study with the Radiological Physics Operation a whole body counter will be transported to Colorado. By collimation counting of the gamma rays from the radon daughter product, Bi-214, determination of the amounts of dusts containing these daughter products can be made in the various portions of the respiratory system. Dust sampling, sizing and analysis will also be carried out to relate the uptake to the environmental conditions. These studies will provide information to plan and evaluate laboratory studies designed to simulate mine conditions.

Carbonaceous Sample Ashing Techniques

Improvements were effected in the cell design of the Tracerlab low temperature asher, which increased the oxidation rate. This asher utilizes atomic oxygen as the oxidizing agent, accomplishing the ashing without high temperature or mechanical disturbance of the sample. In a test of the apparatus 1 gram rods of graphite were almost completely oxidized in 6 hours. A delicate, light gray ash remained which maintained the form of the original graphite rod. This ashing technique may find application in analysis of trace inorganic constituents of organic materials.

Radiation Chemistry

A cell counting and sizing system utilizing the Coulter Counter (model B) and an RCL 256-channel, pulse-height analyzer has finally been adjusted to measure cell volumes in suspensions with a precision of approximately ± 10 percent. A preset-count circuit has also been installed in order to compare the distributions of volumes in populations comprised of equal numbers of cells. Since the volume span of the human erythrocyte is from 86 μ^3 (bidiscoid shape) to 248 μ^3 (spherical shape), a precision of ± 10 percent seems adequate for a number of experiments. Among those planned are: (1) measurement of cell-volume changes as a function of time after irradiation, (2) comparison of the volumes of intact cells and ghost cells under the same osmotic pressure to determine whether they can be differentiated by pulse-height analysis, and (3) comparison of the effects of chemicals on intact-cell volumes under the same osmotic pressure to determine whether chemical-induced structure changes are related to radiation protection.

Thus, in addition to our primary goal of uncovering chemical protective agents for the living cell, we should be able to obtain considerable information on the nature of the radiation-induced, cell-lysing process itself, e.g., answers to these questions: (1) Are the contents of the damaged cell released continuously after irradiation at a variable rate, or are they released suddenly after the cell has reached a critical size? (2) Do chemical protective agents function by diverting free radicals from the cell surface, or do they modify the cell surface so as to make it more resistant to radical attack?

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Aerosol Sampling Study

The measurements of errors associated with sampling relatively large particles at collector filter face velocities much less than the stream velocity were completed for a stream velocity of 9 miles per hour. Sampling rates much lower than isokinetic gave errors consistent with the assumption that the ratio of number of particles collected on the sub-isokinetic sample to the number on the properly collected sample should increase with increasing particle size and with higher stream velocity. Inconsistencies from this expected result were observed for large particles and for 1/2 and 1/4 isokinetic sampling rates. Explanation is sought, but the source of the inconsistency was not identified.

Iodine Studies

The absorption of methyl iodide by various liquid scrubbers was further studied. Various alcohols and aqueous silver nitrate proved to be very inefficient compared with 0.1 N $AgNO_3$ in ethyl alcohol, whose efficiency ranged from 75-95 percent.

Molecular radioactive iodine adsorbed on charcoal was shown to be eluted with an air stream following adsorption of non-radioactive methyl iodide on the charcoal. Iodine previously adsorbed on the charcoal exchanges with iodine in the methyl iodide. Subsequently, the methyl iodide is rather readily eluted with air passing through the charcoal. The exchange is controlled by the relative fraction of inert methyl iodide and the carrier of the radioactive molecular iodine. When the ratio was 1000:1 in favor of methyl iodide, about 23 percent of the iodine was removed with air elution in 4 hours. Thirty-four percent was removed in 4 hours when the ratio of methyl iodide to molecular iodine was increased to 2000.

The half-time for elution of I-131-tagged methyl iodide with air from charcoal was found to be from 4 to 6 hours for three different kinds of charcoal. About 60 µg of methyl iodide per gram of charcoal were initially adsorbed on the charcoal.

These observations further emphasize the potential complex reactions involved when organic vapors are present in a stream from which radioactive iodine must be removed. The relatively slow rate at which equilibria are reached also suggests that many experiments using radioiodine and charcoal, particularly where organic compounds are involved, have likely not taken into account these changes with time.

Treatment of charcoal with bromine vapor was shown to aid the retention of methyl iodide. Apparently the replacement of the radioiodine in the CH_3I with bromine occurs, thus releasing molecular iodine which is retained much more efficiently. In principle, this may be a practical way to insure retention of organic forms of iodine on charcoal.

ISOTOPES DEVELOPMENT - 08 PROGRAM

Acid-Side Promethium Purification Processes

Column runs aimed at optimizing conditions (eluant concentration, pH, flow rate, column length, temperature, etc.) for acid-side, high-capacity purification of promethium continued, using both HEDTA and DTPA as eluants. Noteworthy was the fact that precipitates were obtained in two runs with 0.025 M DTPA, indicating that the maximum usable concentration of this eluant is less than 0.025 M.

Source Forms and Encapsulation

A series of pneumatic impactions was performed during the month. Strontium fluoride (SrF_2), prepared by reaction of aqueous HF with $SrCO_3$ followed by filtration and heating to 1100 C to remove excess HF and moisture, was loaded into a can, welded shut, and compacted at 30,000 psi. An excellent

low-porosity compact of 100 percent theoretical density (4.24 g/cc) was formed. Distrontium titanate powder (Sr_2TiO_4), prepared last month, was compacted at 334,000 psi to yield a low porosity product of 4.70 g/cc density. Several compactions were made with strontium titanate powder ($SrTiO_3$) to:(1) test a new can design (aimed at producing flaw-free clads of uniform wall thickness); and (2) provide samples for non-destructive testing (NDT). The design, which features "top loading", was completely successful. Following NDT measurements, the compacts were sectioned, confirming both the thickness and uniformity of clad and freedom from imperfections predicted by ultrasonic mapping.

One attempt was made to prepare cerium sesquioxide (Ce_2O_3) by in-can reaction of a mixture of CeO_2 and powdered graphite. The attempt was unsuccessful due to plugging of the off-gas line, followed by pressurization and rupture, during heating (1250 C) and outgassing. However, on sectioning the can, part of the contents were found to have been completely reduced to Ce_2O_3. Further attempts will be made with a modified design and altered heating cycle.

In supporting laboratory work, several potentially interesting heat source materials and chemical intermediates were prepared. Strontium sulfide (SrS, ρ = 3.70 g/cc) was made by reaction of $SrCO_3$ with elemental sulfur and is expected to prove to be a high-melting, refractory material with re-entry burn-up potential. A range of high-cesium borate "compounds" was scouted (Cs_3BO_3, $Cs_4B_2O_5$, $Cs_3B_3O_6$, $Cs_2B_4O_7$, etc.). If in the pure form, these would contain up to 87 weight percent cesium. Most success resulted in preparing and identifying $Cs_2B_4O_7$ (61 percent Cs). Further work with these will be aimed at better defining the Cs_2O-B_2O_3 system.

Pneumatic Impaction Source Fabrication

A new single wall impaction can design (M-11) was developed and tested with highly satisfactory results. The design incorporates top loading and rounded internal corners for elimination of local stress risers. Four 2-1/2 x 2-1/2 inch compacts were made with the new design and with $SrTiO_3$, Nd_2O_3 and Sm_2O_3, the latter two as stand-ins for Pm_2O_3. The new design has the additional benefit of apparently yielding higher green density (60 percent versus 50 percent) in the cold prepressing operation. Control of wall material deformation in the new design is markedly improved as indicated by ultrasound non-destructive testing. Wall thickness variation was ± 0.005 inch in a normal 0.165 inch wall.

[redacted]
Testing Manager
Chemical Laboratory

[redacted]

DECLASSIFIED

BIOLOGY LABORATORY

A. ORGANIZATION AND PERSONNEL

No significant changes occurred.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Columnaris

Rainbow trout held at 64 F since January 1 of this year continue to release an increased number of columnaris organisms into the trough water. Mortality of fish has increased but the increase appears to be largely due to another fish disease known as Furunculosis. The concentration of columnaris in the trough water is considerably in excess of 100 organisms per ml of trough water. Of ten fish taken from this trough and placed in individual containers, some showed a continued increase in rate of production and release to water of columnaris organisms; some showed a steady decrease and some showed little or no release of organisms. This suggests that the fish are in a cycling state of infection, recovery, refractory period, re-infection, etc. Our general experience has shown that the infection period is approximately one month in length. We as yet have no knowledge of the duration of the refractory period.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Phosphorus

Gonads of cichlids fed P^{32} chronically for about two years failed to mature at feeding levels of 4.0 μc P^{32}/g food. At a lower level of 1.0 μc P^{32}/g food, growth of testes was clearly depressed to less than half that of controls. No effect was noted at a feeding level of 0.25 μc/g food.

Copper

Rats were injected with tracer levels of Fe^{59}, immediately subsequent to irradiation and/or copper injection. The distribution and excretion of the Fe^{59} was studied as an indication of damage from the radiation and copper treatments. Many of the resulting data can be interpreted only in the light of hematological and pathological studies not yet completed. Urinary Fe^{59} was elevated as a result of both copper and radiation treatments and seemed to correlate directly with the hemoglobinuria observed. Blood Fe^{59} levels were generally reduced by both copper and radiation with radiation having the more pronounced effect.

Zinc

Rainbow trout, after five weeks feeding at levels of 0.01, 0.1 and 1.0 μc Zn^{65}/g fish, Monday through Friday, showed no overt response or mortalities. For the highest feeding level the total intake for the five-week period was 2800 μc/fish.

Zinc-Cadmium

The percent of an injected dose of Zn^{65} or Cd^{115m} present 0.5 hr post-injection was reduced in the liver and pancreas and was increased in the plasma when the amount of stable zinc in the injected solution was increased (to a maximum of 0.19 mM/kg). The Zn^{65} content of the kidney was slightly lowered and that in the prostate was unaffected by the stable Zn, while the Cd^{115m} was increased in both organs. The Zn^{65} level in the small intestine and contents was reduced by elevated quantities of injected stable Zn while the Cd^{115m} level was unaltered. Twenty-four hours after injection each tissue contained the same amount of Zn^{65} regardless of the quantity of $ZnCl_2$ injected, suggesting that the rate of uptake is decreased and the rate of loss increased with increasing levels of stable zinc.

Strontium

Sections of soft tissues obtained from the offspring of pigs on the 625 μc Sr^{90}/day level that died or were killed after weaning were reviewed for evidence of histopathological change. The most striking change observed was the presence of myeloid cells infiltrating several organs in 6 of the 11 animals examined. The infiltrating cells of the myeloid series were most frequently observed in the kidney and heart muscle. In most cases, tissue hemorrhage and necrosis were associated with the infiltrations.

The changed observed differed from the typical sites of extra-medullary hematopoiesis observed normally in newborn swine, in several respects:

1) The lesions were rarely observed in spleen and liver, the usual sites of extra-medullary hematopoiesis.
2) The cells varied markedly in their degree of maturity from many very young cells undergoing mitosis to the more mature cells, which are usually the predominant type in extra-medullary hematopoiesis.
3) The nests of cells frequently were noted to be spreading along the connective tissue structure of the organs, suggesting that the cells were invasive.

In these respects, the lesions were suggestive of myeloma formation. Additional work is in progress to define this interesting and significant lesion.

Radioanalysis of soft tissues and the skeleton from a number of animals ingesting Sr^{90} daily has shown that essentially all of the Sr^{90} is present

in the animal's skeleton. At one year of age and older, the Sr^{90} content of the skeleton of animals that ingested Sr^{90} throughout life is approximately 10 x its daily intake of Sr^{90}. When animals are started on Sr^{90} at nine months of age and are killed or die at one year of age, the Sr^{90} content of the skeleton is about 3 x the daily Sr^{90} intake. This lower Sr^{90} content of the skeleton in the latter animals is a reflection of the slow turnover of bone and thus the failure of the skeleton to equilibrate with the diet in the short exposure period.

Results of analyzing various skeletal portions indicate that rates of growth and remodeling differ in the various parts of the skeleton.

Relative Concentrations of Sr^{90}
in Some Portions of the Skeleton of One-Year-Old
Miniature Swine
$\frac{(\mu c/g \text{ - portion of skeleton})}{(\mu c/g \text{ - total skeleton})}$

Skeletal part	Animal #1 - ingested Sr^{90} daily starting at 9 months of age	Animal #2 - Sr^{90} exposure throughout life
Total skeleton	1.0	1.0
Tibia-fibula, prox. $\frac{1}{4}$	1.3	1.1
middle $\frac{1}{2}$	0.65	1.05
distal $\frac{1}{4}$	1.1	1.1
Skull w/o teeth	0.54	0.94
Pelvis	1.3	1.0
Thoracic vert. - body	1.9	1.0
processes	1.1	1.05

Ruthenium

Eight Palouse pigs whose skin was exposed at 46 sites to Ru^{106} particles nine years ago at dose rates of 200-19,000 μc/hr per site were re-examined. Lesions were visible in all cases with the exception of one of the 200 μc/hr sites, and consisted of various amounts of scarring and hyperkeratosis which were somewhat dose dependent. No changes suggestive of neoplasms were observed.

Iodine

The iodine-131 vapor generator and chamber were tested several times in an attempt to increase the yield of I^{131} vapor and decrease production of water vapor which condenses on the chamber walls. Some modifications and improvements were made, but probable skin deposition still appears to be too low for practical detection of thyroid uptake and half-life determinations of I^{131}.

Neptunium

The Np^{237} toxicity study in sheep is continuing and two additional groups (one ram, one ewe and one wether per group) were tested. In the two previous groups given Np^{237} at 3 mg/kg, all the sheep died within six days. The animals in the last two groups were given 1.5 mg Np^{237}/kg and all survived except one ewe which showed signs of CNS disturbances and apparently died of aspiration pneumonia after inhaling some ingesta. The results of rose bengal-I^{131} liver function tests indicate impaired function very similar to that seen in the previous sheep given the higher dosage. No definite differences in toxicity between rams, ewes, and wethers are yet manifest.

Neptunium

Liver homogenates were prepared from female rats injected with toxic doses of Np^{237}, 18 hours prior to sacrifice. Studies were made of the ability of this homogenate to oxidize C^{14}-labeled palmitic acid in vitro. The Np^{237}-poisoned animals produced only about 1/3 as much C^{14}-labeled CO_2 as did control animals, strongly suggesting that Np^{237} inhibits the oxidation of fatty acids. A final conclusion, however, must await determination of specific activity of the C^{14} of total liver fat and evolved CO_2.

In preparation for studies of the effect of Np^{237} on the incorporation of of tritium-labeled leucine into liver and plasma proteins, methods were developed for the counting of tritium-labeled proteins by liquid scintillation methods.

Plutonium

Skin lesions produced in Hanford miniature swine injected intradermally two years ago with Pu^{239} nitrate at levels of 0.0016-5 μc per site were re-examined. None of the 0.0016 μc sites were visible or palpable. Damage at the higher level sites consists primarily of depressed cicatrices of varying depths and diameters and, in some cases, a keratinized scab, particularly at the 1- and 5-μc sites. Generally, the 1-μc sites manifested more scar formation than the 5 μc, probably due to more ex ensive tissue reaction, sloughing and subsequent loss of plutonium in the 5-μc sites. One of these animals died almost two years after the injection of 78 sites with a total of 67 μc Pu^{239}. At none of the levels were all sites visible. Only about 1% of the total administered dose was retained in the skin and subcutaneous fat. A large amount of the plutonium had undergone translocation to the liver (9%) and the skeleton (7%). Some regional lymph nodes showed very high activity with autoradiography; these nodes showed extensive

calcification and necrosis. These findings indicate that considerable translocation occurs even with superficial contamination, considering that a large amount of the administered plutonium was sloughed during the first few weeks.

Using intestinal perfusion and bile duct cannulation techniques, it was determined that DTPA and TTHA are absorbed almost exclusively from the jejunal segment of the small intestine of rats. This was indicated by the fact that the excretion of plutonium via the bile was markedly increased by perfusion of the jejunum with DTPA or TTHA solutions and was not increased when other segments were so perfused.

Experiments were performed to test the effect of desferrioxamine B (DFAB) in preventing deposition of plutonium or in effecting its removal once deposited in rats. DFAB is a highly specific chelating agent for iron and has not previously been studied for plutonium removal. Results of these experiments are not yet available.

Gastrointestinal Radiation Injury

Further studies were performed to determine the effect of radiation on the leakage into the intestine of intravenously injected polyvinylpyrollidone (PVP). Mice were observed to show only a small effect of irradiation on PVP leakage in contrast to the much larger effects observed in rats. Cysteine was found to exert a greater protective effect than AET in preventing PVP leakage from rats exposed to nitrogen mustard.

Secondary Disease Studies

Studies were made of the cell population of liver sections as affected by injections of rat bone marrow or rat spleen cells. The most significant finding was a leukocytic infiltration of periportal sinuses and central veins, one to three days after injection of 2×10^7 bone marrow or spleen cells in non-irradiated mice. This contrasts with earlier observations in X-irradiated, bone marrow or spleen treated mice, which indicated no significant leukocytic infiltration until three weeks post-treatment. These observations, if confirmed, should provide a means for the early indication of graft-host response in chimeras.

Inhalation Studies

Preliminary studies are in progress to determine the degree of independent biological behavior of the alpha emitters, U^{238}, U^{234}, Th^{230}, and Ra^{226}, in secular equilibrium in the ore dust. Following exposure to uranium ore dust, rat lungs were analyzed for U^{238} and Th^{230}. There was a definite trend away from Th^{230}-U^{238} equilibrium in the lungs, with Th^{230} being retained to a greater extent than U^{238}. The ratio of Th^{230} to U^{238} in the lungs increased from an average value of 1.5 at one week post-exposure, to a value of 2.7 at eight weeks post-exposure. These results are comparable to published studies where these elements were inhaled separately and in relatively pure form.

Microbiology

Experiments on the mechanisms of tryptophan synthesis in Neurospora are continuing, using specific genetic blocks in the synthetic pathway and using carbon-14 labeled "Tryptophan" to stimulate growth. Present results provide very strong evidence that the temperature at which the organisms are grown, in large part, determines the partition of tryptophan between protein synthesis and the tryptophan metabolic cycle. Several-fold increase of protein subsequent to the disappearance of tryptophan in the external medium indicates an extensive internal pool of tryptophan which potentially can be channeled into protein synthesis. The nature of this pool and the mechanism by which it is controlled is being further investigated.

The effects of D_2O on tryptophanase induction and synthesis were determined in E. coli. At 37 C the differential rate of synthesis in D_2O is about one-third the rate in H_2O controls. Moreover, the induction lag is twice as long in D_2O as in H_2O (10 minutes vs 20 minutes). This delay may be related to the previously reported effects of D_2O on RNA synthesis. When these same experiments were carried out at 25 C the induction lag was prolonged to 40-60 minutes and the differential rate of synthesis was less than one-fourth the rate at 37 C.

Columbia River Ecology

Analysis of more than 100 whitefish collected at Priest Rapids in January showed that approximately 80% of these fish had been exposed to reactor effluent, as indicated by their above-background content of Zn^{65}.

Chlorophyll A productivity in periphyton, in January, remained constant and comparable to that for December. The average dry weight productivity of periphyton decreased slightly, but the average organic matter increased, possibly reflecting a decreased silt load in the river. The average P^{32} activity for January, 148 nc/g dry organic matter, was only slightly less than the value for December.

Terrestrial Ecology

Stem flow, the flow of rain from the branches down the stems of trees, is of great importance in many environments. Soil profiles beneath greasewood showed that stem flow did not concentrate moisture beneath this shrub in this environment. Approximately one-fifth of the moisture which fell was intercepted by shrub canopies and litter and did not penetrate the soil. Measurements of pH in soil beneath greasewood showed highest values on the east and northeast sides of shrubs, under the areas of greatest leaf litter accumulation. This is in accord with the previous report that leaf litter is a significant factor in increasing the sodium content of soil in the vicinity of these shrubs.

RC Thompson:es

TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORY

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposia

None

b. Seminars (Off-Site and Local)

Matchett, W. H. Some aspects of tryptophane metabolism in Neurospora. Exchange Seminar Program. Washington State University, Pullman, Washington. February 19, 1964.

Eberhardt, L. L. Radiation in relation to plants and animals. Washington State Department of Game, Olympia, Washington. February 27, 1964.

c. Seminars (Biology)

Mahlum, D. D. Neptunium-237 and lipid metabolism. February 5, 1964.

Silverman, M. S., Head, Microbiology and Immunology, USNRDL, San Francisco. Effects of radiation on infection and the role of infection on radiation injury. February 7, 1964.

Olson, P. A. Further studies on columnaris disease. February 12, 1964.

Leader, R. W., Associate Professor of Veterinary Pathology, Washington State University, Pullman, Washington. Antibody - friend or foe. February 14, 1964.

Dean, J. M. Effects of temperature acclimation on some aspects of carbohydrate metabolism of crustacae. February 19, 1964.

McClellan, R. O. Measures of damage in miniature swine ingesting Sr^{90}. February 26, 1964.

Klontz, G. W., Western Fish Disease Laboratory, Seattle, Washington. Diseases of fish: problems and their solution. February 27, 1964.

d. Miscellaneous

None

II. Articles Published

a. HW Documents

None

b. Open Literature

III. Visits and Visitors

a. Visits to Hanford

J. J. Davis, AEC, Washington, D.C., attended the Columbia River Studies Meeting and discussed research with F. P. Hungate. February 3-7, 1964.

M. S. Silverman, USNRDL, San Francisco, presented a seminar. February 7.

Lars Friberg, Institute of Hygiene, Karolinska Institute, Stockholm, Sweden, discussed inhalation problems with B. O. Stuart and toured facilities on February 7.

Ronald E. Pine, Washington State Pollution Commission, Seattle, toured facilities and picked up cichlids. February 10, 1964.

Approximately 50 science students from the local and surrounding high schools toured Biology facilities. February 11, 1964.

Three members of the Sunnyside Chamber of Commerce, press and radio station toured facilities with Ray Palmer on February 12.

R. W. Leader, Veterinary Pathology, Washington State University, Pullman, Washington, presented a seminar. February 14.

C. Osterburg, B. Pearcey, and A. Carey, Department of Oceanography, Oregon State University, Corvallis, discussed research with R. Nakatani on February 6.

D. Hathaway, J. Howell, L. Lydin, Entomology Research Laboratory, Yakima, used X-ray machine. February 19, 1964.

H. Hollister, A.E.C., Washington, D. C. discussed research with H. A.Kornberg and F. P. Hungate on February 24-25, 1964.

G. J. Klontz, Western Fish Disease Laboratory, Seattle, presented a seminar and discussed research with Drs. Nakatani and Palotay on February 27.

P. A. J. Lister, AERE, Health Physics and Medical Division, Harwell, England, discussed long-term studies with W. J. Bair on February 28.

Twenty-five science fair participants from local schools toured the Biology facilities on February 28.

b. Visits Off-Site

2/3 - V. G. Horstman inspected feed at Pendleton Grain Growers.
2/4 - L. Eberhardt and W. H. Rickard collected samples at Wooten Game Range, Dayton, for fallout studies.
2/3-7 - R. T. O'Brien attended Aero-Space Conference at Brooks AFB, Texas.

b. Visits Off-Site (continued)

2/7-11 - R. C. Thompson gave a presentation of Hanford potentialities at a meeting of DASA officials at the Pentagon, Washington, DC. Discussed research with Drs. MacNamara, Weeks, Joffe at Edgewood Arsenal, Md. and discussed research with Palmiter and Tompkins of the FRC, Wash. D.C.

2/13-17 - L. Eberhardt and D. Watson collected samples at Packwood and Blue Mountains for fallout studies.

2/17-18 - H. A. Kornberg attended the Bio-Med Program Directors Meeting held at HASL, New York City on February 17-18, and discussed FRC report with Tompkins and Palmiter in Wash. D.C. on 2/19.

2/21 - W. H. Matchett presented a seminar at Washington State University, Pullman.

2/27 - L. Eberhardt presented an informal talk to the Washington Department of Game in Olympia on radiation in relation to plants and animals.

IV. Achievements

None

V. Honors and Recognitions

None

VI. Professional Group or Organization Assignments

None

APPLIED MATHEMATICS OPERATION

MONTHLY REPORT - FEBRUARY, 1964

ORGANIZATION AND PERSONNEL

J. LaMar Westra, Technical Graduate, went on permanent assignment in C&AO, Electronic Data Processing Operation, on February 3, 1964.

OPERATIONS RESEARCH ACTIVITIES

In line with General Electric diversification planning for the Area, work on Tri-City Area economic studies continues.

Work was begun on economic-biological interactions under the Biology and Medicine program. Meetings were held at Hanford to advance such work. This will be a continuing effort.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

N-Reactor Department

An experimental layout and procedure was developed for a proposed in-reactor fuel corrosion test.

An appropriate discriminant analysis model was developed to obtain confidence intervals for cladding thickness of "standard" fuels. An analysis on experimental data was completed and corresponding confidence intervals for each of sixteen outer fuel shells and fifteen inner fuel slugs were obtained.

Irradiation Processing Department

The results from production test-572 are being analyzed. The objectives of the PT are to determine the behavior of fuel elements with different support heights and to study the relationships between hot spots and weight loss and between weight loss and fuel element power.

A set of sequential sampling plans with their operating characteristic and average sample number curves were submitted to interested personnel for use in testing whether a new form of resistance temperature probe can withstand exposure to radiation adequately.

Results from an analysis of PT-546 were submitted to interested personnel. The results indicate at high probability levels that there are greater

[redacted]

increases in postirradiation dimensional measurements in present HDS fuel elements than in AlSi fuel elements.

Work is continuing on the reliability study of the Panellit gauge scram system with respect to both the operating continuity and the scram on demand aspects. A considerable amount of work has gone into estimating component reliability and in fitting time to failure distributions.

The analysis of a test to assess the effects of the can and core annuli thicknesses on the quality of the cladding bond is being completed. Quality is measured by the total bad discs and the total bond count.

The patterning of fuel elements in reactor channels for a production test to compare the postirradiation dimensional results of three different types of fuel elements was determined.

A study indicated that no deleterious effects as reflected in the bond integrity measurements are experienced in prolonged use of Diversey 514 as long as the pH factor is controlled.

The results of a tubing test in connection with the manufacture of HDS fuel elements are being analyzed. The test studies the effect of different plating procedures on bond strength and bond count. The effect of various combinations of 55 and 70 mil clad thicknesses on bond strength and bond count, dimensions and closure integrity are also being investigated.

Data from a 5 x 5 factorial design used in investigating the optimum regions for temperature and Pb time are being analyzed. The dependent variables are the bond integrity measurements.

Studies are being made to evaluate the performance of Altrex 1097-LF in cleaning and degreasing by comparison with the standard F process. Similar studies are also being made of a cleaning process called Oakite NST.

A statistical analysis of the dimensional effects of six different procedures used in canning HDS fuel elements is being made.

The frequency distributions, averages, and standard deviations of the maximum and minimum outside diameters of a lot of KVN fuel elements were determined as well as similar results for the fuel elements' ellipticities.

Work is continuing in the analysis of the effects of heat treatment variables in the manufacture of uranium cores on the preirradiation warp.

An additional step beyond the fitting of outlet temperature data in a tube to the circumferential position by a harmonic function and printing the resulting

[redacted]

[redacted]

predicted values for 5° intervals is being programmed. This entails selecting the maximum temperature, computing the R value and assessing the acceptability of the temperature distribution for this tube.

Work on both the reactor outage and preventive maintenance information systems continue. Both systems are in the audit and correction phase of development.

Chemical Processing Department

The use of a sequential three-observation plan with 1σ limits about specification value was recommended to replace current two-observation plan with 2σ limits in the gauging acceptance of finished components.

The use of an exponentially weighted average for calculation of normality of titrant used in determination of plutonium content of weapon components was recommended along with appropriate individual measurement control limits.

The analytical data obtained by five independent laboratories was analyzed for an evaluation of the suitability of three compounds as primary chemical standards for plutonium content.

An informal discussion was held regarding the future work required to construct a model of the Z-Plant plutonium material balance system for accountability purposes.

Contract and Accounting Operation

Consultation services were given in connection with a test to compare four different types of memory tapes used by the IBM-7090 computer. The resulting data are being analyzed.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HL

2000 Program

A study is being made of the monitor system for the GE-412 computer. This system is a group of interlinking programs which coordinate the real time aspects of the 412. Major revisions and, in some cases, complete rewriting of the system will be needed for the first process control application of the computer, the controlling of a C-type experimental separations column.

Work continued on the statistical analyses of gamma absorptiometer data in connection with the rewriting of the paper, "A Data Logging Absorptiometer for Routine Uranium Analysis" for publication in Talanta. Further investigation was done of the numerical problems associated with the calculation of

the solution of the nonlinear system of differential equations which explain mass transfer in the column.

Continued assistance was provided on formulating and fitting mathematical models for radionuclide retention in reactor effluent streams.

The first of the three phases of the EDPM "shoe-box" model of ground water flow has been completed and is functioning.

Additional work has been done on the problem of estimating the total amount of Pu in the Recuplex crib based on Pu density estimates down to a depth of eight feet.

Work is being done on fitting 18 particle size distributions. Based on the criterion of an acceptably small chi-squared value in assessing the agreement between observed and estimated class frequencies, none of the distributions (Weibull, log-normal, mixed exponential) have proved adequate.

3000 Program

Four magnetic tapes were produced by the EDPM program SPIG for checking and demonstration purposes on the prototype numerically controlled rotary contour gauge.

A mathematical study was made to study the sources and magnitudes of errors resulting from postulated misalignment of gears on a rotary contour gauge.

A document, HW-78712, entitled, "A Program for the Numerical Control of a Pulse Increment System" has been issued. This document describes the theory behind a new and experimental system of numerical control, and lists the EDPM program CUPID which produces magnetic control tapes for this process.

4000 Program

Expressions were worked up for estimating the cumulative propagated error associated with calculated percentages of constituent gases.

Metallurgically-oriented expressions were employed for elastic and plastic strain rates to derive (analytically) stress in metals as a function of the elapsed time following discontinuance of a mechanically applied strain. An iterative procedure was outlined for obtaining approximate least squares estimates of the constants in the derived equation.

Work continued on a formal Hanford report describing the quantitative metallographic techniques developed during the past years for estimating

[redacted] [redacted]

the properties of second phase spherical particles embedded in a matrix. Work continued on the modification of the program for analyzing metallographic data collected on the Ziess particle size analyzer.

Solutions were obtained to two partial differential equation and boundary value problems concerned with heat conduction and waste disposal.

Discussions were held on the possibility of constructing mathematical models to describe the propagation of cracks in elastic solids.

Several discussions on the uses of dimensional analyses, similitude, and the theory of models have been held in connection with equipment designs for the reactor containment program.

5000 Program

Mathematical services were provided on curve fitting techniques to be used in metal-through-metal diffusion studies.

Work continued on the calculation of the power function of the Poisson index statistic used to check stability of counting instruments. Power calculations were started for the alternative of a linear drift during the counting periods assuming a fixed denominator in the index and a normal approximation for the Poisson variables. An existing incomplete gamma function FORTRAN program is being revised to extend its calculation range to handle chi-square distributions with extreme degrees of freedom.

The main calculation program of the IRA system was modified to permit the use of several models in fitting gamma ray spectrometer data. Test runs indicate significant improvement in results over the single model mode of operation. Further modifications are in progress to provide a better procedure for selecting the model with the best fit.

IRA file data is being used to analyze the consistency of nuclide energy distributions within different mode code detector combinations. The results of the analysis should help in quantifying the stability characteristics of the various detector systems.

6000 Program

The statistical analysis of data from a study to investigate and compare the Pelecyphorous and Stenomorpha beetle populations in sagebrush and greasewood communities was completed.

A statistical analysis of data from the zinc-cadmium interrelationship study was initiated. Comparison of the percent dose of Zn^{65} with Cd^{115m} tracer

retained by various rat tissues for several Zn^{65} dose levels will be made.

OTHER

An experiment was designed to determine the precision of piezometer measurements in project wells.

Completed the statistical analysis of "questionable" well data from the well survey study. The purpose of the analysis was to calculate the precision with which the location and elevation of project wells can be determined. The information is needed to assess statistically the several sources of variation that may enter the in-place measurement of the permeability distribution of the project.

Work is being done on programming a procedure for estimating orthogonal polynomial effects in the case of nonequally spaced independent variables.

Carl A. Bennett

Manager
APPLIED MATHEMATICS

CA Bennett:dgl

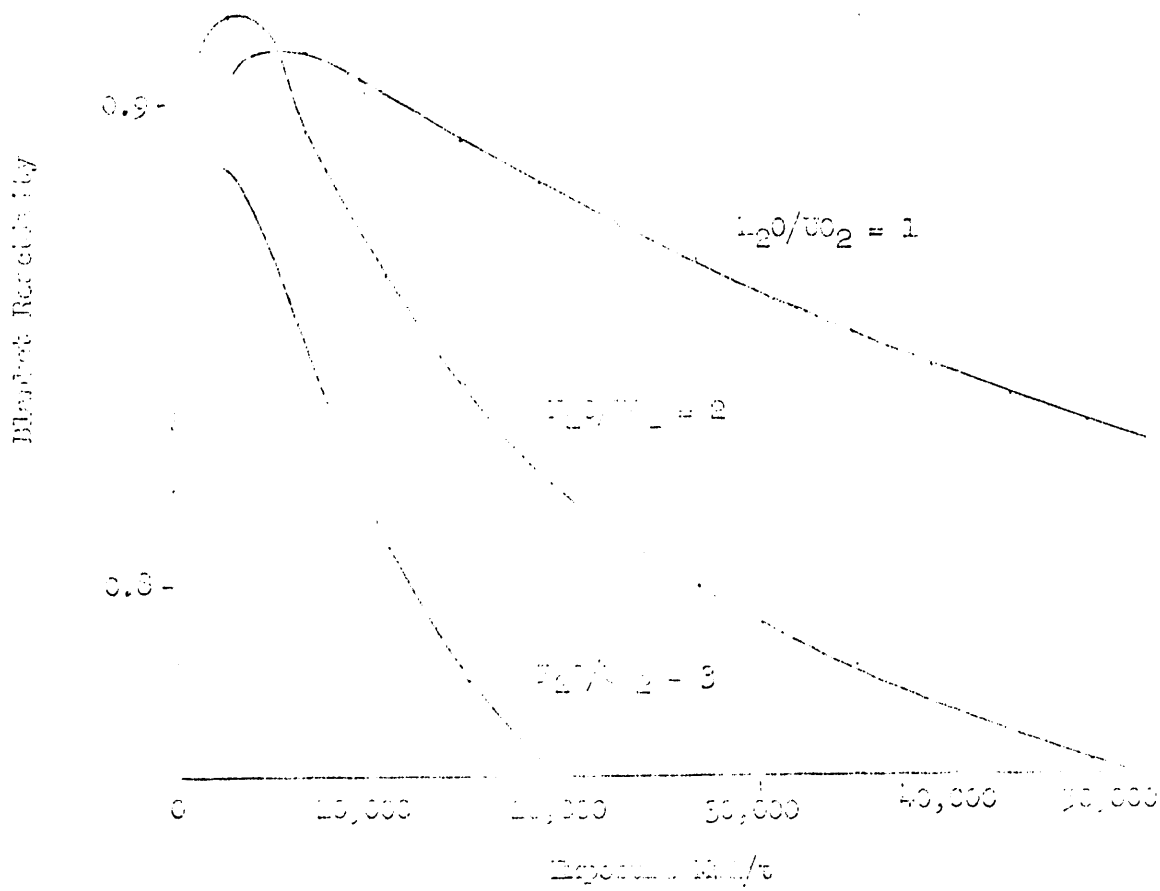
PROGRAMMING

REACTOR PROGRAM - 04 PROGRAM

PLUTONIUM UTILIZATION PROGRAM

Spike Enriched (Seed-Blanket) Reactor

A factor of interest relating to the seed-blanket reactor concept is the conservation of uranium ore relative to a uniformly enriched reactor. A measure of ore utilization would be the megawatt days of exposure per ton of natural uranium fed into the diffusion cascade-reactor complex. If one assumes that 800 megawatt days of heat may be obtained from each kilogram of U-235 in fully-enriched uranium and notes that 4.17 kilograms of U-235 may be obtained as fully-enriched uranium per ton of natural uranium based on the present schedule, the exposure may be computed to be only 3340 megawatt days per ton of natural uranium in the seed. However, in the blanket, plutonium is formed nearly as fast as the U-235 is burned out so the net consumption of fissile fuel is negligible. Thus the exposure attainable is essentially 3340 divided by the fractional amount of power produced by the seed. As was noted in the January Monthly Report (HW-80560 F), if the infinite multiplication factor for the blanket is about 0.8, the minimum seed power fraction is about 0.46 when the seed occupies about 0.2 fraction of the reactor volume. In this case the exposure attainable would be about 7300 Mwd/ton of natural uranium. If the blanket reactivity is about 0.9, the minimum seed power fraction is about 0.33 at about 0.1 volume fraction. The exposure attainable would be about 10,100 Mwd/ton of natural uranium. If the blanket reactivity is greater than 0.99, the minimum seed power fraction would be very small and essentially complete use of natural uranium is achieved. In practice, blanket reactivity is not constant as exposure is accumulated. The variation for various water-to-fuel ratios in a light water moderated blanket is given on Figure 1. Reactivity first rises as the high cross section plutonium is formed and then falls as fission products accumulate and the U^{235} burns out. Except for very brief periods, the blanket reactivity for the smaller water-to-fuel ratio is always above 0.8 which indicates that the attainable heat is greater than 7300 Mwd/ton of natural uranium consumed by the reactor and the diffusion cascade. Since minimum fuel costs occur at blanket exposures of 30,000 to 40,000 Mwd/ton, one may observe that the reactivity at any water-to-fuel ratio is greater than 0.9 for only a portion of the lifetime. Therefore, the maximum attainable heat is greater than 10,000 Mwd/t of natural uranium consumed only for moderate exposures which result in high fuel costs.


0.9
0.8
$H_2O/UO_2 = 1$
$H_2O/U = 2$
$H_2O/U = 3$
0
10,000
20,000
30,000
40,000
50,000

Two series of calculations have been performed on seed-blanket modules of 20 centimeter radius, one with 15% and the other with 20% seed by volume. The seed was 93% enriched uranium and the blanket was natural uranium enriched with 0.3 to 0.35% plutonium to obtain an initial blanket reactivity of 0.99. The accumulated blanket exposure at the end of the various seed lifetimes is given in Table I. Note that each successive seed is less effective in driving the blanket due to the decrease in blanket reactivity.

TABLE I

CUMULATIVE BLANKET EXPOSURE

Number of Seeds	15% Seed Volume	20% Seed Volume
1	18,000	20,700
2	27,500	33,500
3	34,700	42,400
4	41,100	51,100
5	47,000	59,000
6	52,300	
7	57,200	

Using the assumption that a gram of fissile plutonium is equivalent to a gram of U^{235} to determine the net consumption of natural uranium by the reactor and associated diffusion cascade, the variation in Mwd/t of natural uranium presented in Figure 2 is obtained. The decrease in effectiveness in driving the blanket is reflected in the decrease in heat produced per ton of natural uranium. A similar calculation was also made for a uniformly enriched reactor. These data are also presented in Figure 2. The variation of exposure from 10,000 to 50,000 Mwd/ton was obtained by increasing the initial enrichment from 1.4 to 3.3%. Note that as the enrichment is increased, the attainable heat per ton of natural uranium decreases. This is due to two factors at the higher enrichment; first, less of the original U^{235} fed to the diffusion cascade is in the reactor fuel; and second, less heat is produced per net fissile atom destroyed, because of the lower ratio of fertile-to-fissile macroscopic cross section. Note, also, that attainable heat from the uniformly enriched reactor is greater than for either of the seed-blanket reactors studied. The tons of natural uranium required to produce either a reactor charge of uniform enrichment above natural, or a reactor charge of seed in a seed-blanket reactor, decreases as the price of natural uranium increases because the optimum uranium cascade tails composition decreases accordingly. Thus, the heat obtained

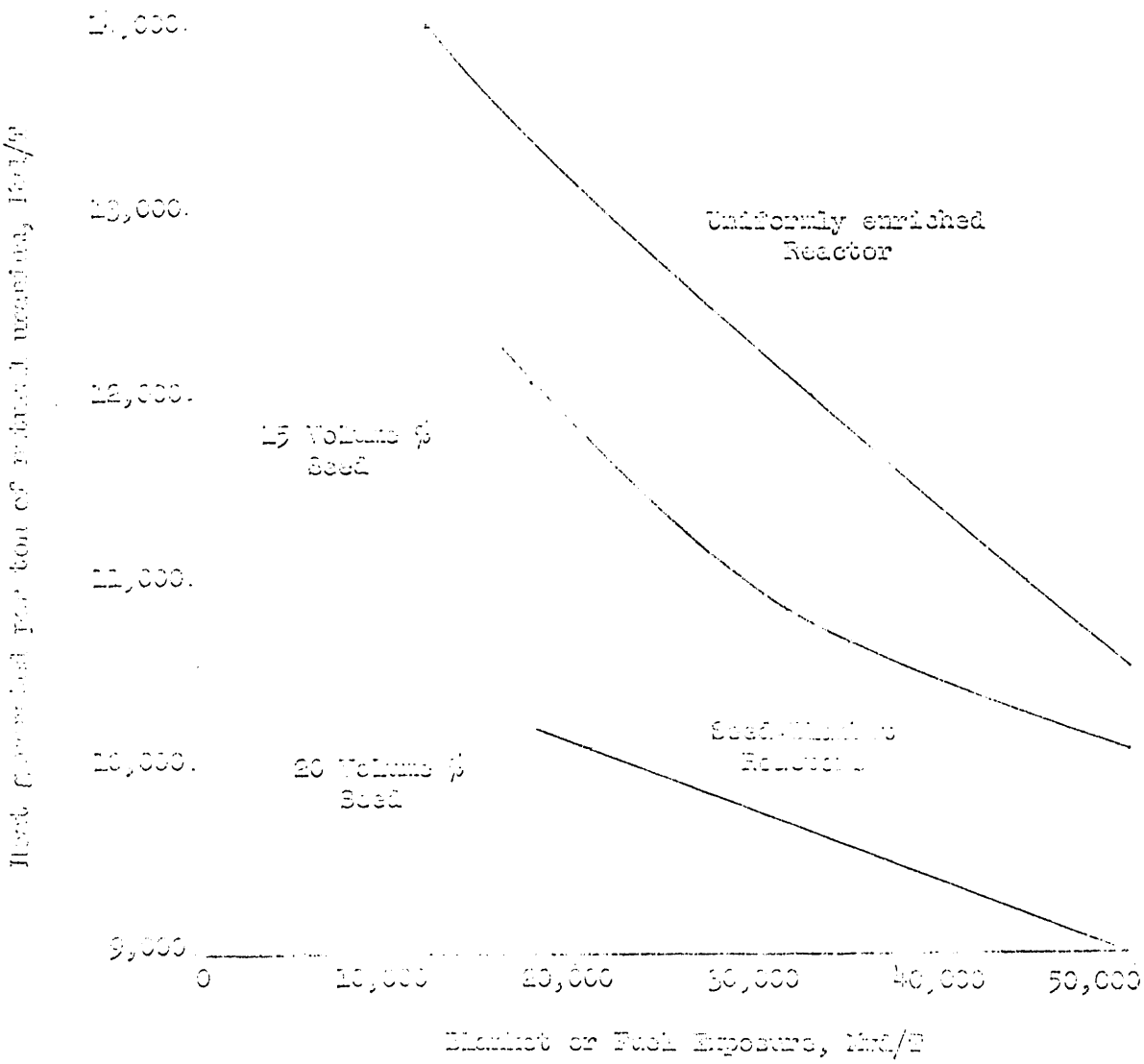


FIGURE 2

UNRECOVERABLE HEAT FROM NATURAL URANIUM
AS A FUNCTION OF BLANKET EXPOSURE

from a ton of natural uranium increases as the price of uranium goes up. The fuel cost is also altered such that the minimum occurs at a lower exposure which also improves the utilization. Similarly, a decrease in jacketing cost will improve utilization by shifting the minimum fuel cost to lower exposures.

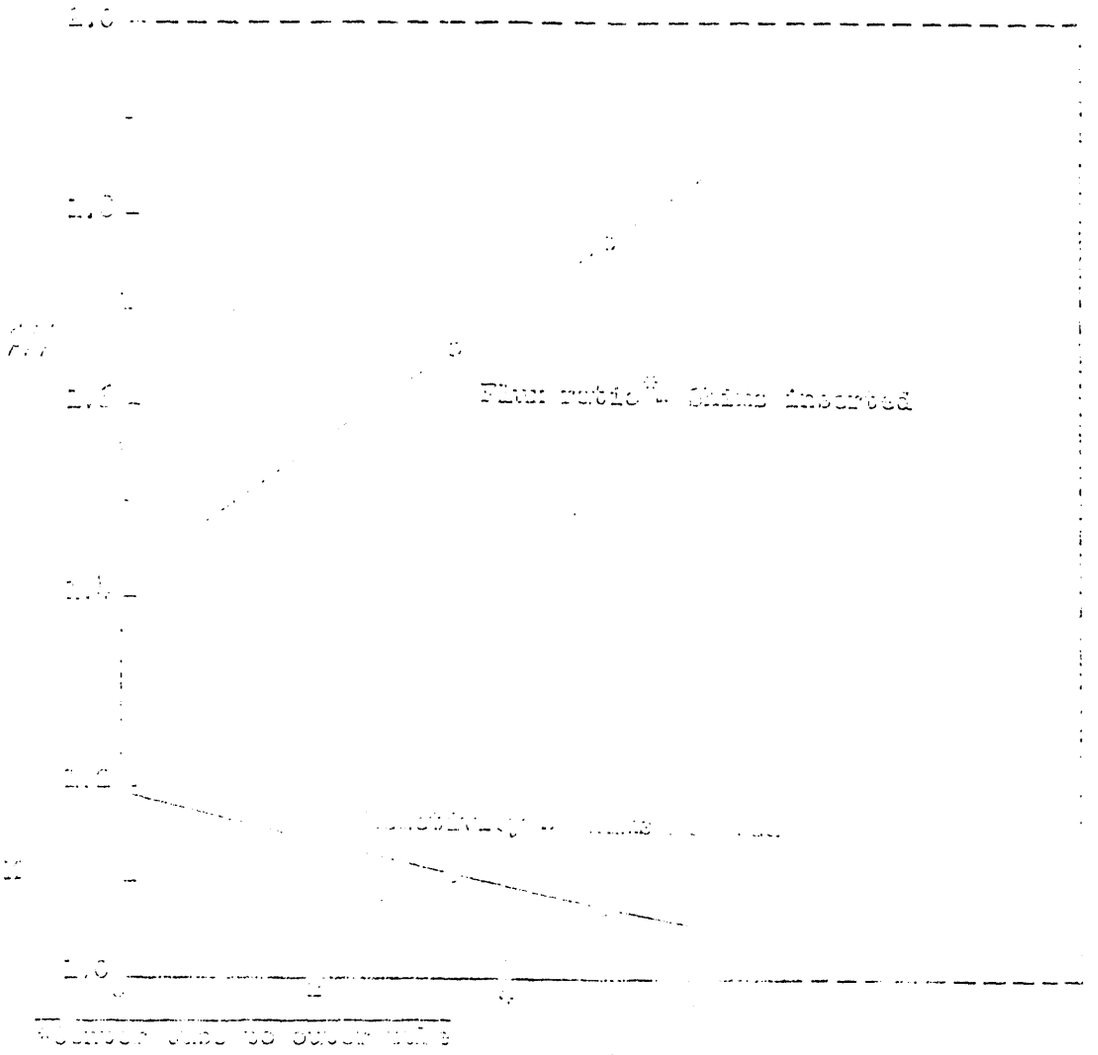
The assumption of the gram-for-gram equivalence of fissile plutonium and U^{235} will be checked by recycling all plutonium to the reactor and computing the required natural uranium to supply a cascade-reactor complex.

PRTR Fuel Loadings

Several one-dimensional burn up calculations were made using the ALTHAEA code to estimate the behavior of the PRTR at various loadings. It was found that the thermal flux is essentially uniform with fuel in 85 tubes, while a smaller number caused a thermal flux peak in the center of the reactor. The flux ratio (center tube to outer tube) is dependent upon the relative strength of shims and control leakage or buckling used. Figure 3 illustrates this variation for a 55-tube case using 1.3a/o enrichment* and uniform shim strengths. The buckled cases were run by Hanford Physics and Instruments Laboratory personnel. The lower curve showing reactivity decrease as buckling increases (lower moderator level) provides a measure of the excess reactivity that can be carried in shim rods for flux flattening. The upper curve shows the resultant flux ratio when shims have been inserted to bring the reactivity to 1.0. Shim strengths were equal over the regions containing fuel, so presumably, better flux shape could be obtained by more judicious use of the shims. Another calculation indicated that the thermal flux would be essentially flat if 70 to 80 milli k of excess reactivity is held in the center six shim rods and the balance of the control is obtained by moderator level adjustment. One may conclude from these studies that adequately flat flux distributions can be obtained with fuel in 55 tubes using shims in the proper strength and location, and in conjunction with neutron leakage, controlled by varying the moderator level.

Several calculations of unusual PRTR fuel configurations were made to explore ways of providing a high flux for a test fuel element. The first case assumed the center process tube to be cooled with light water and surrounded by a one-inch thick light water jacket. This calculation only partially demonstrated the desired results since

* Plutonium added to natural uranium to bring the fissile enrichment level to 1.3 atomic percent.


2.0
1.8
1.6
1.4
1.2
1.0
Fluor rabbit bone absorbed

the enrichment selected for the test element was nearly the same as for the remainder of the reactor and the specific heat produced was only 20% more than in the adjacent six elements. The flux "seen" by the test elements was only 70% greater than that "seen" by the adjacent six elements. Presumably an increased enrichment of the test element would increase the flux peak with only minor disturbance of the flux in the adjacent tubes.

Two configurations using only D_2O were considered. In both cases, the six process tubes adjacent to the test element contained Pu-Al driver elements. These seven tubes were surrounded by 12 or 30 vacant tubes as a buffer region from the 48 normal elements. The 12-tube buffer permitted an unpurturbed flux ratio of a factor of 4, while the 30-tube buffer permitted an unpurturbed flux ratio of a factor of 20 between the center seven elements and the 48 elements in the annulus. Presumably, it is not necessary to locate the test elements and its drivers in the center of the reactor to obtain most of this flux peaking effect.

Crossed Progeny - U^{233}-U^{238}

In the course of updating the MELEAGER cross section tape, the literature was surveyed by Physics Laboratory personnel for the latest resonance integral data for U^{233}. While not fully evaluated by Physics personnel, MELEAGER cross sections were prepared for U^{233} from the following sources:

Absorption cross section,	σ_a 2200 = 575 barns[a]
Fission cross section,	σ_f 2200 = 526 barns[a]
Infinite absorption resonance integral,	$I_{\infty a}$ = 1200 $\pm$ barns[b]
Infinite fission resonance integral,	$I_{\infty f}$ = 743 $\pm$ 24 barns[c]

The major change in these numbers from the generally used numbers is in the $I_{\infty f}$ which was formerly 900 $\pm$ 100 from Reference b. The credibility of the new numbers must be fully established at an early time as will be indicated by the following discussion which may prove to be a pessimistic evaluation.

(a) Leonard, B. R., Jr., Survey of the Status of Low Energy Cross Sections of Fissile Nuclides, HW-69342, April (1961)
(b) R. W. Stoughten and H. Halperin. "Heavy Nuclide Cross Sections of Particular Interest to Thermal Reactor Operation: Conventions, Measurements and Preferred Values." Nuc. Sci. & Eng. v.6, p. 100. (1959)
(c) C. B. Bingham. Fission Resonance Integrals of U^{233}, U^{235}, Pu^{239}, and Pu^{241}, EANDC-Can-19, (October, 1963).

Using these latest data, alpha values for U^{233} were prepared and are presented in Table II from which the corresponding values of eta were prepared for comparison with U^{235}.

From these data, it appears that the superiority of U^{233} on the basis of eta may be limited to soft spectrum, low neutron temperature reactors. At a value of 4 greater than about 0.2, if these physics data prove to be substantially correct, U^{233} has no superiority over U^{235} in terms of its eta value and U^{235} would appear to become superior to U^{233} at r values of 0.30 and larger. At lower neutron temperatures, the foregoing points will shift to somewhat higher values of spectral index, r.

Tentative α values for U^{233} for various neutron temperatures and spectral indexes are shown in Table III.

TABLE II

ALPHA (α) AND ETA (η) OF U^{233} AS A FUNCTION OF NEUTRON SPECTRAL INDEX

Neutron Temperature = 360°C

	U^{233}		U^{235}	
* r	α	η	α	η
0.05	0.14	2.29	0.21	2.07
0.10	0.19	2.12	0.22	2.06
0.20	0.26	2.02	0.25	2.01
0.30	0.31	1.93	0.28	1.96

* r = Westcott Spectral Index
α = capture cross section divided by fission cross section
η = neutron reproduction coefficient, neutrons liberated per neutrons absorbed

TABLE III

TENTATIVE ALPHA VALUES FOR U^{233} AS A FUNCTION OF NEUTRON TEMPERATURE AND SPECTRAL INDEX BASED UPON INCOMPLETELY EVALUATED RESONANCE INTEGRAL DATA

Spectral Index r	Neutron Temperature 20°C	360°C	600°C
0.05	0.13	0.13	0.14
0.10	0.16	0.18	0.19
0.20	0.22	0.26	0.27
0.30	0.28	0.32	0.34

Many applications of U^{233} are limited to soft neutron spectra (well-moderated D_2O or graphite machines) for which α will be satisfactorily low.

This new look at physics data for U^{233}, if substantiated, means that the use of U^{233} with U^{238} may not be as favorable as previously believed. However, the foregoing results must be considered most tenuous and possibly pessimistic because of the uncertainty in the resonance integral values of U^{233}, and because self-shielding has not been fully evaluated. These numbers may not be firmed up until reactor criticality studies are made with close packed lattices.

Successive Plutonium Recycle - Pu^{242} Problems

Possible methods of maintaining a high plutonium value after a number of recycles in thermal reactors are being investigated. Because of the parasitic effect of Pu^{242}, as the concentration of Pu^{242} increases plutonium batch values decrease. As a consequence, methods of either controlling the Pu^{242} buildup or showing that it may have value are of interest.

One method of minimizing the concentration of Pu^{242} build up in the case of successive recycle is to sell or discard a portion of each plutonium discharge batch. This reduces the amount of all isotopes recycled and, since the Pu^{239}, Pu^{240}, and Pu^{241} will return to equilibrium concentrations in the subsequent cycle, the relative concentration of Pu^{242} present is reduced. Table IV demonstrates this effect in a boiling water reactor simulation.

TABLE IV

SUCCESSIVE PLUTONIUM RECYCLE WITH INTERMEDIARY PLUTONIUM SALE TO REDUCE Pu^{242} BUILD UP

Through Five Recycles Cycle No.	No Plutonium Exported 3% Plutonium Lost			27% Plutonium Exported (Sold) 3% Plutonium Lost		
	Cumulative Cycle Exposure	Pu^{242}, %	Pu Value ($/g)	Cycle Exposure	% Pu^{242}	Pu Value ($/g)
1	22,300	8.3	6.70	22,260	8.2	6.71
2	45,900	16.8	5.50	45,750	14.9	5.70
3	69,590	22.2	4.80	69,700	18.1	5.35
4	93,280	25.6	4.30	92,850	19.4	5.30
5	116,600	27.8	4.20	117,150	20.1	5.20
6	140,050	29.3	4.00	141,380	20.2	5.20

A second method of maintaining low Pu^{242} concentrations is to isolate the newly formed plutonium in a fuel element from the plutonium isotope which existed in the fuel initially. The newly formed isotope will have a very low Pu^{242} concentration; whereas, the old fuel will be much higher. All, or a portion, of this high Pu^{242} content plutonium can be sold to (1) another thermal reactor which is less sensitive to Pu^{242} parasitic action, (2) fast reactors in which Pu^{242} is fissionable, or (3) as a source of higher isotopes. Table V shows the plutonium composition initially and at discharge for a Pressurized Water Reactor Simulation. Note that the plutonium value in the last case would be $1.20 per gram higher if the high Pu^{242} content fuel were isolated.

If isotopes formed from neutron capture in Pu^{242}, such as Cm^{244} were used widely as heat sources, plutonium values would be enhanced. Calculations presented in the October, 1963 Monthly Report indicated that a Cm^{244} price of $100/gram would result in a Pu^{242} price of $10/gram which would obviously help maintain a higher plutonium value.

Power Recovery from the Hanford Production Reactors

Some power-conversion cases for the small Hanford production reactors (including multi-product operation) have been examined on the basis of once-through water coolant, and without overboring the graphite channels. Thermal-to-electrical efficiencies of up to 20% appear possible, and preliminary fuel cycle costs appear satisfactory. These cases require relatively low capital costs, compared with overboring and installation of pressurized recirculating coolant.

TABLE V

PLUTONIUM COMPOSITION AND VALUE FOR NEWLY FORMED PLUTONIUM

AND PLUTONIUM FORMED FROM PLUTONIUM IN THE INITIAL FUEL

Fuel	Fuel Exposure Mwd/ton	Plutonium Composition Initial 239	240	241	242	Plutonium Composition Newly Formed 239	240	241	242	Pu Value $/g	Plutonium Composition Formed from Initial 239	240	241	242	Plutonium Composition Total in Fuel 239	240	241	242	Pu Value $/g
Slightly-Enriched Uranium	19,373	0	0	0	0	47	32	12	8	7.40	0	0	0	0	47	32	12	8	7.40
						5,335 grams/ton									5,335 grams/ton				
Plutonium in Natural Uranium	21,318	95	5	0	0	46	35	12	7	7.60	2	36	21	40	35	34	14	13	6.60
						5,690 grams/ton					1,108 grams/ton				6,798 grams/ton				
Plutonium in Natural Uranium	21,764	76	18	5	1	46	35	12	7	7.60	1.5	33	20	46	36	34	14	15	6.40
						5,760 grams/ton					1,522 grams/ton				7,282 grams/ton				

An additional case employing flash boilers with the condensed water preheating the reactor coolant may merit consideration. Although the thermal-to-electrical conversion efficiency would be lower, its simplicity and lower capital conversion cost may give more attractive over-all economics for multi-product operation plus some revenue -- producing power from the small Hanford production reactors.

CODE DEVELOPMENT

SUPERCASE GENERATOR

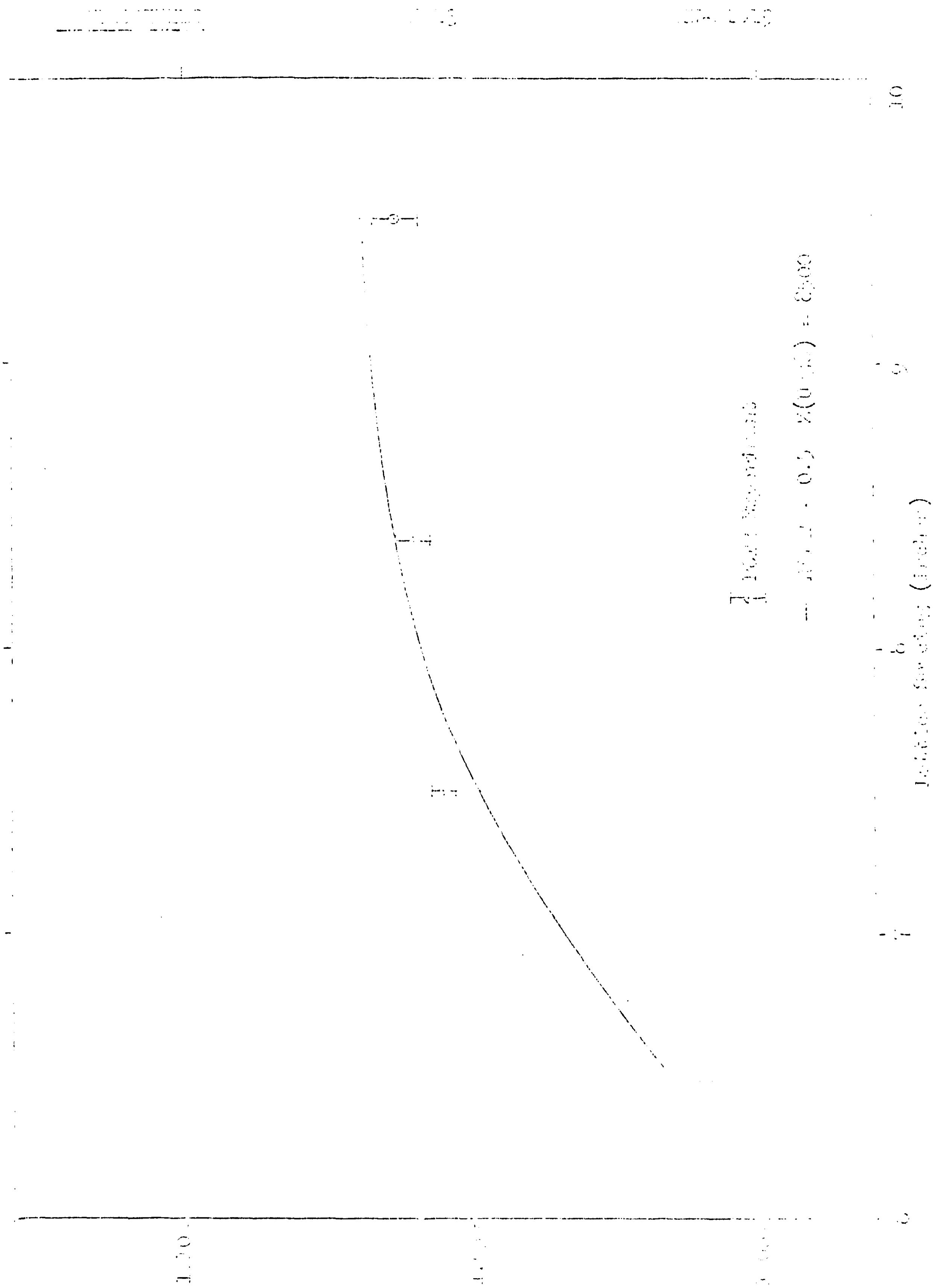
Changes have been made in SUPERCASE GENERATOR, a code which automatically generates data for fuel cost minimization studies, such that the method of changing the data is now compatible with that of the other case generating systems used by Programming. Also, an improved method has been devised for handling discontinuities in curves of exposure versus enrichment for breeder or near-breeder thermal reactors. The problem arises because economics calculations are made on reactivity limited exposure; whereas, some fuels will run to exposure well beyond the durability of present fuel elements even at very low enrichments. So, SUPERCASE now will automatically run some "exposure limited" cases to fill in economic data at lower exposures.

QUICK

Changes in the methods for calculating interest charges have been made in the QUICK economic code. Interest charge calculations no longer need be based on payment every six months as presently required by the AEC. The code is thus more readily adaptable to studies involving private ownership of reactors and nuclear fuels where the payment period may be different from six months.

JASON-MELEAGER

Calibration of the JASON-MELEAGER codes on graphite moderated systems has been completed. Using the values of f, e, and SDPV (determined by the JASON code) in the MELEAGER code values of k_{oo} were calculated for various lattice spacings. Using a value for the RAYK1 calibration factor of 0.5 and $Z(U^{238}) = 8500$, the curve of k_{oo} vs. lattice spacing shown in Figure 4 was obtained. Although the computed values of k_{oo} do not lie on the experimentally determined points, the curve obtained appears to be the best one obtainable without adding additional "calibration factors" to the MELEAGER code. No further investigation of graphite moderated systems is planned at this time as the present factors can be adjusted for specific graphite reactors to match the characteristics more closely than this general calibration shows.



Infinite Multiplication Factor

COMPARISON OF K

Uranium Resource Conservation (YUKON)

The YUKON code is being revised to include output of the results of diverse intermediate calculations and to provide the capability of curve fitting recycle data so that various hypothesized expanding economies can be easily studied.

Nuclear Safety Activities

Work was completed on "The Consequences of Accidental Release of Radioactive Cerium During Shipment", HW-72163 REV1, January 30, 1964. A companion memorandum was completed evaluating the risk of shipments of as much as one million curies of cerium-144.

A document was completed, HW-80865, "Control of Fissile Materials". A draft HAPO Policy Guide is in process for implementing the procedures which apply to shipment, receipt, change of custody and labeling of fissile materials. These procedures were prepared by representatives of the several Departments concerned and have been updated in light of recent changes in shipment regulations.

F. W. Woodfield
Manager-Programming

FW Woodfield:ch

RADIATION PROTECTION OPERATION REPORT FOR THE MONTH OF FEBRUARY 1964

A. ORGANIZATION AND PERSONNEL

Effective February 3, 1964, J. J. Jech transferred from the X-Ray Department, Milwaukee to Radiation Monitoring as Supervisor. D. O. Erickson, Engineer, joined Radiation Monitoring on February 3, 1964, and R. D. Burnett, Engineer, on February 17, 1964. L. G. Faust transferred from Radiation Monitoring to Radiological Development and Calibrations. Darlene W. Jones replaced Betty R. Kent effective February 17, 1964, as Secretary, Internal Dosimetry.

B. ACTIVITIES

Occupational Exposure Experience

There were four new plutonium cases confirmed by special bioassay analysis during the month. All cases were estimated to be less than 1% of the maximum permissible body burden (MPBB, plutonium with bone as reference, is 0.04 μc). Three cases resulted from inhalation during fires in the Weapons Manufacturing Facilities (234-5 and 232-Z), and the fourth case resulted from absorption through the skin when an employee received plutonium nitrate contamination of 20,000 d/m while working in the Purex Processing Facility (202-A).

The total number of individuals who have received internal plutonium depositions at Hanford is 338. With the termination of one employee having confirmed plutonium deposition, there are currently 245 employed.

In February, there were three incidents involving five employees which required special bioassay sampling for plutonium analysis. The following is a brief description of the more significant incidents.

A CPD operator bumped his head on a glove port shield which was contaminated to 1000 d/m plutonium, non-smearable, while working in the Weapons Manufacturing Building (234-5). Examination at the plutonium wound counter showed results less than the detection limit of 1×10^{-4} μc.

Two HL employees were operating an isostatic press in the Plutonium Metallurgy Research Facility (231-Z) when a high pressure line carrying plutonium contaminated hydraulic fluid ruptured. A fine mist sprayed the room causing general contamination to 40,000 d/m. No skin or clothing contamination occurred; however, air sample results indicated airborne concentrations of 1.8×10^{-10} μc Pu/cc based on a four-day collection period.

In addition to the incidents involving plutonium, there was one incident involving an HL employee working in the basement of the Radiochemical Research Laboratory (325) that required evaluation for possible intake of thorium. Wrist contamination of 5000 d/m occurred when the outgassing tube of an aluminum can containing thorium clogged while being heated. The resulting build-up of pressure forced thorium out of the can, contaminating the room to 60,000 d/m. No air sample was taken. Preliminary evaluation of an examination in the Whole Body Counter indicates a deposition less than ten percent of the maximum permissible body burden (MPBB, natural thorium with bone as reference is 0.01 μc).

There were three incidents of special interest which involved exposure of Hanford employees from external sources. These are summarized below:

An HL employee received localized exposure (60,000 c/m) to his left back shoulder from particulate contamination on the neck of an undersuit he was wearing. The contamination was detected during an exit survey following work in the 325 Laboratory Facility Decontamination Room. The employee's supervisor and Radiation Monitoring do not believe that the contamination could possibly have occurred from the work performed that day. It is possible the employee was exposed to the source for a period up to 48 hours. A film study has been made and the localized exposure received will be evaluated.

An HL chemist worked, without monitoring, in an unknown field of radiation while transferring waste into a cardboard container from a hood in the Chemical Effluents Technology Laboratory (222-U). The lower portion of his body received a dose estimated at 630 mrads including 130 mR greater than the 20 mR measured by the beta-gamma film badge dosimeter he was wearing.

At the 105-H Reactor, a fuel element was accidently removed from a reactor tube during splining operation and fell into the charge elevator pit. A measurement of 3 R/hour was obtained from the top of the unit. The fuel element was recovered using remote equipment and personnel exposure was estimated at 45 mR for this period.

Environmental Experience

Concentrations of fallout materials in the air of the Pacific Northwest increased slightly in early February due to an influx of long-lived fission products from past nuclear tests. The influx, which is normal for this time of year, raised the monthly average value to 0.8 pc β/m^3 of air from the record low value of 0.6 pc β/m^3 observed last month.

PRTR Operation released less than half a curie of mixed fission products to the river while flushing the storage basin. The special release was controlled under the provisions of a Deviation from Process Standards.

Material for the annual report on liquid waste disposed to ground in the 200 Areas was compiled. The total sent to ground was approximately 62×10^3 curies for 1962 and 36×10^3 curies for 1963. The bulk of the radioactive material was Ru^{106}. Hanford Laboratories contributed approximately 10^3 curies in 1962, and 1.8×10^3 curies in 1963.

Studies and Improvements

The neutron spectrometer was moved to a location on the X-1 level near the C-2 test hole at the K-West reactor. The Li^6 neutron detector was used at the test hole so that the gamma interference could be eliminated. The neutron spectrum at the test hole is typical of a degraded fission spectrum. The average neutron energy is between 0.5 and 0.8 Mev with very few neutrons above 2.0 Mev. The calibration of the Li^6 detector was checked eight months ago and will be checked again to confirm these energies. This spectrum should be typical of neutrons streaming through tubes filled with water.

Several sets of thermoluminescent phosphor were exposed and read at the high sensitivity settings of the reader. All of these sets showed a neutron sensitivity of about 50 units/rad. A radium-gamma exposure was completed for two commercially available phosphors. Exposures were also made to 1 rem and 4 rems of PuF_4 neutron; these results showed rather poor reproducibility. Some of the difficulty may be due to a faulty timing motor in the thermoluminescent dosimeter reader. The motor was repaired and a new one ordered. It appears that the optimum operating point for the reader is yet to be found.

A large sample of thorium oxide from a lot being processed was obtained for making surface dose rate measurements. The sample was bagged in plastic and an absorption curve was plotted to obtain the unshielded dose rate. After plotting and extrapolating to zero plastic thickness, a dose rate of 81 mrads/hr was determined. Smaller samples were obtained from the same lot, and alpha and gamma energy analyses were made. Both energy spectrums indicated the material was relatively old. A chemical analysis was made to determine the Th^{232} to Th^{228} ratio; it was 2:1. By calculating the decay growth of Th^{228} from a time zero equal to the time of separation, the age of the material is estimated to be 2.2 or 9 years old.

Work continued in evaluating Lexan plastic as a possible neutron dosimeter. Some U^{235} (24 gm. 95%) was obtained and rolled into thin, about 30 mils, plate to be used as the fissioning material. Preliminary exposures to thermal neutrons indicate that the number of observed

tracks per microscope field for a 1 mrem slow neutron dose is about 4 tracks. Only one field need be read to determine a reasonable thermal neutron dose. Ten mil Lexan plastic appears to be best suited to the present technique. The plastic is stiff and thick enough for ease of manipulation under a microscope. The etch pits can be easily seen using a microscope with 400 power.

The requirements for sampling of air, milk, pasture grass, produce, and meat surveillance during 1964 were prepared. The most significant improvement requested was the measurement of moisture content of all grass samples, so that fission product concentrations could be reported on a dry weight basis. Past experience indicates that local pasture grass samples contain about 80% water when freshly cut.

The following steps were taken in the improvement of the off-site environmental air sampling network: (1) closed out the stations at Boise and Lewiston, Idaho and Klamath Falls, Oregon; (2) installed an air sampler at the Pendleton, Oregon, airport; (3) exchanged and recalibrated equipment at Walla Walla.

The two-week exchange schedule for neutron dosimeters was extended to a four-week schedule February 20. This change was incorporated as a result of a study which demonstrated increased sensitivity over the four-week period in spite of the slight increase in track fading. The film badge dosimeters, neutron dosimeters, and the finger ring dosimeters are now exchanged on the same dates.

The form used to obtain information concerning HAPO visitors using a film badge dosimeter was revised and placed into effect February 14. The new form was necessary to provide information required by and for the AEC in AEC Manual Chapter 0525. The information required is the name and address of the visitor's employer, the dates the dosimeter was used, and the visitor's social security number.

The technique for reading alpha tracks on sample slides was revised. This revision involves the use of lower microscope magnification, elimination of the rectangular viewing slots, and a complete revision of the counting techniques. The same total area is examined for alpha tracks as has been done in the past, but with the above mentioned changes, it is possible to do this with a single microscope setting instead of nine different positions. Also, a greatly simplified standard source counting procedure was developed. As a measure of the efficiency of the method, it should be noted that during January (primarily using the old method) 163 samples were read. During February, with the new method in use, 630 samples were read. Although some additional time was devoted to reading slides in February, the great majority of the increase in output was due to the new reading technique.

Arrangements were made to take the Mobile Whole Body Counter to Naturita, Colorado, during the first week of April. The unit will be used for experiments to be conducted by Hanford Laboratories. The unit will suspend counting in 100-B Area on March 27, and resume approximately April 15.

A monograph on I^{131} is to be prepared for the American Industrial Hygiene Association by three Hanford Laboratories' authors, including Mr. J. K. Soldat of the Radiation Protection Operation. AEC approval for this project was obtained this month. The first draft of this monograph is to be completed by the end of 1964.

Research Studies

Effect of Reactor Effluent on the Quality of Columbia River Water (02)

Outlet temperature measurements from several 107 reactor effluent detention basins have shown heat losses during the winter larger than indicated by previous scattered measurements. The loss of heat from these basins would make a measurable difference in over-all reactor heat contribution to river temperatures.

Temperature traverses ceased during the month with the failure of the resistance thermometer, requiring factory repair, and structural failure in the large boat. Point temperature collection continued, including a new off-shore station at Hanford from an anchored barge.

A floating water sampler has been fixed in the Columbia River about one mile below the Hanford Ferry location. The device has a paddle wheel collector which obtains a continuous sample. A continuous temperature measurement of river water is also being made by using the water sampler as a platform and support of the temperature probe and recorder.

Nuclear Facilities Monitoring Guide

Work performed to date has delineated the environmental monitoring aspects of the Guide. Starting in March, emphasis will be placed on the effluent monitoring portion of the Guide, with particular reference to compliance with specific regulations for licensees.

C. RELATIONS

AEC-RLOO issued a request for expressions of interest from commercial firms in performing certain radiation protection services including the processing of film badges, neutron badges, and finger rings; the collection and laboratory analysis of bioassay samples; and the laboratory analysis of off-site environmental samples. A pre-proposal conference for interested firms was scheduled for late April.

Safety meetings were held throughout the section during the month. There was one security violation, and one minor injury.

Six new suggestions were received during the month. Three suggestions were evaluated; one was adopted and two rejected. Mildred O. Wendland received a $15 award. There are eleven suggestions yet to be evaluated.

The three-hour training session on general use and interpretation of dose rate instruments was held for the AEC couriers. Interest was very high. As an outgrowth of this meeting, field training has been requested for these men by the AEC.

All four sessions of the sixth round in the Radiation Monitoring Refresher Course were held. Total attendance was 54, of whom eight were CPD employees. These sessions dealt with the interpretation of dose rate instruments. Discussion revealed significant differences in methods currently used and some concern over the lack of thorough coverage of the subject in our manuals.

The first four sessions of the Radiation Protection Training Course for Exempt Personnel were held. Average attendance has been about thirty. Participation in the discussions and general acceptance of the information has been excellent.

D. SIGNIFICANT REPORTS

HW-79652 - "Evaluation of Radiological Conditions in the Vicinity of Hanford, July-September, 1963", edited by R. H. Wilson.

HW-80877 - "Radioactive Contamination in Liquid Wastes Discharged to Ground at the Separations Facilities through December 1963", by R. H. Wilson.

HW-80892 1 - "Radiological Status of the Hanford Environs for January 1964", by R. F. Foster.

External Exposure Above Permissible Limits	February	1964
Whole Body Penetrating	0	0
Whole Body Skin	0	0
Extremity	0	0
Hanford Pocket Dosimeters		
Dosimeters Processed	3223	5776
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	8744	19,521
Results - 100-300 mrads	115	275
Results - 300-500 mrads	14	30
Results - Over 500 mrads	3	6
Lost Results	0	37
Average Dose per Film Packet - mrad (ow)	6.95	6.09
- mr (s)	29.51	24.12
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	1126	2881
Results - 50-100 mrem	24	24
Results - 100-300 mrem	14	15
Results - Over 300 mrem	4	4
Lost Results	2	5
Fast Neutron		
Filem Processed	166	696
Results - 50-100 mrem	5	27
Results - 100-300 mrem	24	100
Results - Over 300 mrem	0	2
Lost Results	0	3
Hand Checks		
Checks Taken - Alpha	31,618	69,846
- Beta-Gamma	63,533	117,693
Skin Contamination		
Plutonium	20	21
Fission Products	34	84
Uranium	0	1
Tritium	0	0
Thorium	0	0

Whole Body Counter

Subject	Number of Examinations 747-A WBC	1964	Mobile WBC	1964
GE Employees				
Regular	64	119*	107	325
Incident Cases	4	25	0	1
Terminations	5	8	0	0
New Hires	17	44	0	0
Special Studies*	7	12*	0	10
Non-Employees	1	1	0	0
Total	98	209	107	336

Bioassay

Analysis	Current Reporting Limit	Results Above Reporting Limit Feb.	1964	Samples Assayed Feb.	1964
Plutonium	2.2×10^{-8} µc/sample	67	120	630	793
Fission Prod.	3.1×10^{-5} µc/sample	0	0	0	0
Strontium	3.1×10^{-5} µc/sample	0	0	0	0
Tritium	5.0 µc/l	62	137	144	325
Uranium	0.14 ugm/l	0	0	220	346**
Special Studies		0	0	26	60

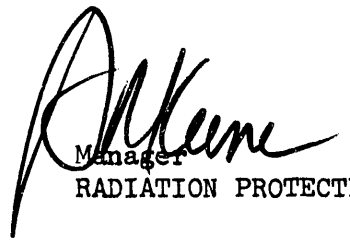
Calibrations

	Number of Units Calibrated February	1964
Portable Instruments		
CP Meter	1,034	2,082
Juno	235	510
GM	497	1,058
Other	225	465
Audits	104	201
	2,095	4,316
Personnel Meters		
Beta-Gamma Film	480	1,130
Rings	150	195
Other Film	164	458
	794	1,783
Miscellaneous Special Services	126	191
Total Number of Calibrations	3,015	6,290

* Five special studies reported last month as regular
**126 not reported last month

Environmental Monitoring

Samples	February	1964
Air		
Filters	343	735
Scrubbers	197	471
Water		
Raw	29	74
Sanitary	77	151
Process	20	45
Vegetation	60	160
Test Well	247	575
Fish	81	280
Waterfowl	0	77
Food Products	29	95
Beef Thyroids	32	63
Measurements		
Control Plots	32	70
Aerial Monitoring	2	2
Ionization Chambers	202	434



Manager
RADIATION PROTECTION

AR Keene:ald

FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

The operating cost control budget was adjusted in February to conform with Financial Plan No. 4 issued by RLOO-AEC. In summary, the authorized funds for the nonproduction programs assigned to Hanford Laboratories were adjusted as follows:

Program	Increase (Decrease)
04	$ 28 000
05	(53 000)
06	(252 000)
08	140 000
Total	$(137 000)

Additional funds were authorized to Hanford Laboratories by the Chemical Processing Department in February for sponsored 03 Program Research and Development. The increase in funds of $116,000 brings the FY 1964 authorization for this program to $791,000.

Additional equipment and miscellaneous capital work order funds were allocated to Hanford Laboratories. The $702,900 added equipment funds raised the current year's allocation to $2,142,000. The miscellaneous capital work order authorization was increased $20,000 for a total authorization of $196,000.

Work is proceeding on the Budget for FY 1966 and Revision of Budget for FY 1965. During the month, personnel requirements were received from the section managers, approved by the Manager-Hanford Laboratories, and forwarded to Contract Accounting. Requirements for special equipment (transportation, office, audio visual, etc.) were transmitted to the appropriate HAPO component having budget responsibility.

Special accounting codes were established during the month for the activities described below:

.3Y Radiation Testing Services for Lewis Research Center - NASA. $96,000 has been authorized for this phase of the work.

.8W C. A. Bennett or an alternate to conduct a course on Statistical Methods in Engineering and the Physical Sciences at Stanford University, June 22 through August 15, 1964. $1,000 will be billed as complete compensation in accordance with the terms established by contract SA-335.

.8X Fabrication of Irradiation Samples for Phillips Petroleum Company. Estimated fabrication cost is $16,000.

.8Z University of Washington Summer Institute - 1964. $500 has been authorized to conduct the summer institute in radiation biology for high school and college teachers between June 22 and August 14, 1964.

.2B J. L. Nelson's participation in discussions on the results of research in the field of mineral reactions for treating radioactive liquid waste on February 17 to 21, 1964, at the International Atomic Energy Agency headquarters in Vienna, Austria.

The following program codes were established during the month:

.23 Nuclear Facility Monitoring Guide (04 Program)
.40 Radiation Effects - General (06 Program)
.41 Nuclear Energy Weapons Effect Studies (06 Program)

General Accounting

Approval letter No. AT-316, AEC Monograph on Iodine 131 - L. K. Bustad, C. C. Gamertsfelder, and J. K. Soldat, was approved by the AEC on February 7, 1964. Approval letter No. AT-198 - Addendum No. 1, Miniature Swine for Hammersmith Hospital, was approved by the AEC on February 26, 1964.

The following new or revised OPGs were issued during the month of February 1964:

OPG No.	Title
1.1	Organization and Policy Guide System
2.1.1*	Transfer of Work to Other Contractors
2.3.10	Business Planning and Transfer Operation - Manager Position Guide
2.3.11*	Manager - Plant Operations Position Guide
22.1.2	Chemical Laboratory
22.3.1	Approval Authorizations
3.7.1	Political Activity
3.4.17	Reduction of Force - Monthly Salaried Employees

OPG No.	Title
3.4.19	Weekly Salaried Employee Appraisal Plan
3.7.2	Recorded Communications to Employees
44.7	Appendix A to HL OPG 44.7
7.15	Military Assignment of Personnel
8.5	Property Management of Protective Clothing and Equipment

*New OPGs

Hanford Laboratories' net material investment at February 1, 1964 totaled $25.5 million as detailed below:

		(In thousands)
SS Material		$ 23 886
Reactor and Other Special Materials		1 358
Spare Parts		355
Yttrium		26
Subtotal		25 625
Reserve: Spare Parts	$78	
Yttrium	26	(104)
Net Inventory Investment		$ 25 521

The cumulative value of nuclear material consumed in research by Hanford Laboratories during FY 1964 (at February 1, 1964) is as follows:

02 Program	$ 18 813
03 Program	160 041
04 Program	477 114
Total	$655 968

Savannah River Operation provided a new price list for heavy water as of January 1, 1964. The price per pound is now $19.82, including $13.55 fund and $6.25 nonfund, representing a reduction of $.16 per pound fund and an increase of $.04 per pound in the nonfund portion. Price allowance for scrap returned has also been reduced. This will result in an additional charge of $1,176 to Cost, based on the January 31, 1964 scrap inventory.

The heavy water inventory at the end of February 1964 showed a loss of 1,589 pounds valued at $21,817 for the PRTR. Heavy water scrap generated during the month amounted to 3,290 pounds, resulting in a $4,244 charge to operating cost. Heavy water accumulated at February 29, for return to SROO, amounted to 14,298 pounds valued at $177,856.

Laboratory Storage Pool activity is summarized as follows:

	Current Month		FY to Date	
	Quantity	Value	Quantity	Value
Beginning Balance	1 824	$1 365 697	1 480	$ 811 520
Items Received	121	74 912	1 359	1 021 317
Items Reclaimed by Custodians	(9)	(5 559)	(160)	(81 563)
Equipment Transfers	(32)	(11 797)	(270)	(109 187)
Items Disposed by PDR			(146)	(11 389)
Items Disposed by Excessing	(1)	(425)	(360)	(207 870)
Ending Balance	1 903	$1 422 828	1 903	$1 422 828-1)

(1- Includes 176 items valued at $122,384 on loan at February 29, 1964.

During the month, 154 items valued at $59,777 were loaned and/or transferred in lieu of purchases. A total of 820 items valued at $321,531 has been redirected to useful purposes this fiscal year. Operating cost for FY 1964 (at February 2, 1964) was $10,768.

Total value of equipment and material in custody of the Laboratory Storage Pool at February 29, 1964, was $2.4 million, including Reactor and Other Special Materials of $310,616; SS Material, $154,800; and Other Materials valued at $479,121.

The following contracts were processed during the month:

SA-333	Great Lakes Carbon Corporation
SA-334	Speer Carbon
SA-338	R. J. Donohue
CA-432	P. E. Church
MRO-71	Beckman Instruments

Personnel Accounting

Edna D. Britch was granted a disability pension on January 1, 1964, and Ralph K. Yeager retired effective March 1, 1964.

A patent award (HWIR 1368) was made to K. D. Coughren for a slip cam selector switch.

Personnel statistics follow:

Employee Changes	Total	Exempt	Nonexempt
Employees at beginning of month	1 808	798	1 010
Additions and transfers in	16	8	8
Removals and transfers out	18	6	12
Employees on payroll at end of month	1 806	800	1 006

Overtime Payments During Month	February	January
Exempt	$ 5 539	$ 9 828
Nonexempt	10 406	18 498
Total	$ 15 945	$ 28 326

Gross Payroll Paid During Month		
Exempt	$ 780 596	$ 780 954
Nonexempt	554 205	557 595
Total	$1 334 801	$1 338 549

Participation in Employee Benefit Plans at Month End	February Number	February Percent	January Number	January Percent
Pension	1 602	99.5	1 599	99.3
Insurance Plan - Personal	408		411	
- Dependent	1 394	99.9	1 392	99.9
U.S. Savings Bonds				
Stock Bonus Plan	154	38.2	158	39.0
Savings Plan	64	3.5	66	3.6
Savings and Security Plan	1 248	89.0	1 257	89.4
Good Neighbor Fund	1 298	71.9	1 299	71.7

Insurance Claims	Number	Amount	Number	Amount
Employee Benefits				
Life Insurance	-0-	$ -0-	-0-	$ -0-
Weekly Sickness & Accident	14	1 186	15	1 237
Comprehensive Medical	89	5 202	76	4 762
Dependent Benefits				
Comprehensive Medical	158	12 974	135	12 876
Total	261	$19 362	226	$18 875

GENERAL

The following personnel changes were made:

J. V. McMaster, Specialist-Measurements and Procedures, transferred to Programming Operation.
E. B. Hutchins, Specialist-Auditing and Economic Evaluations, transferred to Electric Utility Marketing Research Operation, New York City.
K. K. Leeser was transferred in from C&AO to replace Mr. Hutchins.

TECHNICAL ADMINISTRATION

Visitors Center activity:

February attendance	1 338
Average attendance per day open	53 1/2
Cumulative attendance since 6-13-62	62 657
Conducted groups	15 (totaling 466 people)

Plant Tour activity:

	Number	Total
General Public Relations Tour	6	231
Special Tours	3	6

Recruiting results for February follow:

Offers extended	42
Offers accepted	3
Offers rejected	12
Added to roll	7

Advanced Degree - Seven Ph.D. applicants visited HAPO for employment interviews. Two offers were extended; one acceptance and three rejections were received. Two offers are currently open.

BS/MS (Direct Placement) - No offers were extended. One acceptance and no rejections were received. There are no offers currently open.

BS/MS (Program) - Forty offers were extended. One offer was accepted and nine were rejected. Fifty-seven offers are currently open.

Technical Graduate Program - Seven Technical Graduates were placed on permanent assignment. Two new members were added to the roll and none terminated. Current Program members total 62.

Arrangements were completed for Columbia Basin College to offer Metals Engineering Institute courses for which employees may enroll under the Tuition Refund Plan. The first course will be offered this spring with follow-up

courses to be included in the 1964-65 curriculum. A story and photo on this topic were prepared for and included in the GE NEWS.

At month end, there were no open requisitions for nonexempt employees. Six nonexempt employees were hired during the month.

Suggestion plan activity included 33 submissions, 24 adoptions, and 23 rejections. At month end, 131 suggestions were in process of being evaluated.

News releases were distributed on promethium-147 recovery and on Hanford Laboratories' acquisition of a new analog computer. Regional pickup of the releases was good. In addition, the news was mentioned in nuclear publications.

The Hanford Laboratories' Classification Review Committee completed the survey of classification and related security practices in the Laboratories. A report was prepared and transmitted by the HAPO General Manager to the Richland Operations Office.

The Richland Operations Office has approved the 1964 membership list, and all current dues have been paid.

FACILITIES ENGINEERING

At month end, Facilities Engineering Operation was responsible for eight active projects having total authorized funds in the amount of $7,443,500. The total estimated cost of these projects is $10,650,000. Expenditures on them through January 31, 1964 were $2,051,000.

The following summarizes project activity in February:

Authorized projects at month end ---------- 8

New projects authorized ---------- 0

Projects completed ---------- 1

New projects submitted to the AEC ---------- 1
- CAH-131, Temporary Physics & Mathematics Center

New projects awaiting AEC approval ---------- 4
- CAH-114, Critical Mass Laboratory Addition
- CAH-126, Waste Transport System
- CAH-128, Heat Transfer Apparatus for Model Studies
- CAH-131, Temporary Physics & Mathematics Center

Project proposals being prepared ---------- 7
CAH-123, Laboratory Fire Protection System
Atmospheric Physics Building
Biology Laboratory Utilities
Geological and Hydrological Wells, FY 1964
308 Building Addition
309 Building Addition
327 Building Services Addition

The status of active projects follows:

CAH-916 - Fuels Recycle Pilot Plant - Construction is 40% complete compared to a scheduled 27%. Superstructure steel is up and concrete pours are being made above grade. It will be necessary to negotiate a cost for tank vault piping changes resulting from waste calcination design changes, because current estimates by the contractor and the Company are incompatible. Inadequate sleeve and plug tolerances may slow progress in the near future.

CAH-922 - Burst Test Facility for Irradiated Zirconium Tubes - The project was physically completed on February 20, 1964 with a minor exception. The exception consists of decontaminating, modifying, and installing the prototype burst test vessel.

CAH-962 - Low Level Radiochemistry Building - Design is complete. The scheduled date for opening construction bids is March 4, 1964.

CAH-977 - Facilities for Radioactive Inhalation Studies - Design is complete. The Commission has not issued a project proposal revision requesting authorization of construction funds. No progress was made during the month.

CAH-982 - Addition to Radionuclide Facilities - 141-C Building - The Architect-Engineer is revising the detailed design drawings to incorporate the latest comments.

CGH-999 - Plutonium Recycle Critical Facility Conversion to Light Water - A project proposal revision, requesting an extension of the directive completion date to April 30, 1964, was submitted to the Commission on February 12, 1964. The extension will permit changing the project scope to permit the handling of irradiated fuel in this facility.

Some difficulties were still encountered with the control and safety rod snubbers. Corrective action was taken.

CAH-100 - High Temperature Lattice Test Reactor - The Commission authorized Vitro Engineering Corporation to proceed with detailed design on January 31, 1964. Design of the building and services are to be completed as a separate package for early contracting of construction. The target date for completion of this part of the design is May 15, 1964. The building remains essentially as originally scoped.

Design of the reactor and auxiliary systems will be completed later. The Company is planning to recommend a change in the design criteria for the reactor electric heater system as a result of developmental work being performed by Hanford Laboratories.

CAH-114 - Critical Mass Laboratory Addition - The project proposal requesting authorization of design funds is still awaiting approval by Washington AEC.

CAH-116 - PRTR Decontamination and D_2O Cleanup - Vitro Engineering Corporation issued four information drawings. Title I design is about 80% complete. It was originally scheduled to be complete January 31, 1964.

Two used 200 Area stainless steel tanks will be modified for use in the decontamination system in lieu of procuring new tanks.

CAH-119 - PRTR Storage Basin and Experimental Facilities Modifications - Detailed design was started on January 27, 1964, and is scheduled to be completed by June 15, 1964. Detailed design is 6% complete compared to a scheduled 16%. The Company is preparing design criteria for the special underwater equipment. When this work is completed, a project proposal revision will be submitted requesting authorization of funds to procure and install the equipment.

CAH-123 - Laboratory Fire Protection System

The rewritten project proposal is being reviewed by Contract and Accounting prior to submission to the General Manager for approval.

CAH-126 - Waste Transport System - The project proposal was submitted to the Commission for authorization on January 29, 1964. It is scheduled for review the next time the Commission's Project Review Board meets.

CAH-128 - Heat Transfer Apparatus for Model Studies - 185-D Building - The project proposal was transmitted to the Commission on January 29, 1964. It is scheduled for review at the next review board meeting.

CAH-131 - Temporary Physics and Mathematics Center - The project proposal was transmitted to the Commission on February 11, 1964. It will be included on the agenda for the review board meeting.

Engineering Services

Engineering work was performed in support of design and construction on active projects, project proposals, preliminary planning and design criteria for new projects. Principal work items included: (1) field liaison, review of shop drawings and approval of submitted materials on CAH-916, FRPP: (2) preparation of design criteria for project proposal for Variable Spectrum Test Reactor (PPA); (3) scope design of modifications to critical facility CGH-999 for tests with irradiated fuels; (4) review of A-E preliminary design on CAH-100, HTLTR, and consulting engineering on heater element design; (5) study of 327 building addition; and (6) scope and criteria for 100-F Area utilities for Biology Laboratory.

Budget studies are being prepared for seven FY 1966 items: (1) Central Waste Disposal and Decontamination Facility; (2) Chemistry Laboratory Addition; (3) Central Maintenance Facility; (4) Physical Sciences Building; (5) Scientific Liaison Center; (6) Van de Graaff Accelerator; and (7) Biology Laboratory.

Scope design and design criteria were completed for the Containment Testing Facilities. Drawings have been submitted to Washington-AEC Division of Reactor Development for review. Major containment vessels will be released for J. A. Jones for purchase when AEC comments are received.

Engineering and consulting work was provided to research and development personnel on the following: (1) engineering assistance on experimental neutron spectrometer, 105-KE building; (2) engineering assistance on X-ray equipment, 329 building; (3) analysis of 325 building to suit requirements for additional glove box space; (4) engineering assistance on fast reactor concept study; (5) engineering assistance on critical mass experiments; (6) engineering assistance to CPD on instrumentation for fabrication prototype facility and consulting assistance on Project CAC-880, CAC-121 instrument design and dry-air system, 234-5; (7) development of engineering information for purchase of computer trailer; (8) engineering for equipment packaging for arctic Biology program; (9) engineering for installation of waste flow meters; and (10) engineering on liquid sodium loop.

Engineering service was provided on numerous maintenance and laboratory modification and improvement jobs. Major items were: (1) review of intercom system for 325 building; (2) engineering analysis and recommendation of modifications to electronics shop ceiling, 328 building; (3) engineering for improvement of 340 building waste ph monitoring equipment; (4) engineering for improvement of 3702 building ventilation system; (5) analysis of 325-A building emergency alarm systems; (6) modification of 144-F building ventilation; (7) engineering for pipe penetration manholes at PRTR; and (8) engineering of intercom system of 144-F and electrical service improvements, 146-F.

Pressure Systems

Specifications for the reactor simulator vessel (2500 psig) for the containment systems experiment were completed and reviewed with the third-party inspector.

The design of sample bombs used in 325 analytical chemistry were reviewed and a new design recommended.

The 328 building gas storage was reviewed and improvements suggested.

Analysis of the Laboratories process - sanitary water separation was completed.

Facilities Operation

The following table summarizes waste disposal operation:

Item	December	January
Concrete waste barrels disposed to 300-Wye burial ground	2	12
Concrete waste barrels disposed to 200-W plutonium burial ground	5	4
Loadluggers of dry waste disposed to 300-Wye burial ground from the 325 building	3	4
Loadluggers of dry waste disposed to 300-Wye burial ground from 300 area sites other than the 325 building	16	24
Loadluggers of dry waste disposed to 200-W plutonium burial ground from 300 area sites	7	10
Crib waste volume (gallons)	355,000	395,000

The vertical burial pipes at 300-N burial ground have been backfilled and capped with concrete. It appears that some additional earth backfill will be needed on some of the pipes to reduce the reading at ground level to < 1 mr/hr.

Minor contamination was spread at the Wye burial ground during backfilling of a vertical burial pipe. It was similar to the event in December. Designs for an improved system have been developed following the earlier incident and are being studied for future use.

None of the retention basins exceeded activity levels of 1.0×10^{-5} $\mu c\text{-}\beta/ml$ or 5.0×10^{-6} $\mu c\text{-}\alpha/ml$.

The inside surface of tanker #5473 was flushed with water by CPD personnel. The readings through the tanker were reduced by a factor of 2-3. Maximum reading through the tanker is now 1 r/hr. Tanker #5472 is being flushed now.

Alpha emitting material contaminated the floor during work in room 530-A, 325 building. The source of the material was determined to be a leaky container of waste. The area was cleaned without complication.

Oil use for #2 pump at the river pump house has been excessive the past six weeks. Pressure at oil tank has been reduced to 20# to help overcome this problem.

The second floor ventilation system of 308 building was balanced upon completion of the modifications of offices and labs. Also, the H&V alarm system was calibrated and tested.

The centrifugal switch on the cold supply fan at 327 building failed and was replaced.

Larger circuit breakers were installed on #1 and #2 supply units at 329 building to improve operation of new motors. Supply plenum supports had broken and new supports were installed.

Drafting

The equivalent of 142 drawings was produced during the month for an average of 30 man-hours per drawing.

Major jobs in progress are: (1) high temperature gas loop; (2) plutonium powder processing line; (3) fast super pressure power reactor concept; (4) inhalation studies hood as-builts; (5) rotating crystal spectrometer; (6) capacitor discharge test apparatus; (7) 309 building piping service drawings; (8) modifications to 309 chilled water system; (9) refraction compounds glove box; (10) PRTR corrosion and film loop; (11) floor plans, HAPO buildings; and (12) intermediate level waste cask.

Major nonproject jobs in progress during the month were: (1) modify building and laboratory furniture, 141-H; (2) construct swine farrowing barn and pens, 141-F; (3) modify heating and ventilating system, 144-F; (4) install hoods and exhaust ductwork, 1705-F; (5) construct pasture and sprinkler system at meteorological tower; (6) install platform, stairway and monorail, 308; (7) relocate small dynapak and install new dynapak, and renovate room 125, and install platform stairway and monorail, 308; (8) construct office addition and building services, 314; (9) construct thoria laboratory room 417, install emergency generator, modify room 520, construct high temperature labs and mezzanine offices, install floor tile in ceramics laboratory, enclose north entry to basement and enlarge men's restroom, 325; (10) install lighting, fire detectors and construct roof, 3718-A/B; (11) soundproof rupture loop annex, construct manhole and retention waste by-pass line, and install snubbers in high pressure lines, 309; and (12) modify E cell and install safety valves, 327.

Containment systems experiment facilities have been started in 222-T. The crane maintenance platform is completed and access ladders installed. Equipment is being removed by plant forces, and construction of the barrier wall will start the first week in March.

Fabrication work continues on the waste solidification program equipment for 324 building. Racks 2A and 1B were delivered to 321 building for installation and test.

Plans have been completed for initiation of the FY 1964 well drilling program. Major work involves improving existing wells, and a work order has been processed to J. A. Jones to construct 10 wells as a trial to determine methods.

for Manager
Finance and Administration

W Sale:KKL:whm

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output for February was 1224 MWD, for an experimental time efficiency of 77% and a plant efficiency of 60%. There were sixteen operating periods during the month, five of which were terminated manually and eleven were terminated by scrams. A summary of the fuel irradiation program as of February 29, 1964, follows:

	Al-Pu		UO_2		PuO_2-UO_2		Other		Program Totals	
	No.	MWD	No.	MWD	No.	MWD	No.	MWD	No.	MWD
In-Core	0		7	1199.3	78	10173.2			85	11372.5
Maximum				264.6		240.1				
Average				171.3		130.4				
In-Basin	43	3726.4	25	2754.9	27	1454.1			95	7935.4
Buried							1	7.3	1	7.3
Chem. Process.	32	2309.3	35	1965.8	—	—	—	—	67	4275.1
Prog. Totals	75	6035.7	67	5920.0	105	11627.3	1	7.3	248	23590.3

Note: (MWD/Element) X 20 ~ MWD/TU for UO_2 and PuO_2-UO_2

One fuel element failed during the month. A unique problem was encountered in the attempts to identify the tube containing the failed element, in that so much fission gas was released and entrained in the primary coolant that all previously successful detection means were unable to identify the offending tube due to the high fission gas background. A different method of detecting the failed element was devised, which was to remove the fuel element from the high background to the fueling vehicle where, after a short stagnation in air, the coolant air stream was monitored for fission gas. This method was successful.

The demineralized water system was placed in service and a prolonged surface flush of the basin was completed using approximately 180,000 gallons of demineralized water.

Equipment Experience

A total of 56 reactor outage hours were charged to repair work. Main items were:

Valves	25 hours
Helium Compressor	15 hours
Stuck nozzle caps	5 hours

Preventive maintenance required 271 hours or about 7% of the total maintenance effort.

A potted electrical penetration of the containment vessel was found to have deteriorated from the last inspection. The reactor was shut down and the potting compound replaced. This completes the potting replacement of those penetrations installed during the same time period and noted to be of a different hardness from the original installations.

One PRTR process tube jumper, chosen because of extensive service and repair record, was proof-tested to determine the burst pressure. Failure occurred in the straight section (rather than the flared end fitting or curved section) at 8,200 psig which is four times the corrected design pressure.

An improved portable helium leak detector has been fabricated and is in use. The new unit uses a small motor-driven gas pump rather than the vibrator driven type and eliminates the need for an auxiliary battery pack.

A purchase requisition has been issued for spare PRTR zirconium process tubes. Twenty spares are planned for an estimated $100,000.

Three PACE budget items were submitted as a result of an AEC request for FY 1964 Super General Plant items. These were: (1) PRTR Power Level Increase, (2) 309 Building Addition and Facilities Modification, and (3) PRTR In-Reactor Mechanical Property Measurement.

Improvement Work Status (significant items)

Work Completed

Supplemental Battery Charger
Electrical Containment Penetration Modifications

Work Partially Completed

In-line Gas Sampling
Process Tubes Level Indicator
Inlet Gas Seal Replacement
Shim Rod Shroud to Top Cap Modification
Improved RTD Connector Sealant and Bracing
Instrument Power Transfer System
Installation of New Alarm Annunciator
Thermistor Probe Installation in FEEF
Flow Monitor Tubing Snubber Installation
Indication of DC Solenoid Failure
Holdup Tank - High Level Alarm
DT-1 Storage Tank Sight Gauge
Battery Power for Galvanometer Light
Vibration Snubbers for Earthquake Protection

Design Work Completed

P-16 Pressure Interlock
Process Sewer Bypass Manhole #2A
Process Tube Examination Equipment Installation

Design Work Partially Completed

Additional Fuel Storage and Examination Facility
Decontamination Building and D_2O Cleanup Facility
Flux Wire Scanning System
Supplemental Emergency Water Addition
Permanent Installation of Closed Circuit TV
Rupture Monitoring System Modifications
Inspection Manholes for Containment Penetrations

Process Engineering and Reactor Physics

Analysis of the experiments to measure the temperature coefficients of reactivity in PRTR is continuing. An interesting sidelight to these measurements was an experiment in which the level of the reflector was increased in increments of 0.2 inches and the change in full-power critical level observed. By increasing the reflector level 1.65 inches, with the moderator and reflector at nearly the same height, the reactivity of the reactor was increased about 0.07 milli-k.

Tests of the top and bottom shield were completed to determine the contribution of thermal and radiation heat sources to the total heat load. The data have been reduced to equations for predicting the heat load of the shields at varying power levels and process tube temperatures.

Pressure drops and heat transfer coefficients for the moderator, reflector, and top and bottom shield heat exchangers have been experimentally determined for predicting capacity at higher reactor power levels. The first two were very close to the unfouled design point. The latter was near the fouled design point, indicating that some film buildup is present on the tube surface.

A study was completed on the types of tests needed to provide the required assurance of reliability of powering a primary pump with power from the 384 Building. The results showed that the mechanical integrity of the PRTR switchgear had already been demonstrated and that ample tests to prove the operability of the 384 emergency generator are being performed. Therefore, only personnel training tests are needed in the immediate future in addition to periodically placing a pump on 384 power to prove continued operability of the system.

Procedures

Revised Operating Procedures issued	1
Revised Operating Standards issued	10
Temporary Deviations to Operating Standards issued	8
Process Specifications accepted for use	3
Revised Maintenance Procedures issued	1

	February	Total
Drawing As-Built Status:		
Approved for As-Built	12	1 274
In Drafting	(8)	21
In Approval	2	3
Deleted or Voided	(1)	81
		1 379
Scheduled for Review		207
Total		1 586

Personnel Training:	Manhours
Qualification Subjects	189
Specifications, Standards, Procedures	111
Emergency Procedures	10
	310

Status of Qualified Personnel at Month-End:

Qualified Reactor Engineers	9
Qualified Lead Technicians	6
Qualified Technicians	19
Provisionally Qualified Technicians	3

Experimental Reactor Services

The status of the various test elements at the end of February 1964, is shown below. Those elements which had reached their assigned goal exposure or had been permanently discharged for other reasons prior to February 1, 1964, have been deleted from this table.

Test No.	Channel Location	Number	Description	Date Initial Charge	Date Discharged	Approximate Accumulated MWD
14	1956	5097	Moxtyl-Swaged	4/2/62	--	149.5 Repad
14	1352	5098	Moxtyl-Vipac	5/8/62	--	234.8 Repad
14	1758	5099	Moxtyl-Vipac	5/8/62	--	164.9 Repad
48	1156	5150	Moxtyl ($\frac{1}{2}$"x$\frac{1}{2}$" Pads)	8/1/62	--	160.0
54	1542	5116	Moxtyl (Clip-On Pads)	5/8/62	--	166.4
54	1554	5118	Moxtyl (Clip-On Pads)	5/8/62	--	240.1
61	1249	5185	Moxtyl-Physics	5/28/63	--	143.3
61	1354	5186	Moxtyl-Physics	5/28/63	--	132.8
61	1445	5192	Moxtyl-Physics	6/13/63	--	148.5
67	1152	5119	Moxtyl (Repaired Wire)	10/20/63	--	72.6
67	1457	5117	Moxtyl (Repaired Wire)	10/20/63	--	114.9
80	1544	5214	Mcxtyl (1% PuO_2, Swaged)	11/18/63	--	67.8
85	1847	5230	Moxtyl (1% PuO_2, Vipac)	1/30/64	--	19.3
37	1548	1098	UO_2-Physics	5/12/62	--	138.6
37	1550	1097	UO_2-Physics	5/12/62	--	145.9

Three fuel elements were examined in the basin during the month. Broken wire wraps on two fuel elements were repaired. A total of twelve rods from three fuel elements were shipped to Radiometallurgy for testing.

Plutonium Recycle Critical Facility

Startup work for the H_2O moderated PRCF continued through month-end. Estimates of fuel availability continued to dominate scheduling of experiments.

EBWR Core Modifications and PRCF Pump Interlock Changes were completed during the month.

Four Process Specifications were accepted for use.

Fuel Element Rupture Testing Facility

FERTF Test #8, Irradiation of an Intentionally Defected UO_2 Fuel Element, was started on 2-2-64, and continued through month-end. During the month, the fuel element was cycled through sixteen critical periods and gained about 8.3 MWD exposure. No increase in fission gas release was observed which might indicate propagation of the failure area.

Processing of Spent PRTR Fuels

Shipment of Al-Pu fuel elements to Redox for chemical processing was begun.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 17,098 hours. This includes 13,202 hours performed in Technical Shops, 2,971 hours assigned to J. A. Jones Company, 842 hours assigned to offsite vendors, and 83 hours to other project shops. Total shop backlog is 15,011 hours, of which 90% is required in the current month with the remaining hours distributed over a three-month period. Overtime work during the month totaled 125 hours or 0.7% of the total available hours.

Distribution of time was as follows:

	Man Hours	% of Total
N Reactor Department	2 731	16.0
Irradiation Processing Department	3 324	19.4
Chemical Processing Department	413	2.4
Hanford Laboratories	10 630	62.2
Hanford Utilities and Purchasing Department	0	0

LABORATORY MAINTENANCE OPERATION

Total productive time was 17,900 hours of 19,400 hours potentially available. Of the total productive time, 87.7% was expended in support of Hanford Laboratories components, with the remaining 12.3% directed toward providing service for other HAPO organizations. Craft overtime worked during the month was 1.0% of total available hours.

Manpower utilization (in hours) for February was as follows:

A.	Shop Work		1 500
B.	Maintenance		7 300
	1. Preventive Maintenance	2 000	
	2. Emergency or Unscheduled Maintenance	1 500	
	3. Normal Scheduled Maintenance	3 800	
C.	R&D Assistance		9 100

WD Richmond

Manager
Test Reactor and Auxiliaries

WD Richmond:bk

INVENTIONS OR DISCOVERIES

All persons engaged in work that might reasonably be expected to result in inventions or discoveries advise that, to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during the period covered by this report except as listed below. Such persons further advise that, for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

INVENTOR	TITLE OF INVENTION OR DISCOVERY
C. E. Fitch and G. L. Waldkoetter	A Method of Lateral Shift Measurement in Ultrasonic Critical Angle Tests (HWIR-1694)
J. D. McCormack	A Treatment for Charcoal to Improve Retention of Iodine Initially Adsorbed as Organic Iodide.
M. O. Rankin	Device for Recording Air Concentrations of Zinc Sulfide Fluroresent Pigment on a Real Time Scale (HWIR-1693)

H. M. Parker

Manager, Hanford Laboratories

END

DATE FILMED

4 / 29 / 93