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HANFORD LABORATORIES MONTHLY ACTIVITIES REPORT

MARCH 1964

APRIL 15, 1964

[redacted]

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

GENERAL ELECTRIC

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HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT
MARCH 1964

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Compiled by
Section Managers

April 15, 1964

[redacted]

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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PRELIMINARY REPORT

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

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Table I - Hanford Laboratories Force Report

Date: March 31, 1964

	At Beginning of Month		At Close of Month		
	Exempt	Salaried	Exempt	Salaried	Total
Chemical Laboratory	146	130	147	123	270
Reactor & Fuels Laboratory	209	197	210	195	405
Physics & Instruments Laboratory	104	75	109	81	190
Biology Laboratory	44	66	44	65	109
Applied Mathematics Operation	21	5	21	6	27
Radiation Protection Operation	44	97	44	94	138
Finance & Administration Operation	144	117	140	116	256
Programming Operation	21	3	21	5	26
Test Reactor & Auxiliaries Operation	63	311	63	309	372
General	4	5	4	5	9
TOTAL	800	1006	803	999	1802

BUDGET AND COST SUMMARY

March operating cost totaled $2, 501, 000, an increase of $21, 000 from the previous month; fiscal year-to-date costs are $24, 009, 000 or 72% of the $33, 304, 000 control budget. Hanford Laboratories' research and development costs for March 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	$ 105	$ 92	$ 755	$ 1 180	64
03	7	7	245	250	98
04	978	996	10 358	13 785	75
05	101	96	1 030	1 438	72
06	272	243	2 412	3 352	72
08	31	27	156	290	54
	1 494	1 461	14 956	20 295	74
Sponsored by					
NRD	149	135	1 433	1 798	80
IPD	18	25	397	490	81
CPD	146	205	1 255	1 784	70
	$1 807	$1 826	$18 041	$24 367	74%

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

	Increase
04 Program	$ 31 000
05 Program	35 000
08 Program	50 000
Total	$116 000

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

A prototype N-Reactor fuel-fouling detector prepared from Zircaloy-2 clad Th-U alloy rod has operated successfully in a pressurized loop since February 16. Central temperature of the rod is 450 C and has remained reasonably constant.

[redacted]

The irradiation examination of Al–2 wt% Li irradiated for 124 days in a pressurized loop showed a density change from 2.57 to 2.51 g/cm^3.

An N-Reactor single-tube, fluted fuel element has been irradiated to 1800 Mwd/ton at a maximum fuel temperature of 550 C. The measured volume increase is 0.9%, and the irradiation is continuing.

Electron microscopy of replicas from U–2 wt% Zr irradiated to 0.25 at.% burnup shows extensive grain boundary tearing in a band near the outer edge of the fuel and aligned porosity within the grains at the center of the rod. This fuel was irradiated as a Zircaloy-clad rod in a NaK capsule.

An N-Reactor outer fuel component irradiated to 2000 Mwd/ton has been induction heated over its full length to 980 C and water quenched. Inner and outer cladding of the component displayed dimples, probably associated with bond failure during the severe thermal cycle.

Fabrication of eight uranium alloys to coextruded rod stock has been undertaken. Alloys containing 800 ppm Si, 260 ppm Al + 300 Si, and 800 ppm Al cracked during beta heat treatment. All others were successfully fabricated. A U–1.7 wt% Nb–400 ppm Al required gamma phase temperatures for extrusion.

In the studies of brazing alloys the effect of iron and copper additions on the corrosion of the reference Be–Zircaloy-2 braze has been studied. Iron additions up to 1% decrease corrosion resistance, but further additions up to 12% increase corrosion resistance.

The T-section forged N-Reactor outer support being developed could not be alpha formed on the 600 kva welder. The effect of beta phase forming is being investigated to determine whether forming temperature affects support strength. In addition, supports have been formed from Zr–2 wt% Nb–1 wt% Sn alloy under the same conditions used for Zircaloy-2 fabrication. The Nb-Sn alloy should produce still higher strength supports.

Fifteen hot-headed N-Reactor outer tubes coated by resistance welding are undergoing evaluation. Five components autoclaved 36 hr in 400 C steam appear to be in excellent condition.

[redacted]

An irradiated N-Reactor inner fuel element was rupture tested in IRP under conditions where the water temperature was held at 300 C for 15 min after the rupture was detected. The rupture was considerably more severe than in previous tests where cool-down was started after 5 min.

Very little corrosion was found in crevices under N-Reactor fuel supports welded to a zirconium clad sample which was heated to give a heat flux of 600,000 Btu/hr-ft^2 and exposed for 19 days to ammoniated water at pH 10, 288 C. Severe attack had been encountered in previous tests with lithiated water.

The thermocouple element in KER-1 shows no measurable temperature rise due to crud deposition during the 26 days following decontamination of this carbon-steel loop. Good water quality with low crud content is confirmed by coolant analyses.

In an induced vibration test of two N-Reactor target elements in flowing 277 C water, fretting corrosion (1 to 5 mils deep) occurred on one external support where a steel shoe had come off.

Analysis was continued of laboratory data obtained in the heat transfer and fluid flow experiments with a full-scale, electrically-heated model of an N-Reactor fuel column and outlet fittings. Correlations were developed that would allow accurate extrapolation of laboratory data for calculation of pressure drop of a tube of reactor fuel elements.

Transient experiments investigating loss-of-coolant accidents for N-Reactor fuel tubes were started in the heat transfer laboratory. A full-scale, electrically heated model of a half-length column of N-Reactor fuel elements with all the associated inlet and outlet piping was subjected to a sudden opening of the inlet piping to atmospheric pressure. These first experiments were run to determine points of steam formation and to check on the system and equipment response; results were satisfactory.

An N-Reactor stack gas dew point of at least -31 C (-24 F) is necessary to inhibit the hydriding of the Zircaloy process tubes. This dew point is based on recent measurements of the diffusion rate of water vapor through an N-Reactor graphite tube block in the temperature range 350-650 C (662-1202 F). After 68 days, a Zircaloy sample covered with a 200 mg/dm^2 oxide film hydrided when exposed to a He-water vapor-H gas mixture; whereas, a Zircaloy sample coated with a 100 mg/dm^2 film did not hydride.

Graphite burnout monitors from KW- and D-Reactors showed negligible reaction reaction with reactor gas.

A mechanism has been offered that accounts for the half power dependence of the graphite-water vapor reaction rate on the partial pressure of water vapor.

Long term irradiation of N-Reactor graphite in GETR continues satisfactorily. Based on data from the GETR irradiations, a new relationship between GETR and N-Reactor exposures has been determined.

A Li^6 detector was used to measure fast neutron spectra at several positions in the N-Reactor during full core physics startup tests.

Exposures in excess of 5000 Mwd/ton have now been achieved by both vibrationally compacted and swage compacted UO_2-0.48 wt% PuO_2 in PRTR.

One vibrationally compacted, impacted UO_2-1 wt% PuO_2 fuel rod failed during full power operation because of incomplete outgassing of the fuel traceable to malfunction of a diffusion pump.

Preliminary out-of-reactor loop tests of wide pad PRTR end fixtures indicate an acceleration of wear of the process tube when coolant temperatures are increased from 250 to 305 C.

One UO_2 PRTR fuel element that had a broken rod wire wrap was repaired in the PRTR basin.

Short, prototypic EBWR fuel rods were discharged at exposures ranging between 1800 and 2300 Mwd/ton. Others will be irradiated to 5000-20,000 Mwd/ton.

A 70% rejection rate of EBWR fuel rod cladding has caused ultrasonic inspection to proceed more slowly than anticipated.

Test rods for in-reactor comparison studies of surface roughening effects on heat transfer rates were completed.

Initial postirradiation examination of ThO_2-PuO_2 sintered pellets revealed fission product relocation similar to that previously observed in other ceramic fuels.

Six PRTR fuel elements were fabricated. Two were vibrationally compacted, and four were swage compacted.

Results of hydrocarbon contaminated tests, using samples of oil-doped UO_2 fuel, indicate that the UO_2 vacuum outgassing temperature should be limited to 250 C maximum until more test data can be obtained.

A simplified process using "spiral machining" and "coin pressing" techniques was developed anu is now being used for the fabrication of Pu-Al wafers for Phoenix fuel tests. Ten kilograms of Al–20 wt% Pu (8 wt% Pu^{240}) alloy were cast into five cylinders from which 0.020 in. wafers will be machined.

Initial performance of the second generation shim rod assembly in the environment test facility has been far more satisfactory than the previously tested aluminum screw assembly. The zirconium lead screw-ball nut assembly has exhibited no tendency to stick, and motor current to drive the control elements is reduced.

Two vibration monitor pickups have been placed on the inlet jumper of PRTR Tube 1653 and are continuously measuring the vibration level. Comparison of the readings from the two transducers indicates that the transducer output decreases as much as 70% during the first few days of exposure to radiation environment. This reduction appears to be a permanent effect.

Out-of-reactor testing (under prototypical PRTR conditions with impressed vibration) of a 19-rod UO_2 fuel element with offset end brackets showed greater fretting attack than tests with aligned brackets.

Measured dissolution of PuO_2 powder in sulfamic acid decontaminating solution was 0.4 to 1.6 mg Pu/liter in 1 hr at 40 to 70 C.

Calculations were performed to determine the heat transfer characteristics of two designs of inverted cluster fuel elements that employ seven symmetrically placed coolant holes through the cross section and an annular flow passage between the fuel and coolant tube. It was calculated that one of the designs could operate at a tube power of 803 kw satisfactorily with maximum fuel temperatures of 1270 C.

Analysis of the prompt kinetic response of an EBWR type reactor with mixed oxide fuel and with a delay in uranium heating has been completed. The significant result obtained was that the energy release is relatively insensitive to the time constant (for heat transfer from PuO_2 to UO_2 with the fuel element) for values up to five times greater than the initial prompt period.

A charge of aluminum-clad fuel elements undergoing corrosion test in C-1 Loop at 260 C and neutral pH has reached the desired exposure and will be discharged at the next outage. The cladding temperature of a thermocouple element rose steadily during exposure, presumably due to crud deposition, and then on two occasions fell to near its initial temperature, presumably due to thermal shock on reactor shutdown and startup.

Corrosion weight gains for a variety of stainless steels in 650 C (1202 F), 300 psi deoxygenated steam for 28 days are four to six times higher than the weight gains of similar alloys at 550 C (1022 F).

Exposure of Zircaloy-2 samples in various states of heat treatment and cold work in the G-7 Loop at ETR [282 C (540 F)-2.1 x 10^{20} nvt]for 49 days showed substantially higher corrosion for material heat treated in

[redacted]

the alpha + beta region. Samples of cold rolled N-Reactor tubing (86% cold work) showed substantially less corrosion than the same material with less cold work.

Short term corrosion of Haynes 25 and Hastelloy X in CO_2, water vapor, and oxygen at 1200 C (2192 F) showed weight gains in the three gases increasing in the order listed for both alloys.

Microstructures of high burnup samples of PuO_2 showed greatly increased porosity compared to preirradiated specimens. Similar samples of PuN revealed very little change. X-ray diffraction analysis indicated severe structural damage for both samples.

The upper temperature limit of compatibility between thorium metal and beta-Pu_2O_3 powders was determined to be less than 500 C.

A steel-clad nichrome–35.5 wt% PuO_2 cermet fuel element in the shape of a 2 in. ID x 2 1/2 in. long hollow cylinder also was fabricated by impaction.

Four tungsten-clad UO_2 specimens were successfully irradiated to 5×10^{19} fissions/cc (1600 Mwd/ton) at cladding temperatures greater than 2700 C.

Vibrationally compacted UN capsules were fabricated for irradiation tests.

The maximum temperature of 480 C was measured in a vibrationally compacted ThO_2 element irradiated in the GEH-4 facility. No internal aluminum cladding corrosion was observed.

UO_2 pellets were melted by direct induction heating.

The microhardness of UN is a function of crystal orientation, with fourfold symmetry on the (100) plane and Knoop hardness numbers between 440 and 600 (100 g).

Diametral-compression tests were completed with W–UO_2 cermets containing 1, 2, 5, 10, and 50 vol% UO_2.

Thin wall, stainless steel clad, 15 vol% UO_2-stainless steel cermet core fuel pins were successfully fabricated by high energy rate pneumatic impaction, with subsequent swaging to uniform size.

The welding capability of the magnetic force welder has been improved and crack-free closures of CEA (France) beryllium fuel cladding have been obtained.

Project CAH-922, Irradiated Burst Test Facility, has been completed and closed out by Facilities Engineering with the exception of completing the modifications and installation of the prototype unit. It is planned to complete the first burst test in this facility during March 1964.

Although design of the ATR model gas loop has been completed, the design criteria had not been documented. During March 1964 "Design Criteria for the Model High Temperature Gas Loop," HW-81312, was written to formalize the model loop design and to comply with HW-79197, entitled "Manual for Acquiring, Operating and Maintaining Pressure Systems."

Late in February 1964, limited funds were released for installation of the electrical, air and water services, and for fabrication of the test assembly. These segments of model loop construction will be complete by the end of March 1964.

Irradiation Capsules 30 and 31, which contain enriched uranium specimens, were previously processed through final bench testing and were found to have inadequate heat transfer characteristics. The heat transfer characterisitics of these capsules are being altered and will be reevaluated after modification.

Irradiation Capsule 32, which contains high purity uranium specimens of tubular and rod geometry of varying diameter and section size, was subjected to a final bench test preliminary to irradiation. Final external connections were made and the capsule was crated and shipped to the reactor to await charging.

[redacted]

Further work is being done on the metallographic evaluation of the distribution of intermetallic particles in dilute U-Fe-Si alloys.

Two in-reactor creep tests have been started on annealed AISI 304 SS. Test temperatures are 500 C and 550 C (932 F and 1022 F).

Irradiations of Inconel X-750 in two different conditions of heat treatment have been completed. Specimens with a third treatment are currently being irradiated. Hastelloy X-280 tensile specimens irradiated after aging at various temperatures have been tested at room temperature. A furnace and specimen holder are being prepared for 700 C (1292 F) tensile and stress-to-rupture testing of both these alloys.

Tests were performed to determine the effect of irradiation on the fracture behavior in notch bending of annealed Zircaloy-2. Twelve notch bend specimens were tested after irradiation to an average exposure of 2.8×10^{20} nvt. Irradiation caused only a slight embrittlement. It was observed, however, that the effect of increasing loading rate (strain rate) on the ductility was more adverse in the irradiated condition.

The long term irradiation of EGCR graphite continues satisfactorily. Fine cracks have been observed in samples recently recovered.

Irradiation of graphite cloth at 600 to 650 C resulted in very little change in mechanical properties. The increase in breaking strength doubled.

An analytical expression has been developed that relates the relative contraction of lampblack-containing graphites with integrated exposure to neutrons of energy greater than 0.18 Mev.

The first phase of the long term irradiation of boronated graphite was completed successfully. In general, the samples showed larger dimensional changes than did graphite not containing boron. Twenty-two samples were returned to the reactor facility for the second phase irradiation.

All arrangements were completed for the irradiation of boronated graphite in the ETR. The first charging of samples will occur during the April outage.

Calculations have shown that a 20-fold increase in the fast thermal neutron ratio in a Hanford reactor can be obtained by irradiating samples inside a neutron converter fuel element.

Three tubular Zr-clad Th−2.5 wt% U fuel elements have been irradiated to 5450 Mwd/ton at maximum temperatures up to 580 C and have incurred a maximum volume increase of 0.8%. The irradiation is continuing.

Defect tests of Zr-clad Th−2.5 wt% U in 1 wt% Zr rods, as extruded and after varying heat treatments, have shown that separation of the clad from the fuel during corrosion occurs by cracking of the thorium fuel rather than by bond failure.

The Th-U-Zr alloys of varying composition have been double vacuum arc melted and coextruded for form Zircaloy-2 clad rod stock. Defect tests will determine the variation of the corrosion resistance of the fuel with these changes in composition.

Zircaloy-2 clad rods containing a submicron dispersion of uranium carbide are being irradiated in NaK capsules. These samples have accumulated burnup of 0.15 wt% and will be discharged at 0.3 at.%.

Development work in tube and wire fabrication led to the filing of two invention reports covering unique fabrication equipment.

Analysis of the two Segmented Fast Reactor preliminary core designs shows that fuel clad temperature exceeds 1400 F. This appears to be a reasonable upper limit for the use of stainless steel. Possible variations in core design parameters and reactor coolant flow and temperatures are being studied as a means of reducing clad temperature.

Sections cut from single crystals of irradiated molybdenum representing two levels of carbon as the impurity show a difference in the extent of deformation channeling, as revealed by transmission electron microscopy. Channeling occurs to a far greater extent in the low carbon crystal.

Precision lattice parameter measurements on molybdenum single crystal specimens have been completed, and the specimens will now be irradiated. Annealing the crystals for 16 hr at 1300 C (2372 F) resulted in a microstructure free of substructure.

Calculations indicate that the average vacancy makes 2×10^7 jumps during quenching of thin nickel foils. The probabilities of the vacancy encountering (1) other vacancies, (2) impurity atoms, and (3) dislocations are all quite high. The defect structure in quenched nickel consists primarily of vacancy clusters with the number of vacancy-impurity atom clusters increasing with increasing impurity content.

Experimental studies of the creep characteristics and physical damage of plutonium were continued to verify previous data and supply additional data. Expressions relating the steady state creep rate with temperature and applied stress have been derived for the beta, gamma, and delta phases.

Some success has been achieved in maintaining a relatively low sample temperature during cathodic etching in prototype equipment. This development is particularly significant since a temperature rise has been the major disadvantage in applying cathodic etching to plutonium.

A total of 600 EBWR rods have been fabricated for physics tests.

Continued investigation of the thermal cycling behavior of UO_2-W cermets has led to a suggested mechanism responsible for the high UO_2 loss.

Tungsten clad, 80 vol% $W-UO_2$ fuel plates were successfully fabricated for irradiation tests and for determining effects of ultrafine UO_2 and W grain size on thermal cycling behavior.

A demonstration tungsten honeycomb has been successfully coextruded. Hexagonal molybdenum rods coated with 0.025 in. of tungsten by WF_6 decomposition were assembled in a 3 in. OD molybdenum can. Coextrusion produced a 37-cell honeycomb with a 15-mil web. Some distortion of cell shape was caused by the use of wrought molybdenum rods. Powder molybdenum components are under development.

A $W-UO_2$ cermet containing uniformly dispersed UO_2 particles in the submicron size range has been produced and compacted to 92-95% density. Annealing of the compact at 1500 C for 4 hr caused some increase in UO_2 particle size, and annealing for 20 min at 2000 C caused further particle size increases. After heating, the UO_2 primarily occurs at grain boundaries and was still uniformly dispersed throughout the tungsten matrix.

Approximately 1500 lb of ThO_2 were densified by pneumatic impaction. Fabrication of the last 2 tons of thoria fuel elements (approximately 1200 elements) was completed.

Five gold alloy irradiation specimens containing up to 25% gold, were cast and sampled.

2. Physics and Instruments

Work on the determination of N-Reactor lattice physics parameters is continuing. The Lu, Eu, Pu-Al, and U^{235}-Al pins used in the cold N-Reactor pile test have been exposed in the PCTR and counting data from the pins is being analyzed. The elements for the hot test have been loaded into the N-Reactor.

Separate analog simulations of N-Reactor's primary and secondary coolant systems have been placed in operation on the new computer. A combined simulation that ties the two systems together for study of the complete coolant system is now being programmed.

Results of earlier simulation studies have proven useful in placing on automatic operation the actual primary coolant injection system master controller, five primary pump seal pressure controllers, and three primary coolant flow controllers. Initial calibration settings of these controllers were determined by the analog simulation studies.

Required instrumentation systems are in readiness for the next series of N-Reactor physics hot tests. Development work continued on several solid state circuits for use in the low level neutron monitoring system that will be used during the planned initial operational tests at N-Reactor. Encouraging circuit test results have been obtained. General performance and calibration tests were nearly completed on the 24 gamma energy spectrometers to be used in the N-Reactor fuel rupture monitor.

Eddy current retesting of selected N-Reactor heat exchanger tubes has revealed no indications suggesting that defects detected earlier last summer have become enlarged.

Inspection of tubing removed from the steam generators and subjected to decompased vapor phase inhibitor at the laboratories of Combustion Engineering has revealed intergranular corrosion penetrating the tube wall. This tubing has been inspected several times with the Hanford eddy current testers both at Hanford and at Connecticut. All results have been confirmed by destructive tests.

Ultrasonic techniques are being applied to uranium billets to characterize the variations in the clad thickness of coextruded N-Reactor fuel before extrusion. The technique uses the lateral shift phenomena that responds to metallurgical condition (orientation and grain size) of the uranium billet. A boundary wave phenomena indicates forging memory effects in the Zircaloy sleeve.

Fabrication of the X-ray fluorescence test equipment is complete and the tester is being installed in 333 Building. Preliminary tests have indicated that the threshold of detection is 1.0 wt% uranium contamination in the fuel element closure area.

A prototype tester has been developed and delivered to IPD for evaluation. This tester measures the penetration of Al-Si braze into the aluminum jacket of Al-Si fuels. Some interference in the test has been caused by alignment variations in the uranium cores. Quantity testing of production fuels is now being performed.

An ultrasonic test is being developed that detects internal bond defects in I&E Al-Si fuels with a test applied from the outside fuel surface. Apparent advantages of the system include elimination of the complicated internal probe required by the existing method, easier probe alignment, and increased equipment reliability. Difficulties caused by the high attenuation of ultrasound in the uranium fuel core, and thickness variations of the outer and inner clad appear to have been solved by careful selection of test frequency, beam focusing, and electronic gating techniques.

Zirconium hydride is being detected in N-Reactor process tubes by an ultrasonic technique. Defective metal is detected because the longitudinal wave velocity in hydrided regions differs from its value in normal metal, while shear wave velocity remains relatively constant for both types of material. This effect, which does not appear to have been previously recognized, may be of general usefulness for a wide range of new applications.

An interim status report has been issued outlining the nondestructive testing facility's ability to inspect waste storage tanks in the Waste Solidification Engineering Development Program.

Promising experimental results were obtained with the gamma spectrometry method of determining the cooling "age" of irradiated reactor fuel elements by measurement of the I^{131} content. Tests were conducted with fuel of various ages, up to 150 days.

Development of a Pu^{239} liquid sample counting system moved a step forward with completion of a shield and a collimator used in testing various multiplier phototubes and scintilators.

The initial series of criticality experiments with PuO_2-polystyrene compacts and the Remote Split-Table Machine was completed. During the course of these experiments, which included 75 different assemblies, critical dimensions and masses were determined for bare and reflected rectangular parallelopipeds, having geometries ranging from near cubes, to long columns, to thin slabs. The plutonium concentration in the plastic fuel cubes was 1.12 g Pu/cc (2.2% Pu^{240}), with an H:Pu atomic ratio of 15.

Planned critical experiments with Pu solutions are now under way while new fuel cubes with higher Pu^{240} contents are being prepared for further experiments with Pu solids and the Remote Split-Table Machine.

An interesting pulsed neutron source experiment was performed with a small tank (12 x 12 x 8 in.) filled with 3% enriched uranyl nitrate solution. The effective reproduction factor (k_{eff}), as determined from the measured decay constant of the prompt neutrons, was ~ 0.2. It remains to be established whether pulsed neutron source experiments can provide meaningful values of k_{eff} on units that are so far subcritical; however, the current data imply that it may be possible to obtain qualitative measurements even on such low k_{eff} systems.

The IAEA has expressed an interest in the GE Class I shipping container that is currently in the design stage at Hanford. It has been proposed to include the design as an example in the forthcoming IAEA regulations covering the transporting of fissile materials. Required information on the shipping container has been submitted to AEC-RLOO for transmittal.

Most of the internal discrepancies in the data obtained on the neutron scattering law for 95 C H_2O have been identified and eliminated or greatly reduced. Preliminary work has been done on the determination of the scattering law for polystyrene.

Development of components for slow-neutron inelastic scattering by time-of-flight continued.

Considerable work was done on additional programs and refinements to programs for handling Mev neutron total cross section data and results.

A plutonium sample was received from the NBS for isotopic abundance determination as part of the program of certification of this sample as a NBS-Pu Isotopic Standards.

A critical assembly of PuO_2-UO_2 fuels in H_2O was attained for the first time, to our knowledge, in the free world on March 27, 1964. The PRCF went critical with 496 EBWR demonstration fuel rods in good agreement with predictions.

Subcritical measurements with EBWR fuel rods have also been completed for three of a total of six lattice spacings to be studied.

Considerable reactor physics code development work was performed this past month. Modifications were made in the RBU input code, BARNS-II, and PHYSICS CHAIN. The burnup code ZODIAC was extensively revised, greatly increasing its usefulness.

Theory-experiment correlation studies were actively pursued. A series of plutonium aqueous homogeneous, critical experiments were analyzed to provide a test of basic thermal cross section data of Pu^{239}. Uranium and plutonium solution experiments were also analyzed, using various thermalization models. For all cases considered, except for very dilute uranium systems, a better match of the Nelkin spectra was obtained using the Wigner-Wilkins approximation than by using the Wilkins model.

Work is progressing on the PRTR Gamma Scan Facility. To aid in positioning, the head design has been modified.

The Phoenix reactor studies have been extended to include plutonium-boron systems. Calculations for Phoenix cores using B^{10} burnable poison have been performed for a variety of boron loadings and plutonium composites. The early results are encouraging.

A series of nuclear design calculations for a Fast Fuel Test Reactor have been completed and an invention report on a special reactor configuration has been submitted. The calculations emphasized the physics statics aspect of the proposed design.

Useful information has been obtained from a 75 hr test run at 1200 C nitrogen temperature in connection with the HTLTR program. A 1000 hr test at 1200 C has been started.

Detailed planning continued on improvements to be incorporated in the PRTR fuel element rupture monitoring system and the PRTR air exhaust monitor. In addition, a solid state relay control monitoring instrument was developed and applied to detect and indicate sequential circuit operation. With successful operation of the device, replacement of a large amount of wiring will not be necessary.

Offsite fabrication of two in-core neutron flux detecting chambers for uranium-isotope regenerating reactor use moved forward, and information was received that U^{234} required for coatiing the chambers can be obtained. Design work progressed on an improved, more rigid B^{11} beta current detector assembly to eliminate vibrational damage effects. One complete microwave method assembly, to be used for evaluating the effects of neutron flux on the resonance of a tuned cavity, was prepared and moved to KW-Reactor for testing. The klystron, power supply, and all control circuits performed correctly in the laboratory tests.

Wiring of the laboratory model of the multiparameter eddy current tube tester was completed and evaluation tests were started. Initial tests indicate that the various circuits of the equipment are satisfactorily performing their individual electrical functions. A study is being made to determine what improvements can be made in the Model 1004 tubing tester to increase its range of application.

In research on advanced ultrasonic methods for nondestructive testing, the frequency components in four ultrasonic pulses of different shapes were analyzed to determine the correlation between pulse shape and frequency spectrum as effects of diffraction and attenuation. The development of a new gating circuit permitted further experimental studies on the propagation of broadband pulses. These studies verified that the edge of the beam is composed predominantly of high frequencies, whereas the beam center is dominated by low frequencies.

The coupling of the rotating Zircaloy hydride detector eddy current probe to the stationary supporting equipment was accomplished by the fabrication of a transformer with rotating secondary and fixed primary windings. Temperature compensation will be accomplished by incorporating an infrared detector in the probe head.

The new ultrasonic thin-wall tubing tester was put in operation and used to test sheath tubing loaded with ceramic fuel. To eliminate noise signals from the fuel particles, a gating arrangement was developed to exclude signals arising from large fuel particles in contact with the inside surface tube.

Favorable results were obtained in laboratory tests of miniature G.M. tubes to be used for measuring I^{131} uptake in thyroids. The data will be telemetered as one channel of the animal telemetry system which also telemeters temperature, respiration rate, and blood pressure information. Progress was achieved on a smoke inhalation control system for animal experiments, and assembly was started on the alpha energy analysis air monitor that will be used in radionuclide inhalation experiments.

All engineering work was completed and satisfactory field tests made on the wind component analyzer that will be used in atmospheric physics studies. In addition, considerable progress was made regarding sensitive scintillation methods of detecting low energy betas with 3% detection efficiency obtained for Pm^{147}.

General acceptance of improved methods for estimating the dispersion of radioactive materials released to the atmosphere was furthered through

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presentation of these results at a meeting arranged by the USAEC, Division of Licensing and Regulation. The methods developed through Hanford research permit more complete use of on-site meteorological data for improved prediction of the lateral growth of the diffusing plume so as to include the effect of protracted releases as postulated in some reactor accident situations. In attendance at the meeting were representatives of the Advisory Committee on Reactor Safeguards, the Division of Licensing and Regulation, the Division of Production, the Division of Reactor Development, and the U.S. Weather Bureau.

Gamma ray counting of urine samples continued to appear as an acceptable alternative to whole body counting for measuring Cs^{137} in large populations. New urine samples from Alaska showed that Cs^{137} burdens continue to increase there. Richland residents showed nearly the same correlation between Cs^{137} in the body and in urine as the Eskimos.

The plutonium X-ray counter is a very valuable device for studying Biology Laboratory dogs who contain inhaled plutonium. Use of this counter will be exploited while its development is continued.

The studies of a moderated Li^6I scintillation counter that has been used elsewhere for rough work in neutron spectrometry show that its calibration is probably incorrect. Development of a proportional counter spectrometer is proceeding satisfactorily.

The pulsed X-ray machine developed by Linfield Research Institute was installed and tested.

The shield for the University of Washington neutron facility for medical research has been completed and testing and dosimetry measurements have begun.

3. Chemistry

In studies relating to potential new products, several extractant systems were scouted in the hot cells for possible use in $Th-U^{233}$ separations. Promising behavior was shown by three phosphorus compound solvent systems.

Studies of the selective leaching of uranium and protactinium from thoria have shown little promise, even with high-surface-area, spray-calcined thoria.

Dissolution studies of unirradiated thoria indicate satisfactory dissolution in 6 to 8 hr.

Another test was made of the use of cerium metal as a scavenging agent for the removal of polonium from bismuth, using full-level Po^{210} in irradiated bismuth. Although the scavenging was less complete than in a preliminary tracer-level run, presumably because of a shorter time cycle, the results were promising.

Preliminary experiments on the solvent extraction of polonium from nitric acid solutions gave Po^{210}-bismuth separation factors of about 30.

The first phase (removal of Zr-Nb activity on silica gel) of an operation to purify and isolate ca. 1 kg of Tc^{99} was completed successfully.

Samples of a synthetic zeolite were irradiated to 3.16×10^{18} nvt as a preliminary step in ascertaining the usefulness of synthetic zeolites for "hot atom" or other reactions involving neutron irradiation. No changes in the crystal structure were detected.

In studies relating to fuel reprocessing for plutonium recycle, sections of an unirradiated PRTR rod incrementally loaded with PuO_2 and UO_2 were satisfactorily declad and dissolved by procedures applicable to Redox plant processing. A test was made of the effect of normal irradiated fuel (10,000 Mwd/ton) levels of fission product molybdenum and zirconium upon the chloride contamination of UO_2 electrodeposited out of molten chloride salts. The added molybdenum and zirconium caused no increase in chloride content.

In studies of the recovery of uranium from 3% UO_2-97% ThO_2 fuels by molten salt processing, deposits of UO_2-ThO_2 solid solutions have been obtained ranging in Th:U ratio from 1.8 to 0.64.

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Spectrophotometric studies in molten LiCl-KCl have shown the existence of a uranium species not previously seen in molten chloride salt solutions, a species presumed to contain singly-oxygenated uranium(V).

In work related to fixation of radioactive wastes, ruthenium decontamination of Purex acid condensate by electrodeionization was observed to be much more effective than ion exchange alone. Decontamination factors exceeded 25 after passage of 875 column volumes with electrodeionization and were only 2.5 at 25 column volumes with ion exchange alone.

A 14 in. diameter mockup of the Waste Solidification Engineering Prototype spray calciner exhibited suitable performance characteristics.

An ultrasonic atomizing nozzle for spray calciner application showed very high capacity characteristics with water, but plugged with simulated Redox waste.

Several samples of phosphate glasses, doped with Ce^{144} to simulate the radioactivity level that will exist in wastes from power reactor fuel processing, were prepared in the hot cells. Stability of the samples will be observed over a period of time.

The BNL equipment for the hot-cell testing of the BNL continuous phosphate glass process was assembled during the month. Several shakedown runs were made, revealing the need for a number of equipment modifications.

In studies relating to dispersion of radionuclides in the environment, the program for calculating transient unsaturated, horizontal (one-dimensional) flow in soils was completed. Expansion into one program that will analyze vertical flow is underway.

Only minor changes are evident in the areal extent and concentrations of ground water radiocontaminants over the past several months.

Studies of the distribution of fallout-originated Na^{22} have shown its concentrations in urine much higher than in meat, relative to the Cs^{137} concentrations, indicating that meat is not the only source of biological Na^{22} uptake.

Analyses of uranium and thorium content of over 50 rat lungs from animals exposed to uranium ore dust have indicated a significantly greater retention of thorium than of uranium.

Continuing tests of tracers for atmospheric dispersion tests have shown rhodamine B and fluorescein to have suitable fluorescence emission properties to be useful in the dual tracer tests of the Atmospheric Physics Operation.

Radionuclide analysis of sized fractions of Columbia River bottom sediments indicates that the fine clay particles (of less than 5 μ diameter) contain a much higher radionuclide concentration than the coarser fractions.

In studies relating to reactor safety, an irradiated 24 in. NPR outer fuel assembly retained its integrity (although it blistered) when it was heated to 1800 F at a programmed rate and then quenched with 150 F water.

Bid packages for the long-delivery pressure vessels for the Containment Systems Experiment were completed. Experimental test program planning and initial development efforts are underway.

In studies relating to radioisotope recovery and handling, computer codes have been written to (a) greatly facilitate the computation of the quantity and activity of nuclides in decay chains up to 10 nuclides long and with up to 30 different time steps; and (b) allow the computation of bremsstrahlung radiation.

A sample of Penberthy shielding glass (typical of 325A and 324 windows) exhibited an electrical discharge when gamma-irradiated to doses that might be encountered in both facilities.

4. Biology

A new test of K-Reactor effluent is to be started in April with chinook salmon. About 5000 chinook fry were transported from the new Washington State Department of Fisheries salmon hatchery at Priest Rapids. Special attention will be paid to possible interaction of temperature and chemicals on the response of fish.

"Powdex" anion and cation exchange resins were tested for their toxicity on chinook salmon fingerlings. The amounts of these materials that may get into the river are unknown.

Although there is some evidence for an immune response of fish to infection by columnaris, we have just shown they can be reinfected after recovery from the disease. Further, the incidence of the disease is temperature dependent, being low at times of low water temperature and high when the water is warmer. However, deaths of fish exposed to warm water seemed to be due more to furunculosis than to columnaris, in spite of the obvious presence of the latter.

The GI tract appears to be the critical organ in fish chronically exposed to Zn^{65}. Although, based on levels of Zn^{65} found in river fish, this is of no consequence to fish in the Columbia River. These findings, coupled with our fundamental work on the mechanism of GI damage in rats, promise to open new areas of fundamental investigation.

A 3 yr old miniature pig, fed 125 μc Sr^{90}/day for about 2 1/2 yr, was killed when its condition appeared to be deteriorating. The animal contained 1200 μc or about 600 times the maximum permissible body burden for industrial workers. It appears that the animal had lymphoma. This is the first time we have seen an effect of Sr^{90} in pigs at this level.

To assist the Radiological Chemistry Operation in standardizing their procedure for detecting Sr in hair, we have administered Sr^{85} to a miniature swine. Materials for analyses will be sent to Radiological Chemistry.

Intact or splenectomized rats showed the same reaction to acutely toxic doses of either Pu^{239} or Pu^{238}. This seems to eliminate the spleen as a highly significant factor in explaining the differences in toxicity of these two isotopes.

By using DTPA-perfused gut segments, we have found that a minimum concentration of this chelating agent must be maintained before increased biliary secretion of plutonium will occur. This may, in part, explain the usual unsatisfactory results obtained from oral therapy in humans, where the dose levels have always been low.

A dog killed 1 1/2 yr after deposition of 90 μc $Ce^{144}O_2$ in the lungs, showed only slight to moderate fibrosis in the lungs. The changes were considerably less than those seen after the deposition of similar quantities of $Pu^{239}O_2$. This may be the first indication of the derivation of an RBE for alpha emitters in a practical experiment.

Due to crowded conditions in the dog colony, one long-term experimental dog that contained plutonium was accidentally bred. Further, since 60 puppies are expected by July 1, 1964, in our breeding program, temporary dog runs are being constructed on concrete pads.

The enhanced effect of bile salts in producing diarrhea following irradiation of the intestinal tract may be explained in terms of the effect of the resorption of bile salts. By using C^{14}-labeled bile salts, it was shown that the resorption of bile salts was inhibited by prior irradiation of the tract, thus, enhancing diarrhea resulting from bile salt irritation.

5. <u>Programming</u>

The economics codes used in Hanford fuel cycle analysis studies have been generalized to permit more rigorous treatment of private (vs AEC) accounting systems.

The Hanford survey physics codes have been improved to account better for self-shielding of the important plutonium resonances.

The theory and experimental bases for applying variability in wind speed and direction to reactor accident analyses were reviewed in detail with the AEC, and in summary form with a subcommittee of the ACRS. Discussions indicated no basic objections to use of the new evaluation method.

TECHNICAL AND OTHER SERVICES

There were four new plutonium cases confirmed by special bioassay analysis during the month. Three cases were estimated to be less than 10% of the maximum permissible body burden (MPBB, plutonium with bone as reference, is 0.04 μc), and the fourth case was estimated as less than 20% of the MPBB. All four cases were detected by routine bioassay sampling. The total number of individuals who have received internal plutonium depositions at Hanford is 342. With the termination of three employees having confirmed plutonium deposition, there are currently 246 employed.

An HL millwright received a localized exposure, in excess of the 13 wk operational control, to his right thumb while working in the storage basin at the PRTR (309 Building) on March 11. The exposure was incurred during the removal of rollers from a fuel inspection tray when a small piece of irradiated zirconium wire penetrated the employee's glove and was in contact with the thumb for an estimated 3 min. The estimated dose to the employee's thumb was 57 rems including 42 r.

Results of tritium analyses of river water were above detection limits for the first time. The concentrations of 4 pc/ml and 3 pc/ml measured at the 300 Area in January and February samples, respectively, indicate a rate of transport of about 600 curies/day. Analysis of samples taken upriver from Hanford is being started to confirm that the H^3 is from fallout. Information from others indicates that tritium in rainfall from weapons tests is at an all time high and still climbing. No significant exposure is implied from these concentrations.

The development of an experimental layout and procedure for a proposed in-reactor fuel corrosion test was completed.

Third and fourth degree polynomials were fitted to data generated from theoretical equations relating the percent Pu^{240}, percent U^{235}, change in g/ton of Pu and change in g/ton of U^{235} individually to the exposure in Mwd/ton. The polynomials fitted so well that several errors in the data from the theoretical equations were discovered.

The design of a test to assess the effects of carbon, iron, and silicon, each at three levels in the fabrication of uranium on the pre- and postirradiation characteristics of fuel elements was submitted to interested personnel.

The analysis of a test to assess the effects of can and core annuli thicknesses on the quality of the bond cladding was completed and submitted to interested personnel. The report included equations for the response surfaces for different yield variables as functions of the can and core annuli thicknesses as well as graphs depicting the same relationships.

A report on the effects of various heat treatment variables in the manufacture of uranium on the preirradiation warp of fuel elements was completed and submitted to interested personnel. A considerable number of statistically and practically significant effects were found among the core size, quenching medium, delay time, and quench temperature variables.

The first phase was completed of a study of retail trade potential in the Tri-City Area. Data estimates were related to 1962 as a base period. The next phase will carry projections through 1967.

Work was begun on Tri-City Area transportation problems. This included a preliminary analysis of local airline service as a connecting link to long-distance service, and local highway needs for future growth and development.

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A test tape produced for the experimental δ-ω lathe exhibited unacceptable bit-spacing characteristics that have never been encountered before. Extensive checks of the EDPM program CUPID which produces these tapes indicate the trouble arises from a seldom experienced case of machine round-off error induced by a specific geometry. An appropriate test and double precision subroutine is being written for insertion into CUPID to prevent this situation from reoccurring.

Work was completed on the major sections of the FORTRAN language "Bubbles" program for the quantitative analysis of metallographic data collected on the Ziess particle size analyzer. Debugging of the program continues and sections pertaining to special situations will be added in the near future.

A closed form solution was obtained to a two-dimensional mathematical model for the propagation of a small crack in an elastic body. Further studies were made on methods of applying conformal mapping techniques on a similar problem.

A new main calculation program for the IRA system was put into production service during the month and recalculation of the old RCA data started. A new change and error report has been specified for the Program IRA-325. The present report is excessively long, time consuming to produce, and difficult to read. The new report is an improvement in all these areas.

Statistical analysis of data from a Pm^{147} study on beagle dogs was completed. The excretion data of promethium in urine and feces were fitted as power functions of time and the parameters of these functions for inhaled and intravenously administered Pm^{147} were compared.

The statistical analysis of data from the study to investigate the effect of treatment with DTPA on the clearance of Np^{239} and Pu^{239} from various rat tissues was completed.

SUPPORTING FUNCTIONS

PRTR output for March was 1302 Mwd, for an experimental time efficiency of 72% and a plant efficiency of 60%. There were nine operating periods during the month, seven of which were terminated manually and two were terminated by scrams. The longest sustained critical period in PRTR history was realized this month. A summary of the fuel irradiation program as of March 31, 1964, follows:

	Al-Pu		UO_2		PuO_2-UO_2		Other		Program Totals	
	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd
In-Core	0		7	1 366.8	77	10 783.0			84	12 149.8
Maximum				277.6		259.8				
Average				195.3		140.0				
In-Basin	7	569.9	25	2 686.1	31	2 096.5			63	5 352.5
Buried							1	7.3	1	7.3
Chemical Processing	68	5 465.8	35	1 965.8					103	7 431.6
Program Totals	75	6 035.7	67	6 018.7	108	12 879.5	1	7.3	251	24 941.2

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

Heavy water and indicated helium losses for March were 1791 lb and 118,521 scf, respectively.

A total of 76 reactor outage hours were charged to repair work. Main items were:

Primary ion exchanger replacement	19 hr
Shim rods	14 hr
Leak repairs (heavy water and helium)	15 hr
Weld repairs (primary pump seal coolant line)	8 hr

Modifications were completed in the Plutonium Recycle Critical Facility to correct several operational problems encountered in placing the PRCF light

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water EBWR core into service. EBWR fuel was received, inspected, and loading began on March 17, 1964. Criticality was achieved on March 27, 1964. Rod calibration experiments were in progress at month-end.

In the Fuel Element Rupture Test Facility a leak developed at a weld in the cold inlet section of the Rupture Loop. Repairs are complicated by the location of the piping section that is behind heavy shielding. Repairs were started during reactor operation by discharging the fuel element and providing minimal flow for process tube cooling. Repair efforts continued through month-end.

The chemical processing of 35 Al-Pu fuel elements spent PRTR fuel was begun at Redox.

Total productive time in Technical Shops Operation for the period was 17,932 hr. Distribution of time was as follows:

	Man hr	% of Total
N-Reactor Department	2 002	11.2
Irradiation Processing Department	3 542	19.7
Chemical Processing Department	664	3.7
Hanford Laboratories	11 724	65.4
Hanford Utilities and Purchasing Operation	0	0

Total productive time in Laboratory Maintenance Operation was 18,800 hr of 20,300 hr potentially available. Of the total productive time, 74.2% was expended in support of Hanford Laboratories components, with the remaining 25.8% for service to other HAPO organizations. Manpower use (in hours) for March was as follows:

A. Shop Work		1700
B. Maintenance		8900
1. Preventive Maintenance	2500	
2. Emergency or Unscheduled Maintenance	1200	
3. Normal Scheduled Maintenance	5200	
C. R&D Assistance		8200

The heavy water inventory at the end of March 1964 showed a loss of 1792 lb valued at $24,801 for the PRTR. Heavy water scrap generated during the month amounted to 3063 lb, resulting in a $4319 charge to operating cost. Heavy water accumulated at March 31, for return to SROO amounted to 17,359 lb valued at $215,810.

Cumulative data of Hanford visitations follows:

	Number of Visitors	
	In March	Since June 13, 1962
Visitors Center	1312	63,969
Plant Tours	273	n.a.

HAPO professional recruiting activity this month is summarized below:

	Plant Visits	Offers Extended	Acceptances Received	Rejections Received	Open Offers At Month End
Ph.D.	12	6	2	1	5
BS/MS (Direct Placement		3	0	0	3
BS/MS (Program)		8	4	14	47

Six technical graduates were placed on permanent assignment. Three new members were added to the roll and one terminated, bringing the program total to 58 at month end.

Authorized funds for eight active projects total $7,543,500. The total estimated cost of these projects is $10,550,000. Expenditures on them through February 29, 1964 were $2,605,000.

Manager, Hanford Laboratories

HM Parker:JEB:dph

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REACTOR AND FUELS LABORATORY MONTHLY REPORT

MARCH 1964

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 02 PROGRAM

1. Metallic Fuel Development

Fuel Fouling Detector. A prototype N-Reactor fuel fouling detector prepared from Zircaloy-2 clad thorium-uranium rod has operated successfully in KER Loop 2 since February 16 of this year. The central temperature of the fueled rod is 450 C and has remained reasonably constant. The element is due to be discharged at the next reactor shutdown at which time a second detector will be charged in KER Loop 4.

Empirical Fuel Swelling Expression. The additional swelling and operational data from KER Loop 3 and 4 tests of NAE's has permitted the continuing development of an expression for fuel swelling. Analysis of the data from 110 N-fuel components and 20 special co-extruded single tube elements has resulted in three similar expressions, one for each fuel geometry. Agreement between the calculated and experimentally determined fuel swelling is good.

Target Element Development. Post-irradiation examination of the aluminum-2 wt% lithium alloy core of one of eight target elements irradiated for 124 days in a KER loop showed that swelling had caused the central 0.252-inch diameter hole in the target to decrease in diameter by 0.019 inch. The density of irradiated core was measured as 2.51 g/cm^3 as compared to the pretest value of 2.57 g/cm^3. Post-irradiation annealing studies of sections machined from the core are being made to further study the effect of temperature on swelling.

Lithium-Aluminum Alloy Corrosion. Lithium-aluminum alloys with up to 3.5 wt% lithium have been tested for 750 hours in 100 C water. Weight gains and visible changes in the coupons being tested have been very slow since the first few hours of testing. Coupons are being corroded in pH 4.5, 7 and 10 water. Corrosion rates increase with increasing pH.

Another group of these alloys has been autoclaved for 50 hours in 200 C, pH 7 water. Weight changes have been accelerated at this temperature. Alloys of 2.0 and 3.5 wt% lithium have decomposed or are beginning to decompose. Additions of lithium up to 1.5 wt% seem to increase the corrosion resistance of aluminum. Cold work increases the corrosion [redacted]s somewhat.

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Fluted Fuel Element Irradiation. The N-single tube fluted fuel element is in the 10th reactor cycle of irradiation in the ETR. It is currently operating at a maximum specific power of 140 kw/ft with a corresponding maximum fuel temperature of 550 C. The accumulated maximum exposure is currently 180 Mwd/ton and the measured volume increase is 0.9%.

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

Electron microscopy has been completed on replicas from a sample of U-2 wt% Zr irradiated to about 0.25 at% burnup at a temperature below 600 C. This fuel rod showed extensive grain boundary tearing in a band near the outer edge of the fuel and aligned porosity within the grains at the center of the fuel rod. The observed grain boundary tearing is associated with deformation boundaries within grains as well as grain boundaries.

N-Reactor Fuel Evaluation. An outer component (NOE) of an N-Reactor fuel element irradiated to 2000 Mwd/ton in a KER loop has been induction heated over its full length to 980 C and water quenched. This test was conducted in Radiometallurgy by Particulate and Gaseous Waste Research (HL) and is a continuation of the work being done in support of N-Reactor hazards studies. After quenching from the high temperature the surface of the outer clad displayed a large number of small mounds, arranged somewhat in rows, but the integrity of the clad does not appear to have been compromised. The inside clad surface and the outside clad surface at the ends ($1\frac{1}{2}$ inch in from closures) appeared unaffected. It is believed that the severe thermal cycle experienced by the element has fractured the co-extruded bond and the difference in the thermal expansion of the Zr-2 clad and the fuel has resulted in clad buckling to produce the large number of small mounds. An inner tube will be tested in the same manner followed by a detailed destructive examination of the components to further assess the damage and possible hazard potential.

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 irradiation behavior of single tube 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. The test fuels will be irradiated in the beta heat treated condition, with the second phase particle distribution resulting from that treatment. Structural characterization was made of sections of the ingots, primary extrusions, coextruded fuel and heat treated material. The fuel material altered by the Fe and Al additions has a higher recrystallization temperature, a finer alpha extruded grain size and a greater density of second phase particles. Hardness tests at 300-400 C with a range of loads and over several time intervals with the material in the beta heat treated condition have shown that the Fe-Al alloy is harder, but has essentially the same strain hardening exponent and creep rate as the standard fuel. Tensile tests have been completed at room temperature and 100 C for the two materials and show the increased strength and reduced ductility of the material containing Fe and Al additions.

<u>Alternate Uranium Compositions</u>. Studies are in progress to determine the effects of altered fuel compositions upon fuel element fabrication, corrosion behavior, and irradiation performance.

Uranium alloys of eight different compositions were received from Mallinckrodt Chemical Works. These eight alloys are part of a series of 14 uranium alloys planned for irradiation in capsules to study swelling performance. The alloys received from Mallinckrodt consisted of U + 800 ppm Al, U + 800 ppm Si, U + 260 ppm Al + 300 ppm Si, U + 1.8 wt% Zr + 700 ppm Al, U + 1.5 wt% Zr, U + 2.3 wt% Zr, U + 1.7 wt% Nb + 400 ppm Al, and U + 500 ppm P. These alloys, produced as dingots from the bomb reduction process, have less than 60 ppm carbon. Two-inch diameter rods were fabricated from the dingots by gamma extrusion. On quenching from the beta heat treating temperatures the two-inch diameter coextrusion billets of U+Si, U+Al+Si, and U+Al alloys cracked, apparently along the radially oriented grain structures. The U+P alloy was successfully heat treated with no indications of billet cracking. The remaining alloys and three billets of the U+Si alloy were coextruded without prior beta heat treating to 0.6-inch diameter rods. The Zircaloy-2 cladding thickness uniformity is good on all the coextrusions except the U+P and U+Si alloys. The U+Nb+Al alloy required gamma phase coextrusion in order to achieve a clad rod having good cladding uniformity. Seven other alloys of uranium with various amounts of Fe, Si, Al, and carbon are now being produced using vacuum induction and vacuum arc melting.

Evaluation of a series of binary and ternary alloys of uranium with up to 5 wt% total niobium and/or zirconium is continuing. Hardness measurements and metallography are completed on these arc melted alloy buttons. The buttons were rolled into sheet and corrosion coupons made. The coupons were then given alpha, beta, and gamma heat treatments. Metallography and electromotive force measurements are now being done on the coupons. Preliminary corrosion testing of these alloys in pH 7, 100 C water will be done in the next month.

Corrosion of Braze Alloys. The corrosion tests on various brazing alloys have been enlarged to test the effect of iron and copper on the corrosion of the basic Be-Zr-2 alloy as well as several alternate braze alloys, namely Zr + Fe + Cr, Zr + Ni + Cr, and Zr + Ni + Fe + Cr. Results up to 300 hours in 360 C, pH 7 water show that iron decreases the corrosion resistance up to 1%, then continues to increase corrosion resistance up to the eutectic point of 12% Fe. Copper at the first eutectic point of 14% shows a somewhat poorer corrosion resistance than in the other braze alloys; however, the corrosion weight gain of even the worst alloy is still less than 100 mg/cm^2.

To cope with the large mass of corrosion data, a Fortran program has been written. This program, CORAN (for corrosion analysis), was written to make the calculations for weight change and optical properties. The program involves a two-pass system, the first pass to make all calculations, the second pass to make the necessary plots. This has the advantage of short computer time for routine calculations, then a separate plot pass whenever enough data have been gathered to warrant the time. A master file is maintained on magnetic tape, while all data are fed into the computer on punched cards. This program allows complete calculation for each coupon in less than three seconds, an operation that formerly required up to 10 minutes for engineer's time. Even greater time savings will be realized when the plot pass, using the Benson-Lehner plotter, is operable.

N-Reactor Outer Fuel Cladding Dimple Problem. A considerable number of N-outer fuel elements are being rejected in N-fuel production because of small isolated thickened areas in the inner cladding termed dimples. Assuming that the dimples are caused by defective material in the billet components, NRD supplied MFDO with eight billet components, consisting of four N-outer Zircaloy-2 ID billet sleeves cut in half with copper cores. These were canned in copper by vacuum beam welding and extruded at 600 C at a reduction of 12 to 1.

Extrusion force varied by 18% between some billet sleeves; however, between the matching half lengths, the force varied very little if any. After the copper was etched off the extrusions, a few small dimples appeared on the OD surface of one of the Zircaloy-2 extrusions. The dimpled areas are now being prepared for microexamination. The copper cores which were placed in the ID of the billet sleeves did not fully clean up during machining, and the dimples may have resulted from these areas.

Second Generation N-Outer Support. An attempt to alpha form the N.O.T. II support shape in the alpha temperature range on the 600 KVA welder was unsuccessful. Not enough forming force was available on this machine to form the parts at that temperature. Beta phase forming is being investigated to determine if forming temperature has an effect on support strength variability. No accurate determination of forming temperature has been made, but visual estimates (color) are about 900-950 C for the lowest temperatures at which the part can be fully formed with 12 tons force. Four heat inputs and two wire sources (8 categories) are being evaluated. No significant change in strength variability has been noted in the supports in the "as formed" condition. The same categories are being evaluated following a 400 C steam autoclaving for 72 hours. Ten supports have accumulated 504 hours of exposure to 360 C water with no evidence of abnormal corrosion behavior. This group is continuing to accumulate autoclave exposure.

Approximately 60 supports have been formed from 1/8-inch diameter Zr - 2Nb - 1Sn alloy wire. The parts were formed at conditions similar to those used for forming the Zircaloy-2 wire. These forming conditions (high temp-rapid quench) should produce maximum strength in this alloy. Half of these supports were heat treated (575 C, vacuum for 2 hours, furnace cooled) for best corrosion properties, and will be compared to the "as formed" parts for strength properties.

Welding. Fifteen, four-inch long, hot-headed, N-Reactor outer tube sections closed last month by resistance projection welding have been undergoing evaluation. Metallographic examination of five closures show weld nuggets in various states of development in each closure. Another five have been autoclaved 36 hours in 400 C, 1500 psi steam. All five appeared excellent when taken out of the autoclave. Two of these are being thermal shock tested from 600 C to water quench. After 10 quenches they will be metallographically examined for any sign of cracking or other failure. Four pieces had been adjudged to have insufficient welds, but on the strength of the metallographic examination and autoclave results to date it is planned to autoclave these in reverse order of apparent weld quality to determine a minimum point.

Aluminum-Lithium Alloys. Duplicate samples of unclad aluminum-lithium alloys containing 1, 2 and 3 wt% lithium (nominal compositions) were exposed to deionized water at 300 C (572 F) for 10 days. All six samples were completely oxidized. At the end of the test the water in the autoclave was at pH 7.4. A future test will be conducted on irradiated Al-Li alloy samples to investigate the degree of retention of tritium in the aluminum oxide.

Rupture Testing of Irradiated Fuel Elements. An N-Reactor inner fuel element which had been previously irradiated to an exposure of 2100 Mwd/ton was rupture tested in the IRP Loop. The element was defected in the side with a 0.025-inch hole. It was exposed to flowing water at 300 C until 15 minutes after detection of the rupture, and then cooled at the N-Reactor cool-down schedule. The 15-minute delay is considered to be the maximum time required to confirm a rupture in N. There was a raised and torn area about four inches long around the entire circumference. The weight loss before cleaning to remove UO_2 was 60 grams. This element was considerably more severely ruptured than those in similar previous tests when the temperature was held at 300 C only five minutes following detection of the rupture.

2. Corrosion and Water Quality Studies

Corrosion of Zirconium in Heated Crevices. A corrosion test was completed of an electrically heated Zircaloy-2 clad specimen to which three N-suitcase handle fuel element supports had been welded. The specimen had a heat flux of approximately 600,000 Btu/hr-ft^2 and was exposed in TF-2 in recirculating water at 288 C, with NH_4OH added to give a pH of 10. The test was terminated after 19 days due to burnout of the heating element. The specimen was sectioned through the welds and examined under the microscope. Only slight corrosion (less than 1 mil) was found in the crevices under the supports. This is in agreement with observations on a similar specimen exposed under the same conditions reported last month. Severe corrosion had been found in the crevices in previous tests using LiOH for pH control.

Testing of NH_4OH in KER Loops. Data from KER-1 continue to indicate absence of crud problems during the 26 EFPD of operation following the decontamination of this carbon-steel loop. The loop has operated at 282 C outlet, pH 10.0 - 10.1 with NH_4OH. Thermocouple readings from the instrumented crud detector indicate no detectable crud buildup on the fuel elements. Hot crud concentrations measured in the coolant have been low; a maximum concentration of 26 ppb just after decontamination has decreased to about 10 ppb. (The long term average for this loop is 25 ppb crud.) Average total solids has been 3.4 ppm, compared to 1.4 ppm before decontamination. (This difference is also due in part to higher total solids in the makeup water, presently 2.1 ppm versus 1.2 ppm previously.)

The rate of loss of ammonia in KER-1 by radiolysis has been 0.036 lb/day at an addition rate in the feed water of 0.18 lb/day and a concentration of 12-13 ppm NH_3 in the loop. Total gas concentration in the loop averaged 5.9 cc/liter.

Tests were made in KER-3 to determine the effect of residence time (or bleed rate) on the total gas concentration in the coolant. The loop operated at 270 C, pH 10.1 - 10.2 with 14-15 ppm NH_3. The gas content was 13.3 cc/liter at a bleed rate of 1.5 gpm, 27.4 cc/liter at 0.9 gpm, and 45.3 cc/liter at 0.4 gpm. Previous samples have indicated that the total gas is nearly all H_2 and N_2 with a ratio of H_2 to N_2 of 2.0 to 2.5.

Fretting Corrosion of N-Fuel Elements. Two N-Reactor target elements are being tested in TF-7 to determine their susceptibility to fretting attack. External vibration at the calculated natural resonance frequency of the fuel elements (70 cps) is being applied to the external side of the test section, which consists of a horizontal length of N-Reactor Zircaloy-2 process tube. Flow is about 40 gpm; temperature 277 C. After two weeks no fretting has been observed between the inner and outer fuel elements. Fretting was observed on one external support at the point where a carbon steel shoe had come off (probably during charging). The depth of this fretting was estimated to be 1-5 mils. Two fretted areas were also found on the process tube. It is not known whether these were formed during this test or during a previous similar test. However, at least one of the fretting marks on the process tube must have been caused by a carbon steel shoe on a fuel element in this or a previous test.

Corrosion in N-Reactor Secondary System. Tests were conducted to determine whether bronze valve components would be subject to stress corrosion cracking if exposed to ammonia vapors from water leaking from the N-Reactor secondary system. Stressed samples of commercial bronze were exposed to vapors above pH 9 ammoniated water for 30 days at room temperature and at 90 C. Microscopic examination revealed cracks in two samples. Since the bronze is very brittle, these cracks could have been formed while stressing the samples. Metallographic examination will be performed to determine the cracking mechanism.

A test was conducted to determine whether corrosion problems would result if aluminum were used in a floating roof of the deionized water storage tank for the N-Reactor secondary system. Samples of 1100 alloy aluminum were exposed in the laboratory for three days in 150 F deionized water adjusted to pH 9 with NH_4OH. Large pits (30-mil wide) formed in crevice areas and smaller pits (1-2 mils) formed in noncrevice areas.

Evaluation of Cleaning Solution for N-Reactor. The fourth and final ex-reactor loop test to evaluate Alk-14 to clean rust and other oxides from the N primary piping was conducted in the carbon steel TF-5

Loop. (Alk-14 is a proprietary cleaning solution containing ammonia, EDTA, and inhibitors.) A 2.4% solution was added to the loop and circulated for 17 hours at 121 C. Since the solution tended to become depleted, some Alk-14 was added as required to maintain the concentration at $\frac{1}{2}$ - 1%. There was no visually evident corrosion on the N-Reactor fuel elements. Redeposition of iron oxides did not occur, as it had in previous tests where the Alk-14 was allowed to become depleted to very low concentrations.

Another test of Alk-14 showed that this material is effective in removing oxides formed on stainless steel. This indicates it has good potential as a single step decontaminant.

N-Reactor Coolant Analyses. An automatic gas chromatograph designed to monitor dissolved H_2 concentrations in water was installed in the laboratory and testing started. Performance has not been stable enough to permit accurate determination of sensitivity. It appears that it can detect O_2 and N_2 as well as H_2.

The interference of H_2O_2 in measuring O_2 with the Hays automatic analyzer was investigated over a range of H_2O_2 concentrations up to 200 ppb. The presence of H_2O_2 causes an increase in the measured O_2 reading proportional to the concentration of H_2O_2. The interference factor was 0.38 ppb O_2 increase per ppb H_2O_2 present.

3. Gas Atmosphere Studies

Zirconium-Graphite Compatibility Loop. Measurements of the rate of water diffusion through a full size N-Reactor tube block have been completed at 350 to 650 C (662 to 1202 F). When the potential rate that water can move through the tube block was compared with the rate at which a Zircaloy-2 process tube reacts with water, it was determined that a minimum water content of 325 ppm (-31 C dew point) is required to provide a nonhydriding environment for the Zircaloy.

The loop is now circulating an He atmosphere contaminated with 1000 ppm H_2O, 200 ppm H_2 and 150 ppm CO_2, simulating actual reactor flow rates. A 24-inch full size graphite oxidation bar has been inserted. Water vapor which diffuses through the tube block will oxidize Zircaloy-2 coupons which will be analyzed for corrosion and hydriding.

Effect of Thick Films on Hydriding of Zircaloy-2. Zircaloy-2 coupons with various amounts of oxide film are being exposed to an He, 1% H_2, 1% CO, 0.1% H_2O gas at 350 and 450 C (662 and 842 F). After 68 days of exposure, a slow hydriding reaction is continuing on thick film samples (> 200 mg/dm^2) while samples with less than 100 mg/dm^2 are not absorbing gaseous hydrogen. An expanded experiment is continuing with five film thicknesses, four temperatures and three gas compositions.

Graphite Burnout Monitoring. Small, graphite burnout monitors exposed in channel 1880, KW-Reactor, for about three months showed the absence of any peaks that could be attributed to air or water. The highest measured rate was 0.3% per 1000 operating days (%/KOD). Monitors in channel 3478, D-Reactor, for 3½ months showed no peak at 80 inches into the stack, a position characteristic of the graphite-oxygen reaction; however, a peak of about 1%/KOD appeared at the centerline which is indicative of the graphite-water and/or carbon dioxide reaction. Monitors shielded from the reactor atmosphere by 3/8 inch of graphite showed slightly lower rates.

Graphite-Water Vapor Reaction. The rate of reaction of TSX graphite with water vapor, in the absence of hydrogen was reported in HW-81019 A to be of the form:

$$R + k_o e^{-E/RT} \; p_{H_2O}{}^{1/2} \qquad (1).$$

The following mechanism is proposed to account for the observed half-power dependence on water vapor:

$$2\ C_f + H_2O \underset{K_2}{\overset{K_1}{\rightleftarrows}} C(O) + C\ (H_2) \qquad (2)$$

$$C\ (H\ ) \underset{K_5}{\overset{K_4}{\rightleftarrows}} C_f + H_2 \qquad (3)$$

$$C\ (O) \xrightarrow{K_3} CO + C_f \qquad (4)$$

where C_f represents a free active surface site, and C (O) and C (H_2) are sites occupied by oxygen atoms and hydrogen molecules, respectively.

In the present experimental system the partial pressure of hydrogen and carbon monoxide is negligible; hence, the reverse reaction in (3) and the reverse of (4) can be neglected. If it is assumed that reaction (2) rapidly approaches equilibrium and that the forward reactions of (3) and (4) are slow compared to (2), a square-root dependence on water-vapor pressure is obtained for the rate at low pressures. In future tests the effects of hydrogen on the reaction will be studied and the mechanism for the reaction will be re-examined.

Further analysis of accumulated rate data in this study show that measured reaction rates on un-pretreated material may vary by a factor of four from sample to sample. Preliminary treatment by oxidation in air (1.5% to 6%) or outgassing at a high temperature (850 C) for about 70 hours reduces the variation to a factor of 2. The pretreated samples oxidize about eight times faster than the non-pretreated samples, however.

4. Thermal Hydraulic Studies

Analysis of Two-Phase Pressure Drop Data for N-Reactor. 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. Methods were developed for accurate calculation of pressure drop for a tube of reactor fuel elements based on laboratory data. Reactor conditions cannot be obtained directly from laboratory results since small differences exist in dimensions and the laboratory experiments were not run at every single condition that could be encountered at the reactor.

Pressure drops in various portions of the laboratory test section were analyzed separately to find correlations which could be used for extrapolation of the results to the reactor conditions. These correlations are best expressed as relationships between two-phase and single-phase pressure drops since the single-phase pressure drop for reactor conditions is known with good accuracy.

Comparisons of experimental two-phase to liquid-phase pressure drop ratios with those predicted by the Levy momentum model were extended to cover all pressures and system sections. As reported last month, the Levy model represented the experimental data for the outlet piping at 1200 and 850 psig quite well. However, the subsequent comparisons showed that at 350 psig, ratios predicted by the Levy model were 20 to 30% lower than the measured values for this part of the system. For the dummy column and the active (fuel column) channels, agreement between the Levy model and the experimental data was poorer at all pressures, with disagreements in the range of 20 to 50%. However, these portions contain fittings, bends, perforated dummies, and other factors, and it is not surprising that the Levy model does not fit the experimental data exactly.

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In consideration of other correlations, two-phase to liquid-phase pressure drop ratios were plotted against two-phase to liquid-phase specific volume ratios. Essentially, this approach assumes that the two-phase mixture is a homogeneous fluid having a density determined by the pressure and relative fractions of liquid and vapor. For the dummy column, outlet nozzle, and outlet connector, this approach brought the data for the 850, 1200, and 1400 psig experiments into excellent agreement with each other. The 300 psig pressure drop ratios fell 10 to 15% below those for the higher pressures at the same specific volume ratio. For the flow channels in the active section, data at 300, 850 and 1200 psig showed good agreement with each other also, with the 1400 psig ratios falling slightly low. These pressure drop ratios were not always what would be predicted by the homogeneous model, but this approach did bring data for most pressures into close agreement with each other. It was concluded, therefore, that this method would yield relationships suitable for extrapolating the experimental results for reactor calculations.

Determination of Boiling Burnout Conditions for the N-Reactor. A digital computer program was written to determine flow rates at which burnout would occur at various tube powers, system pressures, and reactor inlet temperatures. Plots of burnout flow versus tube power, determined from this program, will be superimposed on the pressure-demand curves developed from the two-phase pressure drop experiments, to show conditions under which burnout, and/or flow instability would occur.

This computer program is based on results of burnout experiments performed earlier with electrically heated short models of N-Reactor fuel elements. Heat fluxes are calculated at the fuel element surfaces at various axial locations. For a specified flow rate, burnout heat fluxes are calculated at the same locations, using relationships between burnout heat flux, mass velocity, and enthalpy which were developed from the burnout experiment results. Point-by-point comparisons of heat flux and burnout heat flux are made, and if the lowest ratio of burnout heat flux to heat flux is between 0.99 and 1.01, the power, flow, and profiles of heat flux and burnout heat flux are printed. Otherwise, the flow is readjusted and the calculations are repeated. Calculations may be performed for a number of pressure and inlet temperature combinations and powers. Comparisons can also be made for various burnout safety factors.

Transient Heat Transfer Experiments for N-Reactor. The laboratory program to determine flow, pressure, and temperature transients following a sudden rupturing of the N-Reactor primary cooling system was started. The experiments are being performed using the electrically heated, full-scale model of the downstream half of an N-Reactor fuel column, with prototypic N-Reactor process tube, and inlet and outlet fittings and connectors. The first experiments involve simulation of a reactor inlet connector rupture. Normal flows, pressures, and temperatures are established in the experimental system through a recirculating flow loop. The transient is initiated by actuating quick-acting valves which simultaneously isolate the test section from the recirculating loop, open the connector to a drain, and supply water to the test section from a pressurized tank. The tank can supply 1000 gallons of water at pressures up to 2500 psig.

Two transient experiments were conducted, with coolant temperature of 400 F and with pressures of 350 and 1400 psig. These first experiments were run to determine points of steam formation and to check on the system and equipment response, with no power generation in the fuel column model. Equipment performance was very satisfactory. Subsequent transient experiments will be conducted at typical reactor operating conditions with power generation.

<u>N-Reactor Boilout Test</u>. One of the hazard problems in the N-Reactor is the case where coolant flow is lost due to a ruptured header, the reactor boils dry, and the fuel reaches a temperature of 2000 F before coolant flow is re-established. When the flow is re-established, large quantities of steam will be released. Since this steam will probably contain fission products, it will have to be contained in the reactor building. The question is whether the quantity of steam released will over-pressurize the building.

A test assembly is being installed in the heat transfer apparatus with a large bypass flow to maintain constant inlet pressure. The test section will be brought up to a temperature of 1500 to 2000 F, and then flow at a constant inlet pressure will be started. Simultaneous measurements of inlet flow, inlet and outlet pressure, rod surface temperatures, and steam or steam-water mixture discharge will be made.

The test section consists of a 23-foot long cosine heater rod installed in a stainless steel pressure tube. Although this combination is not an exact replica of the N-Reactor fuel tube, the results will allow a valid check on the analytical method being used to calculate the event for the reactor.

<u>Hydraulic Tests for Present Production Reactors</u>. It has been thought possible that some of the hydraulic fittings on those reactors

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scheduled for shutdown might become available for use on the remaining reactors. Therefore, a flow test was performed on the use of B-D-F-Reactor CG-558 front nozzles on the rear of B-D-F-type reactor. The only modification made was to cut two inches off the nozzle barrel to facilitate the use of the present B-D-F rear nozzle cap. Preliminary tests indicate that an increase in flow rate of approximately 12% over that flow encountered with the present B-D-F outlet assembly is possible. The mechanical characteristics of the outlet hardware were not thoroughly examined in this short test and only rough, but conservative, estimates of increased flow performance were made. However, the results do suggest that a testing program might be worthwhile to examine the possible advantages of modifying the CG-558 front nozzles for use on the rear of those reactors which will not be included in the shutdown program.

5. Shielding Studies

N-Reactor Shield Evaluation. Data were analyzed from cold N-Reactor N-1 irradiation startup test. Final activation data closely agree with the available fission chamber traverse data. The gamma ray ion chamber data appear consistent. The neutron spectrometer data have been compared with a MAC calculation at the same dose point in one of the other production reactors, and there is reasonable agreement. The N-Reactor geometry is currently being programmed into MAC for calculations to be made soon to provide a better comparison with the neutron spectrometer data. A neutron spectrometer background counter has been recently received from ORTEC and current plans are to remove the neutron spectrometer equipment from the N-Reactor following the N-2 irradiation and install this equipment at the PCTR. Using an N-Reactor lattice, a neutron spectrometer background count will be taken in addition to a spectrum count to see if improvement in the measured spectrum can be obtained.

A front and rear shield calculational model of the existing small production reactors has been made for MAC and a calculation has been made to determine the suitability of these shields if a reactor is rebuilt for power production. Results of the calculation are currently being analyzed.

The MAC computer code abstract was submitted to Nuclear Science and Engineering, and the code and cross section decks have been submitted to the Argonne Code Center.

6. Graphite Studies

N-Reactor Irradiations. The capsules for the long term irradiation of N-Reactor graphite continue to operate successfully. Data from the samples and flux monitors contained in the H-5-3 capsule are being analyzed. Preliminary length-change data show continued contraction for all samples.

Based on data from the flux monitors contained in the eight capsules in the series, through the H-5-3 capsule, and on recent computer calculations of reactor spectra the neutron exposures to the samples have been recalculated. These exposures are somewhat less than previously determined exposures which were based on the three energy group computation made by the GETR physicists. The relationship between exposures in the GETR and N-Reactor has been documented in HW-81026, "GETR Graphite Irradiation Capsules H-4,5,6. Equivalent N-Reactor Exposure."

Neutron Spectrometry. A series of tests was conducted during the full-core physics startup tests of N-Reactor to determine fast neutron spectra at several positions within the core. The purpose of this experiment was to determine if there are shifts in the fast spectra which could cause rates of damage accumulation in graphite to vary significantly. Tests were made at the reactor centerline, one-half lattice unit from the centerline, at the core boundary, and three lattice units into the core from the boundary. A lithium-6 detector system was used in these experiments. The tests were all performed by lowering the detector into a thimble located in ball channel 60 to the desired elevations. Data are presently being analyzed.

B. WEAPONS - 03 PROGRAM

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

C. REACTOR DEVELOPMENT - 04 PROGRAM

1. Plutonium Recycle Program

Fuels Development

PRTR Fuel Exposures. An accumulated exposure in excess of 5000 Mwd/ton was achieved by vibrationally compacted and by swage compacted UO_2-PuO_2 fuel elements operating in PRTR. Both elements contain incrementally loaded UO_2-0.48 wt% PuO_2. Continued irradiation to exposures in excess of 10,000 Mwd/ton are planned.

PRTR Fuel Element Failure. A vibrationally compacted, impacted UO_2-1 wt% PuO_2 element (325 Mwd/ton) that failed during full power operation in PRTR without loss of fuel contained fuel that had been outgassed 48 hours at 530-580 C. Inadequate outgassing is presumed because of malfunction of the diffusion pump. Analysis of archive fuel samples showed organic contamination.

Fuel Element Refurbishing. One UO_2 fuel element (1067) which had a broken rod wire wrap was repaired in the PRTR basin.

PRTR Fuel Fabrication. Two Vipac and four swage compacted PRTR fuel elements were fabricated. Fuel rods for three 19-rod clusters of 0.48% PuO_2 of special Pu-240 concentrations required for physics tests were fabricated.

New dies installed in the swage machine have a six degree taper. A smoother final cladding finish is obtained.

Fifty PRTR tubes and end caps were prepared for magnetic force weld closure on the first end cap.

A full penetration, fillet head weld was developed which eliminates the crevice between the end cap and tubing of vibrationally compacted fuel rods.

After various modifications to the external control circuits to compensate for occasional erratic functioning of the Dynapak machine used in densification of UO_2-PuO_2 were unsuccessful, a major overhaul of the set-pressure unit was made. During this overhaul it was discovered that the set-pressure piston was defective. Necessary repairs were made, and the machine returned to operation.

Vacuum Outgas Tests of Doped UO_2 Fuel. Samples of oil-doped UO_2 fuel were vacuum outgassed at 250 C and 500 C in order to verify the possibility of hydrocarbon contaminants breaking down under severe outgassing conditions. At the 500 C outgassing temperature, hydrogen was the principal gas driven off, and there was a significant amount of short chained hydrocarbons. At the 250 C outgassing temperature, the amount of evolved gas was negligible. It was concluded that vacuum outgassing temperatures should be limited to 250 C maximum until further tests determine the nature of hydrocarbon decomposition at higher temperatures.

Fuel Impaction Studies. Approximately 1500 pounds of ThO_2 for test irradiation purposes were pneumatically impacted in three weeks. The entire run was made using one 4-inch diameter tungsten carbide punch, tool steel die, and back-up plug. The tungsten carbide punch has now been used in over 200 impactions.

Phoenix Fuel. A simplified process utilizing "spiral machining" and "coin pressing" techniques was developed and is now being used for the fabrication of Pu-Al wafers.

Ten kilograms of Al-20 wt% Pu (8 wt% Pu-240) alloy were cast into five cylinders from which 0.020-inch wafers will be machined. Radiographs indicate that nearly half the required 9000 wafers can be obtained from these castings. The unused material will be recast and remachined.

PRCF Fuels. Preparation of cladding components and core materials for the fabrication of PRCF fuel rods is in progress. End caps have been received. The PuO_2 has been blended to a uniform Pu-240 content and is being calcined. Blending of the PuO_2 with UO_2 for impaction is beginning.

Eighty 3-foot long, 0.505" ID, Zircaloy clad fuel rods were fabricated by vibrational compaction for the Physical Constants Test Reactor. The UO_2-0.9 wt% PuO_2 fuel cores are compacted to 85% TD. This particular density is required to equate the grams per inch of core of these rods with other pellet rods with which they will be compared.

Fuel Element Development. Continued out-of-reactor loop testing of a wide pad PRTR element having misaligned top and bottom end fixtures revealed 0.001-0.005-inch wear of the process tube and element support pads after eight weeks. Very little had been noted after one and five weeks. The accelerated wear may be related to an increase of coolant temperature from 250 C to 305 C after five weeks of operation.

Evaluation of Fuel Impurity Effects. Gas samples from irradiated, impacted UO_2-1 wt% PuO_2 that had been deliberately contaminated with oil containing 87.2% hydrogen and 6.8% methane in 41.4 ml (STP) extracted from the 1.27 cm (0.500-inch) diameter by 7.0 cm (2.75-inch) long fuel compartment. Seventeen other capsules containing various quantities of water and oil impurities were irradiated to 1 x 10^{19} fissions/cc (500 Mwd/ton) at 394 w/cm (12 kw/ft) rod power.

Irradiation Testing Program for EBWR Prototypic Fuel Rods. Twelve test rods (six have prototypic fission gas plenums) containing vibrationally compacted, outgassed, impacted UO_2-1.5 wt% PuO_2 are being irradiated in the MTR. Seven additional capsules were discharged at exposures ranging from 0.41-0.64 x 10^{20} fissions/cm^3 (1800-2300 Mwd/ton UO_2). Five will be irradiated to 1.41-5.65 x 10^{20} fissions/cm^3 (5,000-20,000 Mwd/ton UO_2). One discharged capsule contained 3.70 ml of total gas in a void volume of 0.97 ml at 0.6 x 10^{20} fissions/cm^3 (2000 Mwd/ton UO_2). The heat flux on the Zircaloy cladding (1.07 cm or 0.42-inch OD) was 316 w/cm^2 (10^6 Btu/hr-ft^2). Thirteen new capsules (one with fission gas plenum) were fabricated with outgassed, impacted UO_2-1.5 wt% PuO_2 and are ready for shipment to the MTR-ETR.

Two capsules containing vibrationally compacted UO_2-2.5 wt% PuO_2 and physically mixed UO_2 and 2.5 wt% PuO_2 were discharged after reaching 1.5-1.6 x 10^{20} fissions/cm^3 (5300-5600 Mwd/ton UO_2).

Test Elements With Roughened Surfaces. Eight, Zircaloy clad, vibrationally compacted UO_2 fuel rods, each having different surface roughness characteristics, were fabricated for an in-reactor comparison study of methods for increasing allowable heat transfer rates into reactor coolants.

ThO_2-PuO_2 Irradiations. Five capsules containing ThO_2-PuO_2 (2.2 to 18.5 wt% PuO_2) hydrogen sintered pellets were discharged at 0.8 x 10^{20} to 1.3 x 10^{20} fissions/cm^3. Center fuel temperatures exceeded the melting point. Five similar capsules were discharged at 1.1 x 10^{20} to 1.4 x 10^{20} fissions/cm^3. Center fuel temperatures were 1700-1800 C. Post-irradiation examination of the latter five specimens revealed that radial fission product relocation is associated only with columnar grain growth in the fuel, similar to the behavior of UO_2, UO_2-PuO_2, ZrO_2-PuO_2, and MgO-PuO_2.

Corrosion Water Quality Studies

Plutonium Oxide Dissolution. Tests were conducted to measure the dissolution of PuO_2 powder in each of three sulfamic acid decontaminating solutions at 40 C and 70 C. From 0.4 to 1.6 mg Pu/liter was dissolved in one hour with no agitation. This dissolution rate is of the same order of magnitude as measured by ORNL for sintered PuO_2 in boiling 7-14M HNO_3.

Soluble Poison in PRTR Moderator. A literature survey was conducted to evaluate methods of removing boric acid from the PRTR moderator (proposed for use as a soluble poison during startup). Ion exchange techniques should be satisfactory for this purpose. Since only anion (borate) need to be removed, supplemental anion exchange units should be installed to prevent depletion of the main mixed-bed cheanup resin. It appears that a continuous spectrophotometric procedure would be suitable to monitor the boron concentration in the moderator.

Corrosion of Various Iron and Nickel Base Alloys. Various iron and nickel base alloys are presently being tested in 650 C (1202 F), 3000 psi deoxygenated steam. Following 28 days of exposure, the corrosion weight gains range from 16.6 mg/dm^2 for Hastelloy X to 707 mg/dm^2 for AISI 316L stainless steel. The corrosion weight gains for the same two alloys exposed 28 days in 550 C (1022 F) steam were 3.53 mg/dm^2 and 129 mg/dm^2, respectively.

Reactor Components Development

PRTR Pressure Tubes. PRTR tube 6079, discharged from channel 1354 on March 3, 1964, was cut into eight pieces. The average exposure received by this tube was 9.47×10^{20} nvt fast (326 SMWD). Two pieces having deep fret marks were shipped to Radiometallurgy.

Ultimate strength and elongation data from tests on ring tensile specimens were obtained on specimens machined from the annealed portion of irradiated PRTR Zr-2 pressure tubes. These data were compared with values obtained from burst tests and from conventional tensile tests of similar material. Ring tensile tests provide a reasonably accurate and precise measure of hoop strength and elongation.

Pressure Tube Monitoring. Two abnormal defects were observed during the examination of eight process tubes this month: (1) The existence of a blister defect in the tube in P.C. 1948 was established. On-site nondestructive inspection records indicated the presence of this defect but provided no information as to its nature. The tube was accepted because borescope examination after autoclave treatment showed no observable defect. (2) A fretted area, associated with a wide pad at the fuel element (F.E. 5170) lower end bracket in P.C. 1653 was found to be 0.013-inch deep. This fretting mark formed over the lower half-length of an old fretting mark produced by a narrow pad some time ago. The old mark, however, had been measured to be only about three mils deep. The fretting marks associated with the other two wide pads of F.E. 5170 were both less than one mil in depth. This behavior is atypical but so has been the general fretting behavior of P.C. 1653.

Second Generation Mechanical Shim Rod for PRTR. A new lower drive unit was assembled using the three zirconium lead screws. Two of the screws had been autoclaved; the third was installed in the as-machined condition. The new assembly was installed in the environment test facility and operated with a new driving head. Initial operation of this assembly appears to be much more satisfactory than was the aluminum screw assembly. No screeching or squeaking exists in the new assembly; the A and B rod run more nearly the same speed; and, significantly, the motors require less current to drive the assembly either up or down. No case of the ball nuts sticking on the zirconium screws had occurred in the 300 hours of operation to date.

Two lower heat sinks to protect the drive assembly were received from the vendor in February. These were tested, rejected because of leakage, and returned to the vendor. Subsequently, they were repaired by the vendor and returned. Because of their fabrication history, the two heat sinks have been subjected to the following tests:

1. Pneumatic test at 85 psig with the heat sink immersed in water.

2. Furnace test with cooling water flowing through the heat sink. Furnace temperature at 240 C and 3 gallons/hr of cooling water. Test continued for 150 hours.

3. Repeat the pneumatic test.

Neither heat sink showed any indication of leakage during the testing. The heat sinks are being installed on the shim rod prior to additional testing. It is planned to use this shim control rod in PRTR tests.

Fretting Corrosion Investigations

Ex-Reactor Tests. The eddy current fuel element was subjected to 695 hours of operation at temperatures greater than 400 F in the EDEL-1. During the test period the drift problem in the electronic detection system was isolated as being in either the coils and/or the transducer lead wires. Additional tests are planned to correct the drift.

During operation of the loop it has been determined that the relative displacement of the fuel element in the pressure tube changes at a frequency of about 1.25 cps. It was also noted that the makeup pump in the loop has a pumping frequency of 1.2 cps. The possible interaction of the two components is being investigated.

A fretted sample of Zr-2 was discharged from the EDEL-1 Magnedash autoclave after a total of 4,141,400 impacts at 2 cps. The fretted area was 2.0 mils deep compared to two previous samples which had fretted areas 13 and 22 mils deep after 4,073,375 and 31,114,000 impacts. The operating conditions were similar during all tests. Hydride concentrations were 200-300 ppm in the sample with 13 mils penetration and 400-500 ppm in the sample with 22 mils penetration. Hydride concentrations at nonfretted areas were 10-50 ppm. Analysis of the sample with the 2-mil penetration has not been completed.

In other tests metallographic examinations of seven samples from previous tests in TF-2 were made to determine the effect of fretting on Zircaloy-2 hydriding. All the samples had fretting marks induced by striking a Zr-2 tipped pendulum inside a Zr-2 ring. Three of the samples examined showed no increase in hydride content at the fretted area from that at the non-fretted area. Four of the samples showed a factor of 1.5 to 3 increase in hydride content at the fretted area. Total concentrations were 100-250 ppm at the areas of increased hydriding.

A PRTR UO_2 fuel element with $\frac{1}{2}$-inch wide end bracket supports 60° out of alignment is being tested in TF-7 at 530 F in pH 10 (LiOH) flowing water. External vibration is being employed to induce fretting. After a one month exposure fretting was noted on one support on each end bracket; after two months it was noted on two supports on the top bracket and three supports on the bottom bracket. A 0.5-1 mil deep fretted pit was found in the pressure tube at the bottom bracket location after two months of exposure. A similar test with aligned end bracket pads showed no fretting under similar conditions.

<u>In-Reactor Tests</u>. Work on the PRTR inlet jumper vibration monitor continued. At present, two transducers are placed on Tube 1653. This particular tube was selected because the most recent inspection revealed a fretting penetration of 13 mils and a significant increase in the number of contact marks. The second transducer, which had not been previously subjected to a radiation environment, was placed below the first transducer (originally on Tube 1348) to

determine possible effects of radiation environment on the vibration measurements. There does not appear to be a transient or short term effect; however, the long term effect may be significant. Transducer No. 2 was new prior to this reactor installation; therefore, its indication would have been more nearly correct during the first few hours presumably. Apparently the PRTR environment reduces the output signal some 60 to 70%. The two transducers may not put out the same signal in any case because one is located two inches below the first.

An amplifier-recorder system for three additional channels of vibration measurement has been constructed and is being calibrated at this time. The system will be installed in the PRTR at the next outage.

The inlet line to the Magnedash autoclave in the PRTR is being relocated to keep particulate material in the reactor from plugging the autoclave.

Design Analysis

Nuclear Safety of Mixed PuO_2-UO_2 Fuels. Analysis of the prompt kinetic response of an EBWR-type reactor with mixed oxide fuel in which internal heat transfer to the uranium is delayed has been completed. Analog computer calculations were carried out and compared with an approximate analytic calculation. The time delay, expressed in terms of the time constant for heat transfer from PuO_2 to UO_2 within the fuel was varied from 0.001 to 1.0 second and the initial prompt period from 0.013 to 0.005 second.

The analog data obtained indicate that the energy release in a prompt excursion, terminated by Doppler feedback, is relatively insensitive to the time constant up to a value nearly five times the initial prompt period. This is due to the higher PuO_2 temperature achieved which tend to maintain a high heat transfer rate despite the increased time constant.

Comparison of analog results with the analytic model was carried out for an assumed exponential burst shape in the latter. It was found that this approach yielded much lower energy releases than the analog calculation but showed a stronger dependence upon the time constant. It is concluded that the exponential burst assumption strongly overestimates the Doppler feedback and therefore should be modified to a two-term or clipped exponential to yield better results. This work is reported in HW-80255, "Self-Limiting Prompt Excursions with a Time Delay in the Doppler Effect," by V. W. Gustafson and R. E. Peterson.

Fuel Re-use. Detail design on the liquid sodium loop for use in making engineering evaluations of the fuel re-use concept is about 40% complete. The long delivery items such as the oxygen control and indicating system, the electromagnetic pump and the electromagnetic flowmeter are on order for delivery in July when construction is scheduled to start.

Preparation of the final report on the present fuel re-use study on four thermal reactor types was begun. The fuel re-use concept was found to be economically advantageous in all cases, exhibiting a small advantage in the high neutron economy reactors and a greater advantage in reactors having higher parasitic absorption. The results will be presented as values of "fuel re-use" rods (dollars per pound) which will permit the thermal reactor operator to achieve the same minimum fuel cycle costs as if using slightly enriched uranium fuel at some given economic conditions. A sample range of values, as determined by this study, are from $34 per pound in a heavy water moderated reactor using Zircaloy-clad fuel, to over $100 per pound in a pressurized water reactor using a stainless steel clad fuel. These calculations are based on a 4.75% interest rate on the fuel charge and at a fuel cost corresponding to a $30 per pound fabrication charge for slightly enriched uranium fuel.

Thermal Hydraulic Studies

New Fuel Designs. Calculations were performed to determine the heat transfer characteristics of two candidate inverted cluster fuel elements (Mk IVA and Mk IVB) to be charged in the PRTR rupture loop. Both designs employed seven symmetrically placed round coolant channels in the fuel element cross section for cooling in addition to the heat transfer surface at the OD of the element. The Mk IVA element design utilized additional cylindrical fuel pieces within the seven interior flow passages to increase fuel cross section area. In the initial proposed tests these fuel pieces in the center of the coolant holes will not contain fuel and will not generate appreciable heat. At a tube power of 830 kw, it was predicted that the heat transfer conditions for the Mk IVA would be satisfactory and the maximum fuel temperature would be 1270 C. However, it was found that the Mk IVB would have to be run at a lower power to avoid excessive fuel temperatures and premature boiling in several coolant channels.

PRTR STudies. A document describing a simplified analysis for relief valve sizing has been completed and will be issued as HW-81380, "Analysis of Pressure Transients in the Plutonium Recycle Test Reactor for Sizing of Relief Valves." The analysis is concerned with the primary system pressure transient following a loss of pumping power with a reactor scram at the high-high pressure trip. The results show that the present relief capacity is adequate to reactor powers of 120 Mw. The effects of reactor inlet temperature, initial pressurizer gas volume, scram delay, and bubble lifetime are considered.

2. Plutonium Ceramic Fuels Research

Plutonium Ceramics Irradiations. Samples of irradiated PuO_2 (22×10^{20} f/cm^3) and PuN (20×10^{20} f/cm^3) have been examined in Radiometallurgy. The PuN did not appear badly cracked but did appear to have reacted slightly with the aluminum heat sink. The microstructure of the PuO_2 exhibited greatly increased porosity, while that of the PuN was practically indistinguishable from the pre-irradiation microstructure. X-ray diffraction patterns for both specimens showed very diffuse lines, indicating severe structural damage.

PuO_2 Cermet. A steel-clad nichrome - 35.5 wt% PuO_2 cermet in the shape of a 2" ID x $2\frac{1}{2}$" long hollow cylinder was fabricated. The 0.030" fuel matrix was clad by about 0.100" of mild steel on each side and the cylinder approximates fuel elements contemplated for the 630A reactor.

Thermal Cycling of W-β-Pu_2O_3 Cermets. Investigations are in progress to determine the behavior of a W-20 vol% β-Pu_2O_3 cermet during thermal cycling.

β-Pu_2O_3 Compatibility. The upper temperature limit of compatibility between thorium metal and β-Pu_2O_3 powders was determined to be less than 500 C.

3. Ceramic (Uranium) Fuels Research

Direct Induction Coupling to UO_2. UO_2 pellets were melted by direct induction heating. Direct induction coupling to the UO_2 was accomplished when the preheated pellet reached about 1200 C. The pellets were maintained entirely molten or with a solid surface and a molten core by controlling the input power.

Materials and Information Exchange. Twenty UO_2 single crystal specimens (0.250" diameter x 0.031" thick) were prepared, characterized, and sent to State University of Iowa for use in a solid state emf study. In addition, four UO_2 single crystals (0.250" diameter x 0.375" long) were prepared, characterized, and sent to Northwestern University for use in ultrasonic attenuation experiments.

Microhardness of Uranium Mononitride. The micro-identation hardness of single crystal uranium mononitride was determined as a function of crystal orientation. Knoop hardness of the (100) plane of UN indicated four maxima per 360° rotation of the indenter. The average maximum and minimum hardness (at 100-gram load) was 440 and 600 KHN.

Diametral Compression Testing of Cermets. Diametral compression tests have been carried out on W-UO_2 cermets containing 1, 2, 5, 10, and 50 vol% UO_2. The effect of UO_2 particle orientation on tensile strength was relatively large for cermets with less than 5 vol% UO_2. Small variations in density have a large effect for cermets containing less than 10 vol% UO_2.

Stainless Steel-UO_2 Cermet Fuel. Thin wall, stainless steel-clad, 15 vol% UO_2-stainless steel cermet core fuel pins were fabricated by pneumatic impaction with subsequent swaging to uniform size. Powdered and solid stainless steel end caps were used in the "one-shot" Nupac process, using 0.006-inch, 0.010-inch, and 0.035-inch wall thickness stainless steel cladding. Excellent bonding was achieved between the cap, can, and cladding.

Beryllium Welding. Modifications to the magnetic force welder to include integral preheating and increase magnet current have been completed. Welding capability has improved and crack-free welds in CEA (French) beryllium have been obtained.

Irradiation of Uranium Mononitride. Two low density (70% TD) uranium mononitride fuel capsules were fabricated for irradiation at a surface heat flux of 656,000 Btu/(hr)$(ft)^2$.

UO_2 Irradiation Studies. Four tungsten clad UO_2 capsules were successfully irradiated at cladding surface temperatures greater than 2700 C to 5×10^{19} fissions/cc (1600 Mwd/ton). A 0.051 cm (0.020-inch) diametral gap was left between fuel and cladding to accommodate the 25% fuel volume increase between room temperature and 2800 C.

Thoria Irradiation Studies. A central fuel temperature of 480 C (900 F) was measured in an instrumented ThO_2 fuel element being irradiated in the GEH-4 facility (MTR). Metallographic examination of aluminum cladding from four irradiated thoria elements showed no internal corrosion.

BeO-PuO_2 Impaction Tests. Ten grams of PuO_2 (60-100 mesh particles) are being mixed with BeO for impaction studies. The nickel provides a shield for alpha radiation, thus simplifying the handling of the BeO-PuO_2 mixture.

4. Basic Swelling Studies

Irradiation Program. Irradiation capsule No. 32, which contains high purity uranium specimens of tubular and rod geometry of varying diameter and section size, was subjected to a final bench test preliminary to irradiation.

Irradiation capsules Nos. 30 and 31, which contain enriched uranium specimens, are currently being modified to increase their heat transfer capabilities.

A Marotta pressure regulated control valve, which will be used to control the helium pressure on the high pressure irradiation capsule, is currently undergoing bench testing in order to determine its long term stability. On attempting to raise the set pressure to the 70 kg/cm^2 (1000 psi) range, a 112 kg/cm^2 (1600 psi) rupture disc in the line blew out well below its rating, possibly as a result of tightening the holddown washer too tightly, causing it to partially cut through the disc.

Second Phase Particles in Dilute Uranium Alloys. Evaluation continues on the quantitative metallographic data obtained on the distribution of intermetallic particles in dilute U-Fe-Si alloys. Initially, the precipitated intermetallic particles were counted and classified by size with the Zeiss Particle Size Analyzer. These data were then used to calculate the average volume fraction and total dispersed particle density. Also, an attempt was made to estimate the three dimensional particle size distribution. Alternative methods of calculation are under consideration.

Restrained Irradiations. 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.

Electron microscope examination has been completed on replicas from two U-2 wt% Zr fuel samples irradiated with center fuel temperatures in the beta + gamma$_1$ region (~ 720 C - 1328 F) to an exposure of approximately 0.01 at% B.U. Despite the low exposure, fission gas pores, less than 0.5μ in diameter, can be seen in electron photomicrographs of the center of the fuel. The pores are found predominately at the interface between the islands of retained gamma$_1$ second phase and the former beta, now alpha, matrix. The collection of pores at these interfaces has been observed on other U-2 wt% Zr fuel samples irradiated at similar temperatures but to much higher burnups, approximately 0.2 at%. In the higher exposure samples, however, those pores at the $gamma_1$-matrix interface were an order of magnitude larger, 2μ, and the matrix between the islands of second phase $gamma_1$ contained a large number of fission gas pores as well. In order for pores to form in these samples during this short low exposure irradiation, the fission gas has either diffused very rapidly to sites at the interface or has been collected by moving phase boundaries.

5. Irradiation Damage to Reactor Metals

Alloy Selection

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. Oxidation tests were completed by Chemical Metallurgy personnel on specimens exposed to 15 Torr water vapor in helium at 1038 C (1900 F). Specimens were run in this environment for 100, 200, and 300 hours. Similar specimens are also being tested in 650 C (1200 F) steam at 3000 psi. These specimens, examined after 336 and 672 hours, showed little effect of the exposure. The test is now continuing to a goal of 1000 hours.

Tensile specimens of TZM, Cb-752, Cb, Ta, Ta-10 W, and Cb-1 Zr are being exposed to atmospheric air and dry air for 14 days at 260 C (500 F) to determine if a significant difference in mechanical properties results. These tests will help develop procedures for handling refractory metal alloys irradiated in the ATR gas loop facility.

In-Reactor Measurement of Mechanical Properties

Two creep capsules containing annealed 304 stainless steel have been placed in the reactor. The capsules are operating at 500 C and 550 C (932 F and 1022 F). During reactor operation about one-half of the power required to maintain temperature is supplied electrically, the remainder being supplied by gamma heating. During reactor outages temperature can easily be maintained by electrical heating alone. Temperature control of these capsules appears to be superior in all respects to the Zircaloy-2 test capsules. Significant creep rates have not yet been established.

Irradiation Effects in Structural Materials

Inconel X-750 specimens in two test heat treatments have been irradiated in the G-6 position of the ETR to exposures of 1.4×10^{20} nvt (fast). Specimens in a third test heat treatment are currently being irradiated. Metallographic examinations of unirradiated heat treated wafers have shown that high temperature aging produces heavier grain boundary precipitation and coarser in-grain (gamma prime) precipitate than does the lower temperature aging treatment. The lowest temperature aging treatment [1000 F (538 C)/50 hrs] gives a slightly lower DPH value than do the higher temperature ages [1200 F (649 C)/50 hrs, 1550 F (843 C)/24 hrs, 1300 F (704 C)/20 hrs], namely 328 as compared with 359 and 351, respectively.

Hastelloy X-280 specimens that were solution-treated and aged at temperatures from 1000-1700 F (538-927 C) were irradiated at 540 F (282 C) in the G-7 loop of the ETR to exposures of 2.75×10^{20} nvt (fast) and tested at room temperatures. Tensile strength and reduction of area were higher but followed the same trend as those of controls. There was little or no change in ductility. Yield strengths for specimens solution-treated, or solution-treated and aged at 1000 F (538 C) were higher and followed the same trend as for controls. Specimens aged above 1000 F (538 C) showed a lesser increase in yield strength due to irradiation than specimens aged at or below 1000 F (538 C).

The furnace and associated control system for tensile and stress-to-rupture testing at 700 C (1292 F) have successfully undergone initial tests. Modifications in the design of existing specimen grips for use at 700 C (1292 F) in air or argon have been made.

The low cycle fatigue apparatus was tested. The original specimen design proved inadequate for the desired high strain tests. Buckling at low strains constituted a major limitation. A new specimen has been designed to withstand 35% strain without buckling.

An improved NaK capsule design (GEH-22-2) was sent to Phillips Petroleum for comments. It is proposed to irradiate a number of high temperature alloys in this capsule at controlled temperatures.

Tests were performed to determine the effect of irradiation on the fracture behavior in notch bending of annealed Zircaloy-2. Twelve notch bend specimens were tested after irradiation to an average exposure of 2.8×10^{20} nvt. It was found that irradiation caused only a mild embrittlement. However, it was observed that the effect of increasing loading rate (strain rate) on the ductility was more adverse in the irradiated condition.

Damage Mechanisms

The objective of this program is to determine the mechanism by which defects produced during neutron bombardment interact with dislocations to modify the plastic deformation characteristics of the metal. The investigation is currently concerned with the role of interstitial impurities in alpha-iron.

Deformation twins were observed in several electron transmission microscopy foils irradiated to 1×10^{19} nvt and strained a few percent in tension. This is the first time twins have been observed in an iron foil deformed at room temperature and the observation is believed to be an indication of the change in the mode of plastic deformation produced by the irradiation.

Additional work has been done on developing a theory of strain rate sensitivity which is consistent with a thermally activated mechanism of plastic deformation.

Environmental Effects

Experiments designed to determine the oxidation resistance of N-Reactor process tube material and the effect of various heat treatments on the behavior of reactor grade Zircaloy-2 during irradiation were partially completed during the report period. Specimens were exposed to the in-core environment of the G-7 loop at 540 F (282 C) for 49 days. Estimated neutron flux intensity and integrated flux were 5×10^{13} nv and 2.1×10^{20} nvt (> 1 Mev).

Weight gains continue to show that Zr-2 oxidation rates in the G-7 loop are greater than for nonirradiated exposure by at least an order of magnitude except for two specimens of N-Reactor material which were rolled to a total of 86% cold work. Weight gains for these specimens were one-fourth to one-half the values for similar but less heavily cold worked material exposed in the preautoclaved and as-etched condition, respectively. Heretofore, no significant effect of cold work on corrosion has been noted either in- or out-of-reactor.

Material heat treated above the $\alpha/\alpha + \beta$ transition and furnace cooled and alpha annealed oxidized at rates higher by a factor of two or more than material which has not been so treated. Normal in-reactor oxidation rates were restored to this corrosion-prone material by heat treatment in the beta region, followed by an air quench prior to the final alpha anneal. These results have confirmed out-of-reactor findings at Westinghouse.

Hydrogen Addition to ETR G-7 Loop Coolant

Operation of the G-7 loop for one or possibly two ETR cycles with hydrogen addition to loop coolant for the purpose of reducing oxygen levels is essential to the evaluation of the effects of fast neutron irradiation on Zr-2. Preliminary investigation has indicated that hydrogen could be added to the G-7 loop water with minor modifications of existing equipment.

ATR Gas Loop Studies

Gas Chromatograph. The silica gel column for CO_2 analysis has been installed and flow and temperature parameters were readjusted to permit analysis of H_2, O_2, N_2, CH_4, CO, and CO_2 on a single sample. Sensitivities are better than 1 ppm for these impurities. The valve leakage problem reported last month has been minimized so that leakage is now equivalent to 0.2-1.0 ppm of N_2.

Corrosion of ATR Structural Materials. The corrosion resistance of Haynes 25 and Hastelloy X has been determined in atmospheres of CO_2 and water vapor at 1200 C (2192 F). Haynes 25 exhibits much better corrosion resistance in 100 mm Hg CO_2 pressure than in an equivalent oxygen pressure: weight gains after 100 min are 0.6 and 10.2 mg/cm^2, respectively. After 100 min in H_2O vapor at 1200 C, the weight gain is 1.8 mg/cm^2, somewhat higher than that obtained in CO_2. Hastelloy X shows similar oxidation behavior in oxygen at 25 mm pressure, CO_2 at 100 mm Hg, and water vapor at 25 mm Hg, in tests of 8000-minute duration.

6. Gas-Cooled Reactor Studies

EGCR Graphite Irradiation. The seventh capsule, H-3-7, in the series of long term irradiations of EGCR graphite was removed from the GETR after successful completion of four reactor cycles. All 24 samples were recovered intact. Cracks are now in evidence in a few of the samples. They appear as fine cracks located randomly, long circumferential cracks, and radial cracks. Some are on samples having high exposures and some on samples with lower exposure. They appear on all the graphite types being irradiated. The ones on the higher exposure samples do not appear to have increased in size during the last two irradiation periods.

All 65 flux monitors were recovered and have been returned to Hanford with the samples. Sample length data, monitor activation rates, and temperature charts are being analyzed.

Irradiation of Graphite Cloth. Graphite cloth retained the properties which make it an interesting high temperature material after exposure to 6750 Mwd/At$_K$ at 600 to 650 C in a Hanford test facility. A sample of National Carbon Company Grade WCA graphite cloth was wrapped around several cylindrical nuclear graphite specimens during irradiation. A set in the cloth after irradiation reflected the shape of the solid samples; however, flexibility of the cloth seems unchanged. Properties of the irradiated cloth and a remnant of the piece from which it was cut, were measured by National Carbon Company. With a limited amount of material, it was difficult to detect significant changes in weight, thickness, thread count and electrical resistance. However, breaking strength increased about a factor of 2, an improvement analogous to that observed in bulk nuclear graphite.

7. Graphite Radiation Damage Studies

Irradiation of Lampblack Carbon. Samples of carbons made from lampblack and heated to 1400 C have been irradiated for a total of 3.1×10^{21} nvt ($E > 0.18$ Mev) in ETR noninstrumented capsules during four successive exposures at temperatures near 900 C. During each irradiation the contraction rate has been decreasing. The saturation curve fits an equation of the form

$$\Delta L/L \text{ (percent)} = -4.17 \, [1 - \exp(0.0713 \, \phi t)]$$

where ϕt is in units 10^{20} neutrons/cm^2. This curve fits all previously measured points including the newest point at $\Delta L = 3.7\%$. The expression predicts a maximum linear contraction of 4.2% for this material.

Irradiations of the same material are also being conducted at about 650 C at Hanford. The exposure attained to date is still too low (6×10^{20} nvt, $E > 0.18$ Mev) to define a saturation value. However, if the saturation value of the above equation also applies to these irradiations, the data fit the following equation:

$$\Delta L/L \text{ (percent)} = -4.17 \, [1 - \exp(-0.107 \, \phi t)].$$

8. Boronated Graphite Studies

Boronated Graphite Irradiations. The first phase of the long-term irradiation was successfully concluded. The capsule performed well throughout the entire irradiation period and the temperatures of the samples were within the pre-irradiated specification of ± 150 F. The average temperatures of the two 1000 F (nominal) sections were 900 and 885 F, respectively, and the average temperature of the 650 F (nominal) section was 700 F. The slight increase of temperatures as a function of exposure became negligible during the last irradiation period.

With the exception of the six ANL samples, no samples showed any visual effect of oxidation. The ANL samples were considerably less pure and some pitting oxidation was observed. The pitting was typical of a metallic impurity catalyzed reaction. Analysis of gas from the capsule at the termination of the irradiation indicated a high purity helium atmosphere. No detectable amount ($< 0.01\%$) of H_2O, H_2, HD, D_2, HT, or T_2 was found. Minor amounts of CO_2 (0.02%), O_2 (0.02%), and N_2 (0.27%) were detected.

Physical property measurements on the samples are still in progress and the results will be reported next period; however, certain preliminary observations can be reported now. With the exception of the 7 wt% grey material, the samples contracted in the parallel direction in excess of expectation for similar but nonboronated graphite. The 7 wt% grey material expanded slightly. No large effects of different irradiation temperatures or different boron concentrations were observed. For the transverse samples the length changes were small and variable, but generally positive. Again, no large differences were obtained between samples irradiated at the two temperatures or two boron contents.

Twenty-two of the 42 samples in the first irradiation were installed in a new capsule for additional exposure. This work was concluded without incident despite the surface activity of some samples exceeding 15 rem/hr. The new capsule contained two samples (7 wt% black; 7 wt% grey) which had been replicated for electron microscopic examination. The remainder of the new samples had flat ends so that pre- and post-irradiation changes in length across the ends on a diameter could be made. The use of flat, rather than spherical ends, reduces the accuracy of measurement due to slight nonparallelism but was considered desirable to show possible effects of thermal neutron gradients in the samples. The new capsule was installed in the reactor and all tests indicate a leak-tight assembly.

ETR Irradiation. Application of boronated graphite in the Enrico Fermi fast breeder reactor will result in graphite damage from a neutron spectrum unfamiliar to available testing facilities. Characterization of damage from fast neutrons ($E > 0.18$ Mev) and that from the n,α reaction is desirable to predict the operational problems and conditions that will exist in the Fermi reactor. Irradiations are planned that will determine damage and damage rates in environmental neutron fluxes of several fast-to-thermal ratios. These data will be analyzed to predict the separate effects from the two damage mechanisms.

The relatively hard neutron spectrum of the ETR will initially be used to obtain gross damage information at approximate Fermi thermal and fast flux exposures. Twenty samples in seven capsules are planned, with initial charging of a portion of these to begin during the April outage. Additional irradiations are planned in which the spectra incident on the samples will be tailored by shielding of the thermal neutrons and by increasing the proportion of fast neutrons.

Neutron Calculations. Calculations are being made to determine the feasibility of irradiating boronated graphite inside neutron-converter fuel elements. Calculations made to date have been concerned with boronated graphite irradiations in a Hanford irradiation facility. For this case it is desirable to increase the fast, or damaging flux, ($E > 0.18$ Mev), and to decrease the thermal flux causing the B^{10} (n,α) Li^{6} reaction. A Pu-Al element was chosen for this study because the high thermal conductivity of the Pu-Al permits use at a higher heat flux than would otherwise be possible. Results from these calculations indicate that an increase in fast-to-thermal ratio of 20 is possible.

9. Aluminum Corrosion and Alloy Development

Tests in C-1 Loop. A charge of aluminum-clad fuel elements undergoing corrosion test in C-1 Loop has reached the desired exposure (28 days at temperature to date) and will be discharged at the next outage. These elements are being exposed at 260 C outlet water temperature and neutral pH. This is the first test charge in C-1 to accumulate sufficient exposure to obtain meaningful corrosion measurements. The cladding temperature of the upstream fuel element, as measured by a thermocouple, rose steadily during exposure at an average rate of about 3 F/day at a heat flux of 10^5 Btu/hr-ft^2. On two occasions when the reactor was shut down and started up again, the cladding temperature (after full reactor power was restored) returned to near its initial value and then rose again. The temperature rise is believed to be due largely to crud deposition on the fuel element (although it may be due in part to oxide buildup) which is sloughed off as a result of thermal shock on shutdown. Crud content of coolant samples and a reduction in pressure drop across the test section confirmed that crud and/or oxide had been sloughed off when the temperature was reduced.

10. 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 1.9 x 10^{20} fission/cm^3 (5450 Mwd/ton). The fuel elements are operating at a maximum temperature of 515 C, a surface heat flux of 62 cal/sec-cm^2 (8.3 x 10^5 Btu/hr-ft^2) and a specific power of 57 watts/gm (174 kw/ft). Weight measurements made at an integrated exposure of 1.9 x 10^{20} fissions/cm^3 indicated a maximum volume increase of 0.8%. The previous maximum fuel volume increase was 0.5% at 1.3 x 10^{20} fissions/cm^3 (3600 Mwd/ton).

Thorium Defect Testing. Metallography of cross sections through the defects of ruptured fuel specimens show that the rupture mechanics for the heat treated specimens are similar to those previously reported for as-extruded specimens. The thickened diffusion zone between the Zircaloy-2 clad and the Th - 2.5 wt% U - 1 wt% Zr core produced by vacuum heat treatments of 750 C for one hour and 800 C for three hours were still intact. Tearing of the cladding from the core by internal forces produced by corrosion product buildup was propagating through the core material inward of the diffusion zone. The diffusion zone exhibits more corrosion resistance than does the core material. The bond behavior of the specimens that exhibited longitudinal splitting was the same as that of specimens that formed a localized blister. The cladding on the as-extruded specimens that exhibited splitting behavior was very fine grained and showed very little thinning in the defect area whereas the larger grained cladding of the heat treated specimen underwent considerable thinning in the blister area.

The annealed structure of the Zr-2 clad resulting from heat treatment of the core material has increased ductility and lower strength which permits the improved blister-type rupture.

Thorium-Uranium Alloys. The fabrication, irradiation, and defect corrosion behavior of the Th - 2.5 wt% U - 1.0 wt% Zr alloy has been described. The effects of higher zirconium and uranium compositions on the structure, fabrication, and defect corrosion behavior of thorium base alloy fuels have not been thoroughly studied. A program has been initiated to study these effects.

Nine additional alloys have been double vacuum arc melted and coextruded to 0.525-inch diameter x 72 inches long rods with 0.025-inch Zircaloy-2 cladding. The copper canned billets were preheated to 760 C and extruded on the 700-ton Bliss press. The extrusion constants ranged from 17.2 to 20.2 tons/$inch^2$. Samples have been cut for extrusion evaluation and the balance of the stock will be used for defect corrosion testing. Cursory examination of the as-extruded rods and cut samples indicate that coextrusion was successful on all ten compositions.

Irradiation of Fine Carbide Uranium Fuel. Zircaloy-2 clad rods containing a submicron dispersion of uranium carbide are being irradiated in the ETR to investigate the ability of finely dispersed uranium carbide to control swelling of metallic uranium. The first NaK capsule test is currently undergoing its third cycle of irradiation. To date, the fuel samples in this capsule have accumulated a burnup of 0.15 at% (one-half of planned goal exposure) at a maximum fuel temperature of 550 C. Two additional capsules are scheduled for insertion next month. Goal exposures for these capsules will be based on observed performance of the fuel rods now under irradiation.

Tube and Wire Fabrication. Development work in tube and wire fabrication led to the filing of two invention reports covering unique fabrication equipment. Fabrication of small tubing is often accomplished by swaging the tube over a hardened steel mandrel. A swaging machine has now been developed which will slightly increase the diameter of the tubing in a single pass and thus permit easy removal of the mandrel. This not only reduces fabrication time but permits reuse of mandrel stock. In the fabrication of wires, gripping the wire for drawing or testing purposes without physical damage to the wire has been difficult. A ball gripping device, easily tightened or removed from the wire stock, has made drawing of long wire lengths far simpler.

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

Thermal Hydraulic Studies

Fog Cooling of 19-Rod Fuel Elements. A long, electrically heated, 19-rod bundle was designed for the fog-cooled heat transfer studies. It will have a heated length of 76 inches and use ceramic warts to provide a rod spacing of 0.074-inch.

One of the basic criteria for effective heat transfer in the fog-cooled reactor concept is the ability to distribute the two-phase coolant so that the liquid portion will contact the heating surfaces. In the present study two proposed distribution techniques will be examined for their influence on heat transfer. One will be the introduction of the two-phase coolant to the test section through the use of a nozzle (supplied by AECL) which will discharge a high quality steam annulus surrounding a slightly subcooled water core. This places the liquid phase in the inner portions of the bundle where it will be most effective.

To permit a prototypical approach of the two-phase mixture into the test section, the test section current will be introduced radially. Copper webs, 0.10-inch wide and 5 inches long, will interconnect the copper end rods of the bundle. The outer 12 rods will be connected by webs to a nickel sleeve with an ID equal to the ID of flow housing. Special electrical heads will be clamped to the OD of the sleeve and ceramic rings plus O-rings will seal and insulate the sleeve from the pressure tube.

A second technique to be studied will be the use of rougheners along the pressure tube wall in the form of circumferential ridges with a 0.050- by 0.050-inch cross section spaced 3.25 inches along the length of the tube. These rougheners are intended to redistribute the water which may accommodate along the walls of the pressure tube. To accommodate these rougheners in the test section, an oversized pressure tube will be used which will permit the insertion of ceramic sleeves along with the bundle. The junction of each sleeve will be recessed to hold a stainless steel ring which will act as the roughener.

12. Advanced Reactor Concept Studies

Fast Supercritical Pressure Power Reactor. The document reporting the economic evaluation of the 300 Mw(e) Fast Supercritical Pressure Power Reactor is being printed. Notice was received that a patent application will be filed by the AEC on HW-IR-1685 - Nuclear Reactor Fuel Element. This is the second patent application to be filed on the FSPPR.

Segmented Fast Reactor (SFR). The analysis of SFR Cores I and II pointed out two major temperature problem areas: (1) clad temperatures of 1717 F and 1757 F, and (2) maximum fuel temperatures of 7090 F and 6116 F. Reduction of the maximum fuel temperature does not present as much of a problem as does the clad temperature. The clad temperature is determined primarily by the coolant temperature, which for the cores analyzed was as high as 1696 F at the hot channel outlet. To reduce the maximum coolant temperature, some changes in core power density or in the primary coolant cycle are necessary.

A survey of current stainless steel technology indicates that a maximum clad OD temperature of 1400 F appears to be reasonable for preliminary design purposes. Using this as a design criterion, a short parametric study was made to determine possible gross core properties.

Study of typical design values for coolant temperature rise Hot Channel Factors used in fast reactor designs resulted in a choice of a factor of 1.55 for preliminary design of the SFR. Using this hot channel factor ($F^{E}_{\Delta t\ coolant} = 1.55$), a maximum coolant outlet temperature of 1350 F and bulk coolant outlet temperature of 1200 F, the average coolant temperature rise must be limited to 273 F instead of the 300 F initially assumed. Hence, the reactor inlet temperature must be 927 F and the total core coolant flow $W_{tc} = 97.2 \times 10^{6}$ lbs/hr.

Two means of decreasing the coolant temperature rise due to the radial nuclear flux peaking were assumed. The flux peaking factor indicated from the analysis of Cores I and II was $F^{N}_{R} = 1.3$. In one case it was assumed the flow in the reactor was orificed such that the coolant outlet temperature remained 1200 F in all channels. In the second case it was assumed that by zone enrichment a value $F^{N}_{R} = 1.1$ could be obtained and the flow orificed such that the coolant temperature from all channels was again maintained at 1200 F.

Other parameters and their range of values used are:

1.	Power Density	300-500 Kw/l
2.	Coolant fraction	0.3-0.6
3.	Number of fuel assemblies per core	426
4.	Number of fuel pins per assembly	127-331
5.	Fuel pin diameter	0.200"-0.250"
6.	Fuel clad thickness	0.015"
7.	Fuel thermal conductivity	1.6 B/hr-ft-°F
8.	Fuel surface temperature (max)	1650 F
9.	Total power from core and axial blkts.	1960 Mwt
10.	Maximum coolant vel.	35 fps.

Curves were plotted from the calculations made in the survey to aid in selection of new cores for study.

The pressure drops for these cores are not yet calculated. It was assumed that if the maximum coolant velocity were kept at 35 fps the pressure drops would be approximately the same as for Cores I and II, i.e., 100-150 psi.

This parametric study will be used to establish core design values for Core III and further cores to be studied.

Plutonium-Fueled Fast Compact Reactor Studies. A formal report is in preparation summarizing work done to date on compact plutonium-fueled cores.

13. Critical Flow Studies

The document HW-80535 RD, "Steam-Water Critical Flow from High Pressure Systems," was issued. This document reported the results of the preliminary phases of the high pressure and temperature critical flow program. In this preliminary work the critical discharge of steam-water mixtures from a short piece of pipe (0.505-inch ID and L/D = 20) was studied at upstream pressures approaching 2000 psia. The pipe length had a 20° conical approach section. Inlet enthalpies of approximately 425, 460, 485, and 530 Btu/lb were considered. When a two-phase mixture entered the test section, flow-choking conditions were observed to occur at the end of the test section, and the data demonstrated good correspondence with the predictions of the advanced theory. When compressed water entered the pipe and was required to flash to a steam-water mixture in the pipe or in the conical approach section, the location of the primary choking was observed at the entrance rather than at the exit where it is normally assumed to exist. Flow rates were considerably greater than predicted by theory based upon corresponding exit conditions.

14. Upstream Boiling Burnout Studies

Data were analyzed from two test sections in which upstream boiling burnout was detected at low pressures. The first test section was a 0.444-inch ID by 60-inch long Inconel tube with a 24-inch long uniformly heated section. Burnout runs were conducted at 145 psia outlet pressure. Of five runs with $G = 1 \times 10^6$ lb/hr-sq ft, two resulted in upstream burnout when the inlet subcooling was less than 30 Btu/lb. Of 12 runs with $G = 1.5$ to 2.0×10^6 lb/hr-ft^2, there were no upstream burnout runs. However, these runs did not include runs with inlet subcooling less than 26 Btu/lb. The run with inlet subcooling = 26 Btu/lb and $G = 1.46 \times 10^6$ lb/hr-ft^2 did result in almost simultaneous burnout indications along the full length of the test section.

This behavior of upstream burnout occurring when inlet temperatures are near saturation is typical of the results obtained in the high pressure upstream burnout studies.

The second heater rod was quite similar to the first one except that circumferential slots were machined in the outside surface of the rod to produce localized areas of high heat flux. There were 23 slots of 0.10-inch width and one-inch center-to-center spacing. The high heat flux regions (slots) had heat flux values which compare to the rest of the rod as peak-to-average = 3.56, and peak-to-minimum = 4.88.

Burnout tests were conducted with inlet coolant conditions ranging from 26 to 123 F subcooling at flow rates of 1.0 to 2.5×10^6 lb/hr-ft^2, and resulted in upstream burnout in every one of the 15 runs. Upstream burnout always occurred at locations of bulk subcooling, even though outlet steam qualities of greater than 13% were achieved. The burnout location varied from within two inches of the inlet of the heat section to within three inches of the outlet of the heated section and always occurred in the slots, i.e., the high heat flux areas. Upstream burnout could not be attributed to variations of slot heat flux from one slot to another because the heater rod was changed end for end, and two runs were repeated with the results that burnout occurred at identical coolant conditions and distances from the inlet end.

Preliminary designs have been made for three different test sections. The test sections would allow comparison of burnout (upstream) in thick-wall versus thin-wall test sections (wall thickness ratio $\gtrsim$ 6.1), in small diameter versus large diameter test sections ($\gtrsim$ 0.24-inch versus $\approx$ 0.44-inch), and in test sections having different inlet and outlet hydraulic calming lengths.

D. DIVISION OF RESEARCH - 05 PROGRAM

1. Radiation Effects on Metals

Wafers have been cut from irradiated and fractured molybdenum single crystals containing 20 and 450 ppm carbon and are being examined by transmission electron microscopy. Both crystals were cut so that the surface of the wafer was parallel to (111). Some difference in the microstructure was expected, as the crystals containing low carbon were greatly reduced in cross section during deformation while crystals with higher carbon content failed after comparatively small deformation. Both specimens showed the presence of (1) defect clusters and (2) channels with traces parallel to (110) and (112). However, there is a marked difference in the number and width of the channels present in the two samples. The occurrence and distribution of channels is directly related to the extent of deformation in these crystals, as expected. The next set of crystals to be examined will be sectioned parallel to the (110) plane, which is presumed to be the primary slip plane.

X-ray lattice parameters have been determined for molybdenum single-crystal rods for which lengths had previously been measured by precision methods. Within experimental error, all crystals yielded the same value of a_c. No evidence of subgraining was observed; the previous annealing treatment presumably yielded single crystals essentially free of substructure. These crystals will be submitted for irradiation and will provide data on both length changes and lattice parameter changes as a function of irradiation dose.

Considerable effort has been expended toward a series of theoretical calculations on processes which occur during quenching in nickel foils. Starting with a known quenching temperature of 1675 K and an initial quenching rate of 3×10^3 °K per second, and with vacancy formation and migration enthalpies of 1.35 and 1.50 electronvolts, respectively, it has been possible to obtain an expression for the total number of jumps made by a vacancy during quenching. An approximate solution has been obtained with the aid of the 7090 computer. This approximation gives $n = 2 \times 10^7$ jumps. Assuming that the equilibrium vacancy concentration of 2.9×10^{-4} at 1675 K is retained during the quench, only 1.25×10^3 jumps are required of a vacancy before it encounters another vacancy. In nickel foils of 99.997, 99.97, and 99.4% purity, only 5×10^3, 0.9×10^3, and 1.4×10^2 jumps, respectively, are required for a vacancy to encounter an impurity atom (based on the oversimplifying assumption that all solute atoms are in solution and uniformly dispersed). For a dislocation density of 10^6 cm^{-2}, 6.7×10^3 jumps are required for a vacancy to reach a dislocation. Quenching with this particular set of experimental parameters should result in very few single vacancies, and clusters of vacancies associated with impurity atoms should predominate in the less pure materials. Ignoring a drift flow toward dislocations, the majority of vacancies are not expected to be annihilated at dislocations or subgrain boundaries.

2. Plutonium Physical Metallurgy

Experimental studies of the creep characteristics of the stable phases of high purity virgin plutonium were continued. Expressions relating the steady state creep rate with temperature and applied stress have been derived for the beta, gamma, and delta phases.

The electro-refined plutonium received from LASL has been evaluated metallographically. The metal consists of large grains, approximately 0.2 mm diameter, and contains a considerable number of microcracks. The microcracks are located primarily at the grain boundaries.

Refinements in electron and optical metallographic techniques are being investigated. Plutonium quenched from the beta phase to -80 C (-112 F) and -176 C (-234.8 F) should contain retained beta at room temperature. Unambiguous identification of retained beta by metallographic techniques was attempted but was not achieved.

Installation of the new glovebox is under way, and considerable effort has been devoted to the design and development of specialized apparatus which is to be installed in it. Effective temperature control has been demonstrated on a sample under conditions similar to those existing during cathodic etching. Plutonium etching has been handicapped because of the tendency for the sample to overheat and exceed the transformation temperature during the etching process. With this objection removed, a potentially valuable metallographic technique becomes available for use with plutonium.

E. CUSTOMER WORK

1. Radiometallurgy Laboratory

Examinations and Measurements

The results of routine examinations and measurements are, or will be, reported as part of the sponsoring research and development programs. During the period February 18, 1964 to March 27, 1964, Radiometallurgy Laboratory output included:

Metallography samples processed	116
Photomosaics	9
Autoradiographs	29
Replicas (electron microscopy)	5
Burnup dissolutions	25
Fission gas sample collection	8
Vacuum induction furnace runs	17
Decladding dissolutions	51
Tensile tests	106
Notch beam tests (cryogenic)	12
Rockwell hardness tests	54
Density determinations	45
X-ray analyses	6
Negatives processed	754
N-Reactor quench tests	1

Equipment

Annealing Furnace. The annealing furnace was received from the vendor after being modified and repaired. Functional testing will be started as soon as possible.

"E" Cell Metallography Facility. Installation of the metallograph blister on the east side of "E" cell was completed by construction forces. Fabrication of metallographic process equipment for installation in cell "E" continued in the shop.

Induction Heating Equipment of N-Reactor Fuel Elements. Installation of the induction heating and quenching equipment was completed and one outer N-Reactor element was successfully heated to 980 C and quenched. Additional heat-quenched tests are in progress.

Microhardness Blister for "I" Cell. The new microhardness tester was received and design of remote controls is in progress. Fabrication of a sample transfer conveyor and viewing window frame was completed. Modification of the microscope optical relay tube was completed and a platform was installed to support the blister shielding.

Metallograph Camera. Automated controls and an electric shutter were installed on the metallograph roll camera to reduce the operator fatigue and human error associated with the preparation of photomosaic reproductions of metallography samples.

Waste Handling and Disposal. Drawings were completed on a waste handling cask capable of handling six cans of waste during burial. An AR was written to obtain authorization for funds to purchase this cask.

Stereo Zoom Hot Cell Microscope. Design drawings were completed for a positioner to be used with the stereo microscope. Tests are in progress to determine the type and amount of light needed for remote viewing through the microscope.

High Temperature Tensile Testing Machine. The Baldwin tensile testing machine was removed from "H" cell to permit installation of the new high temperature tensile testing unit.

2. Metallography Laboratories

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

An attack-polish method of polishing ceramics and cermets has been used elsewhere with reasonable success and this method was adapted recently for use with Buehler Automet polishers located in the 306 Building. By using the Automet polishers with a slurry of gamma alumina ($<$ 0.1

micron in size) in a 30% hydrogen peroxide solution for rough and final polishing, it was possible to obtain very good results on samples of UO_2 powder in tungsten. The UO_2 particles were retained with excellent flatness and the tungsten matrix came off the final polish with a good etch at the grain boundaries. Other types of samples will be tried to see what ones could benefit from this procedure.

3. High Temperature Lattice Test Reactor

Compatibility of Structural Materials with the HTLTR Environment. The evaluation of structural materials for the HTLTR continued with the resumption of the 1200 C, 1000-hour test in which the candidate materials are exposed to a nitrogen gas-graphite environment. The two previous runs of this test were stopped before completion because of breakdowns caused by deterioration of test specimens or the test chamber. In the current run the materials known to be unstable have been eliminated and the test chamber is protected from external corrosion by control of the furnace atmosphere.

The 1-5/8" diameter heating element and the graphite core block surrounding the element were removed from the small mockup after 504 hours of operation above 900 C. Upon removing the brick it became evident a few bricks in the vicinity of the grounded end had been molten at some time during the run. Temperatures as high as 3400 F were probably attained in that the bricks are generally rated at 200 to 300 F below their actual melting points.

Visual inspection of the heating element revealed little damage had occurred. There was a small area which appeared to be oxidized near the middle of the rod, directly beneath one of the sight tubes. Also, the grounded end of the element was somewhat damaged. Apparently, the arcing mentioned last month was occurring at this location.

Considerable loss of graphite from the outside surface of the core block occurred at the hotter locations. A greenish film covered the surface at these locations. Samples of this greenish film; a white, glassy appearing deposit on the sliding graphite collet; and, small metallic appearing spheres located on top of the graphite core block were submitted for chemical analysis. Heating element No. 3, a solid $\frac{1}{2}$-inch OD x 39 inches long, has been installed in the small mockup. Samples of nickel A, Hastelloy B, Inconel 600, molybdenum, TD Nickel, B_4C and BN were placed in the mockup to determine their resistance to the atmosphere found in the small mockup. Also, a few graphite electrodes were placed throughout the brick in an attempt to obtain information concerning current leakage through the brick at elevated temperatures.

4. EBWR Fuel Elements

EBWR Plutonium Fuel Element Fabrication. EBWR fuel rod fabrication requirements for the PRCF were completed. To date, nearly 600 of these UO_2-PuO_2 rods have been fabricated by either the resonant plate or resonant beam vibratory compaction techniques. The resonant plate technique involves a shorter vibration period.

Ultrasonic inspection of the EBWR tubing proceeded relatively slowly (40-50 tubes/shift) because the high rejection rate necessitates detailed examination of the defects. Approximately 70% of the tubes were rejected for cracks and impressed metal inclusions.

Purification of Oil Contaminated Fuel Supplies. Recent tests have shown that vacuum (6×10^{-3} mm Hg) outgassing at 250 C for 8 to 12 hours is sufficient to remove oil contamination from a deep bed of UO_2-PuO_2 powders. Higher temperatures ($\sim$ 600 C), combined with vacuum, should be avoided due to the possible formation of carbon and hydrogen residues in fuel supplies. Improved quality control over crushing operations and recent modifications of the disc pulverizers have minimized the possibility of oil contamination of UO_2-PuO_2 fuel supplies.

5. NASA Fuel Development

NASA Cermet Work. Continued investigation of behavior during thermal cycling of UO_2-W cermets has led to a suggested mechanism responsible for the high UO_2 loss. The proposed mechanism involves initial high temperature dissociation of UO_2, forming uranium metal which moves rapidly down the tungsten grain boundaries. Reoxidation of the metal on subsequent cycling, and the difference in expansion coefficients of UO_2 and W, would cause a "wedging" of UO_2 upon cooling. Repeated reheating could allow UO_2 to vaporize further into reopened boundaries, and eventual free movement of UO_2 to the surface.

Cermet Fuel Fabrication. Tungsten clad, 80 vol% W-fully enriched UO_2 cermet plates were pneumatically impacted. The 7/8-inch by 1-inch rectangular cermet plates had a thin, uniform powder clad of less than 0.005-inch. The edges of the plate were clad with 1/16-inch of tungsten. One hundred percent bonding was obtained between the cladding and cermet. The plate will be irradiated at high temperatures.

Tungsten clad, 80 vol% W-UO_2 cermet plates were pneumatically impacted. A thin, uniform tungsten powder clad of less than 0.005 inch was achieved. The tungsten used in both the cladding and cermet had a particle size of less than 1 micron. The UO_2 was an ultrafine powder obtained from France. The plates, having 100% bonding between cladding and cermet, will be used for basic studies concerning UO_2 loss upon thermal cycling.

Ultrasonic Drilling. A 61-hole hexagonal grid was fabricated using an ultrasonic drilling technique on a ½-inch thick, 50 vol% W-UO_2 trial specimen.

Thermal Stability of Bismuth Trioxide. Bismuth trioxide, Bi_2O_3, is perhaps a good material from which to produce polonium. Since the problem after irradiation is to separate the Po from the residual Bi_2O_3, this experiment was run to yield data concerning the thermal stability of Bi_2O_3. A 1.3395 g sample of yellow Bi_2O_3 lost an average of 0.3 mg/hr during a 15.3-hour soak at 546 C in vacuum.

UO_2-Nb Irradiation Studies. UO_2-25 wt% Nb impacted cermet pellets (96% TD) were encapsulated for eight-hour irradiation tests. Component compatibility and mechanical stability will be evaluated after irradiation at 1130 watts/cm.

Honeycomb Fuel. An extrusion which demonstrates the fabrication of a tungsten honeycomb was successfully pushed. The resulting tungsten honeycomb consists of a hexagonal array 5/8-inch across the flats with 37 cells. The web is 15 mils thick between cells and 7.5 mils thick on the outside. The length of good extrusion (end defects cropped) is 20 inches. The shape of the cells varies from true hexagonal due to asymmetric flow of the large grains in the Mo sacrificial material during extrusion. The billet consisted of a 3-inch OD molybdenum can which closely contained a bundle of 37 tungsten clad, hexagonal molybdenum rods, 0.309-inch across flats, 4 inches long, sealed in vacuum by EB welding of end plugs. The rods were coated with 25-mil tungsten by the WF_6 decomposition process. The billet was preheated to 1800 C in argon by 3000-cycle induction and extruded at 9:1 ratio and 60-inch per minute ram speed. No lubricant was used on the billet; "Oil Dag" was used on the container and ZrO_2 coated die. The surface of the extrusion was excellent. The distortion of cell shape appears due to growth of large grains in the wrought Mo components during preheat, and based on observations of other extrusions can be eliminated by the use of as-sintered powder components.

Submicron Dispersion of UO_2 in Tungsten. Evaluation of the 10 vol% UO_2-W powder produced by coprecipitation of uranium and tungsten from a urnayl nitrate-ammonium metatungstate solution revealed the UO_2 particle size to be in the 0.1 to 0.2 micron range. Macroscopic dispersion of the UO_2 was not uniform. The heterogeneity of the powder was attributed to the stratification of the precipitates during centrifuging. Subsequent attempts in improving distribution by mechanical means have been unsatisfactory. As a result, a revised procedure for producing the powder has been developed.

In the revised procedure ammonium paratungstate is dissolved in water and precipitated by the addition of nitric acid. The tungsten precipitate is flocculant and has physical characteristics similar to the uranyl tungstate. The flocculant precipitate is allowed to partially settle and the clear supernate removed by decanting. Uranyl nitrate is added to the highly acidic floc and uranyl tungstate is precipitated by pH adjustment using ammonium hydroxide. The resulting precipitates do not stratify on centrifuging and compacts made from the reduced powders have uniform dispersion of the submicron UO_2 particles.

A relatively large batch, approximately 270 grams, of 20 vol% UO_2 powder has been produced for initial studies in the evaluation of stringering during extrusion and to evaluate the effect of cold work and recrystallization upon the UO_2 particle size and distribution.

In a separate experiment a higher content product has been produced as a single chemical compound. Uranium tetrachloride was added to ammonium metatungstate to produce $UO_2(WO_3)_3$. This product was precipitated by adjusting the pH to about 9. Since the uranium and tungsten were precipitated as a compound, the product should be inherently uniform. The precipitate has been calcined and reduced producing a uniform dull grey powder containing approximately 46 vol% UO_2. This powder was densified by high energy rate compaction for characterization by electron micrography.

Tungsten-UO_2 Development. Tungsten-UO_2 powders produced from coprecipitated oxides have been densified by high energy rate compaction techniques at 1200 C. The densities of the resulting compacts as determined by the mercury poresimeter method ranged from 92-95% of theoretical.

The as-compacted material was characterized by metallographic techniques. The UO_2 particle size in a sample containing approximately 15 vol% UO_2 ranged from submicron to about 1 micron in size, while the tungsten grain size ranged from less than 1 micron in size to about 4 microns. After annealing at 1500 C in vacuo for 4 hours, particle size increase of the UO_2 occurred, with the UO_2 being primarily in the tungsten grain boundaries. A small fraction of the

UO_2 was found within the tungsten grains. Heating for 20 minutes at 2000 C caused a greater increase in UO_2 particle size than the four hours at 1500 C. Both the UO_2 and tungsten had a grain size of 5 to 10 microns. In all samples the UO_2 was uniformly distributed throughout the tungsten matrix.

6. Other Customer Work

Vibration Testing of Cermets. A satisfactory apparatus has been developed for vibration testing of cermets at high temperature. Testing will be carried out to determine the effect of thermal cycling on the fatigue strength of tungsten-UO_2 cermets.

Thoria Fuels Development. Fabrication of the third and fourth tons of thoria fuel elements (approximately 1200 elements) was completed on March 19, 1964, and shipped to the 100-F Reactor for charging. Discharge and replacement of the current thoria reactor loading, 1 ton - 80% bulk density and 1 ton - 65% bulk density, is scheduled to occur on April 1 and April 15, 1964, respectively.

Fabrication of Sol-Gel ThO_2. Sol-gel ThO_2 prepared by Process Engineering Development was fabricated into cylinders by dry pressing, extrusion, and slip casting. Dry pressed specimens were obtained with a bulk density of 93% TD by firing at 1150 C for one hour. Extruded and slip cast specimens fractured during firing. The partially dried sol exhibited extensive plasticity and was well suited for extrusion. Thorough drying of extrusions should eliminate fracturing during firing.

Irradiation Specimens for Phillips Petroleum. Irradiation specimens consisting of gold-aluminum and lithium-aluminum alloys are being fabricated by a coextrusion process. Five gold alloys, containing up to 25% gold, were cast and sampled. Cladding components were machined from an aluminum-silicon alloy.

J. J. Cadwell

For Manager
Reactor and Fuels Laboratory

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PHYSICS AND INSTRUMENTS LABORATORY

MONTHLY REPORT

MARCH 1964

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

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.

The Lu, Eu, Pu-Al and U-235-Al pins used in the cold N-pile test have been exposed to a uniform flux in the PCTR. Counting data from the pins are being analyzed to determine the effective mass of each pin relative to other pins of the same material. From this information correction factors will be obtained for use in the calculation of the N-lattice parameters. Chemical and isotopic composition of the pins will be determined by chemical analysis and mass spectroscopy, respectively.

The elements for the hot test have been loaded in N-Reactor. After irradiation, the elements will be transported to the 300 Area where the detector pins will be removed and counted. The preparations for transportation are complete. The counting equipment is being checked to insure proper operation.

Recheck of k_{ex} for PCTR N-Reactor Wet Mockup

A quick, independent recheck has been made to determine k_{ex} from original raw data for the PCTR N-Reactor wet mockup lattice. The results demonstrate the validity of previous detailed calculations by Johnson, et al., as published in HW-72696. A report, HW-81457, "Verification of k_{ex} for N-Reactor Mockup from PCTR Measurements," has been written to record the primary data used in the analysis.

Spline Worths in K-Reactor Mockup

Analysis has been resumed on reactivity worths and flux traverses for both standard and gray splines in the K-lattice. The section of the report dealing with worth has been completed. The flux traverse curves are complete, and the rough draft is nearly complete for an informal report.

NPR Utilization Studies

Physics statics calculations for the NPR lattice were continued during the last report period. The purpose of these calculations is to develop an adequate set of physics methods to be used for NPR utilization investigations.

In studies reported previously, effective cross sections were derived on the basis of a completely homogenized cell. This scheme produced an unrealistic graphite temperature coefficient. In the present studies, the thermal cross section averaging was obtained by two TEMPEST runs: one for the graphite region only, at the appropriate graphite temperature, the other for the rest of the cell at the appropriate coolant temperature. The calculated graphite temperature coefficient using the present cross section scheme agreed with the accepted value to within < 2%. All calculations described employed the HRG code for the epithermal cross section averaging and the HFN diffusion code for the reactivity calculations.

Instrumentation

Preparation for the scheduled N Reactor physics "hot" tests was essentially completed. Water-cooled thimbles for the detectors were installed and tested satisfactorily with thermocouples added for measurement of water temperature. All instrumentation required for the hot tests is now ready for use.

In support of the general N-Reactor low level operational testing program, instrument development is being conducted to provide an improved low level neutron monitoring system. A low noise solid state preamplifier is under development to satisfy the need for a circuit that can distinguish fission counter alpha pulses from amplifier noise despite high detector and cable capacitance. To provide greater sensitivity, a boron-lined proportional counter was chosen for use in the initial operational tests. Initial tests with the counter and preamplifier were promising. In addition, considerable design modifications are being incorporated in the commercial main chassis instrumentation to meet existing needs. In particular, the main pulse amplifier and the discriminator are being redesigned. Additional work will be required on both the pulse calibration portion and on the dc amplifier.

General performance and calibration tests were conducted on the gamma energy spectrometers, which will be used in the slow scan portion of the N-Reactor fuel rupture monitoring system. Most of the 24 spectrometers were tested with the use of scintillation detectors and both Cs-137 and Co-60 test sources. Minor problems were found in several units, including incorrect grounding, amplifier oscillation, open meters, and bad delay lines. A majority of the difficulties were corrected during initial testing and calibration. Plans were established for the forthcoming full scale operational tests.

Detailed design was completed, and the installation work was nearly finished, for the sound reducing enclosure surrounding the instrumentation console assembly of the reactor fuels testing loop at PRTR.

Technical assistance was rendered to personnel of Process Engineering, IPD, regarding the many problems associated with calculating shielding temperatures and relating these to reactor parameters.

An invention report was prepared describing a method of detecting reactor fuel element failures based on the alpha activity of the reactor coolant.

Promising results were obtained in experiments at C Reactor storage basin regarding the use of gamma spectra measurements for determining the I-131 content of irradiated reactor fuel. The necessary shielding, collimator, detector, and electronic instrumentation were assembled and a series of experiments were conducted in cooperation with Reactor Design, IPD. Preliminary data indicate that a significant change in the spectra occurs over the first 150 days of fuel element cooling. Work will be continued to establish the range, accuracy, and value of the system in determining the cooling age of irradiated fuel.

System Studies

The N-Reactor secondary coolant system analog simulation was combined with the control system simulator and tested for proper operation. A check was made to determine that all computed steady-state values of the important process variables were within acceptable tolerances and that the controller actions were correct and properly calibrated. Tests were run on the closed loop systems to determine the critical system frequencies and the controller gains required to produce sustained oscillations at low amplitudes.

It was found that the usual method of optimizing one control system at a time results in a relatively unstable over-all system by the time all controllers are so adjusted. Interactions between the control systems apparently combine to increase the effective gain of each sub-system. Therefore, a performance criterion which properly weighs the effects of all the significant process variables is needed. Tests to determine suitable criteria are in progress.

The N-Reactor primary system simulation, including a single-node, three delayed-grouped reactor model, a primary pump and associated piping model, a steam generator bypass valve model, a mixing and transport lag model of the primary coolant temperature relationships and a circuit for calculating the primary coolant volume contraction and expansion rates, was patched on the computer. A second pump model was added to this board for use during runs involving only the primary coolant system.

The primary and secondary system patchboard has been placed on the computer and is ready for debugging. This circuit will be used for all scram runs and others where primary-secondary interactions are significant. Thus, three patchboards are available for use, on short notice, to study various aspects of the N-Reactor system.

Most of the N-Reactor control system simulator is operational. Both equipment racks are assembled and wired. The planned total number of plug-in boxes has been reduced from 35 to 30. Four of these are yet to be constructed; 15 are checked out and are in use; the remainder have not been tested. Some difficulties experienced in the checkout of the control system simulator were traced to unstable conditions which exist in the Philbrick amplifiers when they are connected to the computer trunk lines. Compensating circuits were designed and installed and operation is now satisfactory.

Assistance was given in placing the primary coolant injection system master controller, the five primary pump seal water pressure controllers and three primary coolant flow controllers on automatic operation during the month. The controller settings determined by analog simulation studies of the injection and primary coolant flow systems proved to be acceptable settings on the actual process equipment. The prediction that the master injection controller gain circuit would have to be modified to permit a lower gain setting was verified, and the necessary changes were made to both the Master "A" and Master "B" controllers.

The seal water injection controllers required even lower gain settings than predicted because of a finite "dead band" which exists in the valve positioning equipment. The five controllers were modified to have a gain range of 0.05 to 0.5 and the gains were set at 0.2 to achieve satisfactory operation.

The primary coolant flow controllers were found to tend toward instability above 3300 RPM. Gain settings which were sluggish at lower pump speeds were unstable at 3400 RPM. It is expected that an abrupt non-linearity exists in the turbine steam admittance valve drive mechanism but further tests will be required to confirm this.

The first of two transport lag simulators for the N-Reactor simulation has been completed and wired into the primary coolant loop simulation. The second simulator has been debugged except for the memory drivers and sense amplifiers. Printed circuits for the memory drivers have been made and are now in the process of being wired. Complete system checkout will begin as soon as the driver boards are completed.

Conceptual work was essentially completed and a report written on a new model concept for large nuclear reactors. The concept is intended for use in control systems studies of large reactors in which control of the flux profiles is desired. An analog computer study verified the expected results and it is concluded that there exists a best equivalent structure for the model whose parameters can be determined from actual measurements on a reactor. The results are equally useful for steady-state optimization and for automatic control considerations. The potential method of modeling a large nuclear reactor describes a concept by which measurements can be made on an actual reactor by simple methods. Easily performed measurements provide meaningful information for the control of the flux or temperature profile of the reactor. For manual operation, static measurements indicate the magnitudes of the forward and interaction elements. The flexibility of control, either manual or automatic, can be improved by utilization of the model which is specifically developed for control applications. The manual repositioning of pertinent control rods to achieve a particular flux or temperature redistribution potentially requires only minor control rod manipulation to obtain the data and minor data handling (a matrix inversion) to convert to the new control rod position information. The changes could easily be based on current measurements so that fuel depletion, fuel charging, poisoning, shadowing, and other effects are included automatically by the measuring procedure. New reactor applications might be more feasible because of the improved control capability offered by the ease of prediction of control rod positions by the control model concepts.

SEPARATIONS

Critical Experiments with PuO_2-Plastic Mixtures

The present series of critical experiments with PuO_2-polystyrene compacts and the Remote Split-Table Machine have been completed. During the course of these experiments, which included 75 different assemblies, critical dimensions and masses were determined for bare and reflected rectangular parallelopipeds, with geometries ranging from near cubes, to long columns, to thin slabs. The plutonium concentration in the plastic fuel cubes was 1.12 g Pu/cc (2.2% Pu^{240}), with an H/Pu atomic ratio of 15. Analysis of the critical data is continuing.

The critical thickness of an "infinite" slab composed of the above fuel compacts and fully reflected with Lucite was previously given as 2.6 ± 0.2 in. A slight variation in extrapolation length has been observed with curvature over the core boundary of the reactor, or with change in core shape. A more detailed analysis of the data, considering the variation in extrapolation length with core shape, has been made which better defines the critical thickness of the "infinite" Lucite-reflected slab as 2.46 ± 0.04 in. For a hypothetical homogeneous Pu-water mixture at an H/Pu atomic ratio of 15 (Pu density of 1.62 g/cc), the equivalent critical thickness for a water-reflected slab is computed to be 2.19 in.

Analysis of data from gold foil irradiations for flux distributions has given a value for the extrapolation length of a Lucite reflector, and the difference in reflector savings on inserting a cadmium sheet between the reflector and the core (PuO_2-polystyrene compacts) of the critical assembly. The difference in reflector savings as obtained from the flux distributions is in qualitative agreement with the measured difference in critical dimensions. The results are summarized below. The foil data were fitted to a cosine function throughout the core region of constant cadmium ratio.

Dimensions of Critical Assembly	30.9 x 30.9 x 23.9 cm.	30.9 x 30.9 x 29.0 cm.
Reflector	Lucite of 8-in. thickness on two ends	Lucite of 8-in. thickness on two ends plus 0.03-in. Cd sheet.
Core Composition	H/Pu = 15, ρ_{Pu} = 1.12 g/cc	H/Pu = 15, ρ_{Pu} = 1.12 g/cc
Effective Extrapolation Length	5.85 ± 0.86 cm.	3.28 ± 0.61 cm.

The difference in reflector savings between these two systems is thus 2.57 ± 1.06 cm. From a comparison of these results with experiments involving partially reflected Lucite systems, the combination of Cd sheet and Lucite is equivalent to a Lucite reflector $\simeq$ 0.9 ± 0.5-in.

Experiments with Plutonium Solutions

Planned critical experiments with Pu solutions will now be continued in Hood #1 of the critical assembly room, while new fuel cubes are being prepared for further experiments with Pu solids and the Remote Split-Table Machine in Hood #2. Change-over from one hood to the other, and check-out of control instrumentation for the solution experiments, is well underway. In preparing to re-initiate experiments in Hood #1, six of the essential valves were replaced in the Pu solution handling system. Contamination problems were encountered in connection with these replacements when valve fittings were improperly installed. This necessitated considerable clean-up of Pu contamination. However, this was satisfactorily accomplished and airborne contamination within the critical assembly hood is now below mask level.

Pulsed Neutron Source Experiments

It remains to be established whether pulsed neutron source experiments can provide meaningful values of k_{eff} on units which are far subcritical. An interesting pulsed neutron source experiment was recently performed with a small tank (12" x 12" x 8") filled with three percent enriched uranyl nitrate solution. The solution contained 374 g U/ℓ, 400 g H_2O/ℓ, and 253 g NO_3/ℓ. The effective reproduction factor (k_{eff}) was computed from theory to be $\sim$0.2 with the vessel reflected with Lucite. The pulsed neutron source experiment yielded a value of $\sim$0.15, which is in surprisingly good agreement. Unless these results are unique to this system, the data imply that it may be possible to obtain qualitative measurements of k_{eff} even for systems which are far removed from criticality; if so, these results are highly significant in regard to nuclear safety checks on in-plant equipment. The uranium solution used in this experiment would have a k of only about one, even in an infinite system.

Critical Mass Laboratory Instrumentation

A close coupled nuvistor pulse preamplifier and fission counter were added to Channel #2 of the safety circuit at the Critical Mass Laboratory. The fission counter has a lower thermal neutron sensitivity than the proportional counter previously used. This decrease in sensitivity will reduce, by a factor of more than 10, the possibility of overloading the instrumentation by too high a count rate. The output of the combination produces a maximum pulse of 0.35 volts with a rise time of 0.2 to 0.5 microseconds. The additional amplitude gained with the new preamplifier will raise the pulse out of the noise level and reduce the possibility of false scrams originating in this channel.

Consulting Services on Nuclear Safety--Criticality Hazards

1. Nuclear Safety in HL

   Specification C-15 was issued to cover the loading and unloading of PR cans in the Critical Mass Laboratory.

Specification J-3 was revised to cover the storage of cardboard boxes containing dry plutonium waste in the 231-Z Building.

Temporary Specification No. 4 was issued to cover the processing of plastic metallographic samples containing 26-50 g plutonium metal specimens in the 231-Z Building.

2. Nuclear Safety in CPD

The annual audit and review of CPD criticality hazards control was begun in March. Members of the review committee met with representatives from CPD on three occasions. Purex operations were reviewed on March 11, the 234-5 Building on March 24, and Redox on March 25. Purex and portions of the 234-5 Building were also toured. A tour of Redox and a further tour of portions of the 234-5 Building are to be arranged. Upon completion of these tours, a report will be prepared by the review committee detailing comments and recommendations. This review was undertaken at the request of CPD.

3. Nuclear Safety in NRD

A nuclear safety study of processing 1.95 w/o U^{235} enriched fuel in the 300 and 100-N Areas was begun for NRD. Critical parameters were estimated for 1.95 w/o fuel elements, scrap, and solution. A preliminary review was made of the charging and discharging areas of 105-N Building; it was concluded that the fuel elements could safely be handled in these areas with relatively minor changes in equipment. From a review of the fuel fabrication pilot plant in the 306 Building, it was also evident that only minor equipment changes would be required to ensure nuclear safety.

4. Nuclear Safety of Off-Site Shipments

Approval was given to ship 224 kg of 1.47 w/o and 213 kg of 1.61 w/o U^{235} enriched uranium metal to National Lead of Ohio via rail car, and to ship 6.3 kg of plutonium nitrate solution to Rocky Flats via the Redwood car.

5. GE Class I Shipping Container

The IAEA has expressed an interest in the GE Class I shipping container which is currently in the design stage at Hanford. It has been proposed to include the design as an example in the forthcoming IAEA regulations covering the transporting of fissile materials. Required information on the shipping container has been submitted to RLOO.

(Note that a Class I container is so designed to be safe in any number and arrangement in normal conditions of transport and accidents; the materials used in the construction of each container absorb neutrons and prevent interaction with other containers, thus each container, by virtue of its design, is effectively isolated from all other containers.)

6. Nuclear Safety Training and Education

A new series of lectures in nuclear safety ("A" series) were begun at the Critical Mass Laboratory during March. The "A" series is designed to introduce technically oriented personnel with the technical bases and computational methods used to prevent criticality accidents. About 16 lectures are required for presentation of the material. Twenty-five persons from various Hanford components (principally CPD) are enrolled in the class which meets each Monday at the Critical Mass Laboratory.

Separations Instrumentation

Promising results were obtained in general laboratory testing of the solid state portable gamma spectrometer developed to monitor the 384 keV photons from plutonium in hoods and glove boxes.

Development work was continued on the sample counting system to be used to measure the amount of Pu-239 in liquid sample vials. A laboratory mock-up of the shield and collimator was assembled and tests were conducted with various multiplier phototubes and NaI crystals.

METALLURGY - Nondestructive Testing

(This new section reports work of the Testing Methods Operation which was transferred March 1 from N-Reactor Department to Hanford Laboratories.)

N-Fuel Testing

Uranium Billet Testing

One of the problems in the N-Fuels plant is to improve over-all yield of coextruded fuels. A major reject category has been inner-clad variation of the outer-tube fuel. Parameters which affect the cladding variations include the metallurgical structure of the uranium billet and of the Zircaloy component. The ultrasonic technique used to test the I.D. of the uranium billet makes use of the lateral shift phenomena and responds to the elastic properties of the metal to an approximate depth of .020 - 0.25 inch. The method responds to changes in both structure orientation and grain size.

A similar test is applied to the zirconium components prior to extrusion which makes use of boundary wave phenomena to indicate changes in the Zircaloy structure. Forging memory effects can be detected in the Zircaloy which have been shown to add to inner clad variation of the coextruded N fuel. The results of the study show that there is an influence of both uranium and Zircaloy structure upon the inner clad variation of the coextruded outer-tube fuel. Analysis of the combined uranium and Zircaloy test data has demonstrated a potential pre-characterization of the extrusion assembly.

Uranium Enrichment Testing

A nondestructive tester for measuring N-fuel billets enrichment is being developed which identifies billets as either 0.72, 0.95, 1.25, or 1.60% enriched in a two-minute test period. The tester utilizes a sodium iodide crystal and a single channel analyzer to isolate and measure the 185 keV gamma activity. The plan is to test all incoming billets for enrichment classification. There is a technical problem in connection with radioactive daughter buildup that affects the accuracy of the tester. The gamma emission from the daughter products affects the total count under the 185 keV peak. The age of the billets from the time of physical or chemical separation determines how near the material is to secular equilibrium. This equilibrium buildup is governed by the 24 day half-life of the Th-234 daughter of U-238. The average age of the material at the enrichment tester is three months with possible cycle times as short as ~ 5 weeks. It should be possible in the next few months to test very short cycle (green) material and to evaluate tester accuracy for this application.

X-Ray Fluorescence Test Development

The X-ray fluorescence test is being developed to detect small amounts of uranium contamination in the N-fuel element braze area. This contamination is presently being detected by autoradiographic and alpha counting techniques. The successful application of this technique would allow the discontinuance of the expensive autoradiographic process and so result in lowered fuel costs.

In theory, the tester provides a penetrating X-ray beam which illuminates the braze area and fluoresces the uranium encountered. This fluorescent energy is monitored by scintillation counter and pulse height analysis techniques. The analog output of a count rate meter is used as the contamination indicator. This varying DC signal actuates a recorder to plot the relevant information. The fuel element is rotated slowly and scanned by the above techniques to provide an indication of uranium contamination in a small volume of the braze material.

The mechanical construction of the tester has been delayed somewhat by the recall of borrowed X-ray equipment. New equipment has been obtained, however, and the construction will be completed shortly. The mechanical construction of the equipment has been complicated by the requirement to monitor both I.D. and O.D. braze areas concurrently, on both inner and outer size fuel elements. These requirements have necessitated the design and construction of specially mounted crystal detectors. The crystals, with selected phototubes of similar characteristics, were mounted and they performed satisfactorily in preliminary tests. The equipment should be undergoing evaluation testing during the following month. Standards, constructed from zirconium replicas of a fuel element with simulated contamination in the braze area, are available for this testing. These will be used as a reference to establish the amount of contamination present. Preliminary tests have indicated that the threshold of detection attainable is approximately 1.0 weight percent uranium.

Uranium Surface Contamination Tests

A problem exists with the detection of low level uranium on the end caps of the N-fuel elements. Uranium is brought to the surface in microgram quantities during the end closing operation of the fuel element fabrication. A nondestructive tester has been developed that measures this surface contamination in 10-100 μg quantities in 10 minutes. The uranium is measured by detecting alpha emission with a zinc sulfide scintillator. Several fuel elements have been tested per eight-hour shift and the reject rate has run at 10 percent. The fuel elements are pre-selected from autoradiograph data and only potentially contaminated ones are examined with the alpha tester. Fuel elements with concentration greater than 15,000 ppm are rejected.

A neutron fission thin film detector technique is being evaluated as an alternate method of measuring uranium contamination. It is a more powerful tool because microscopic details of the contamination can be resolved. This method measures the density of fission tracks in a thin plastic film. The film is located on the test surface during a neutron flux exposure which allows the plastic to collect the fission fragments and record the corresponding fission tracks. The tracks are enlarged by chemical development. The atoms of uranium near the surface are the fissile material and thus the fission track density is a measure of the contamination.

Irradiated Fuel Testing

A nondestructive testing station is installed at 105 KE for the purpose of measuring clad and closure bonding, core integrity, and outer cladding thickness of irradiated N fuel. The station has been available for small quantity testing for about a year, and available fuels have been tested. Completion of the project rests on accumulating sufficient data to demonstrate station usefulness, finishing the instruction manual, and demonstrating consistent reliable operation. Difficulties are still showing up occasionally in the underwater equipment. A jerky carriage motion was traced to an improperly located idler sprocket. A drive shaft coupling separation was traced to stripped set screws. Two more crystal holders have broken at a brazed junction; these are being replaced by solid machined pieces. Scum buildup on the rolls occasionally causes erratic rotation of the fuel elements; new rolls are being designed to alleviate this problem.

Zirconium Hydride Detection

The method of detecting hydride in Zircaloy-II reactor process tubes which is being developed generates a complex wave reverberation pattern in the tube wall. By analysis of the received wave form, the difference between the transverse and longitudinal velocity changes caused by the hydride can be determined. Evidence exists that the longitudinal velocity is a function of the amount of hydriding in the transmission path, while shear velocity is affected to a much lesser extent. The time difference between longitudinal and shear pulses is thus a measure of the amount of hydriding in the test sample. Hydride detection measurements on the tubes known to contain hydrogen concentrations are being confirmed. Preliminary tests on two cylindrical samples containing 20 and 700 ppm hydrogen showed a very slight increase in longitudinal velocity and no measurable shear velocity increase. However, structure effects from the heat treatment also appear to affect the longitudinal velocity and higher concentration samples are being prepared for further testing.

Al-Si Fuels Testing

Penetration Test Development

Development of a prototype tester which measures the depth of Al-Si erosion of aluminum jackets on Al-Si fuels has been completed and the instrument has been delivered to the Fuels Production Facility for evaluation tests. During these tests leads to the sensing coils were broken, and it was necessary to modify the coil form and rebuild the coils. Testing of production fuel in quantity has shown that the tester can detect Al-Si penetrations. However, the alignment of the uranium cores is also detected. This unwanted interference is being studied to see if any simple method can be devised to circumvent this difficulty.

Irradiated Fuel Testing

A nondestructive testing station for external bond testing irradiated Al-Si fuels is installed at 105-C. Its purpose is to provide insights to the physical behavior of Al-Si fuel elements under irradiation. The tester has been in use for about a year. A test is now under way whereby ten fuels with unbonds--detected from a group of 294 irradiated fuels--are being rerun with different crystals so as to accumulate reproducibility data. This project will soon be completed and operation responsibility given to IPD. The instruction manuals were completed and distributed to maintenance and operation personnel. Instruction sessions are also completed.

Bond Test Development

A test for detecting internal clad bonding defects from the exterior of the fuel element using an ultrasonic pulse echo technique is under development. Apparent advantages include elimination of complicated probe systems, easier alignment and adjustment and increased reliability. By using transducer focused near the inner clad and passing the ultrasound through the outer clad and uranium an unbond region can be detected by pulse echo. Problems in uranium attenuation and uranium grain size differences in normal fuel cores and variations in outer and inner clad thickness are being solved by proper center frequency, band width and gating. Correlation test comparisons of ultrasonic maps with autoradiographs have been very successful. Improved autoradiography techniques have been developed for comparisons. An improved prototype tester has been fabricated, debugged, and is being used for correlation studies.

NEUTRON CROSS SECTION PROGRAM

Scattering Law Measurements for H_2O at 95°C

Measurements were continued to determine systematic errors which existed in the measurement of the scattering law for H_2O at 95°C. At incident energies of 0.239 eV and greater a substantial effect was due to half-order reflections. A severe internal discrepancy which occurred for a neutron energy change of 0.075 eV is being investigated.

Scattering Law for Polystyrene

A study of the scattering law for polystyrene has been started. The interest in the scattering law for polystyrene arises from its use in critical assemblies in the Critical Mass Laboratory. The status of infrared absorption and Raman spectroscopy of polystyrene is being reviewed.

Methods of preparing suitably thin samples for scattering measurements are being investigated. Preliminary scattering measurements have been made.

Time-of-Flight Spectroscopy for Slow Neutrons

Development of components for the measurement of slow-neutron inelastic scattering by time-of-flight has continued. Test rotors have been satisfactorily operated in low-pressure helium atmospheres. Thin slabs from a large single crystal of copper are being studied for monochromators.

Fast-Neutron Cross Sections

Most of the month has been spent on additional programs and revisions of existing programs for the processing of the MeV total cross section data. Part of this work was due to the use of punched paper tape for data storage of the data obtained in the last measurement series. An urgent request was received from LRL - Livermore for a measurement of the nitrogen total cross section which will be included in the next run. Listings of the Hanford total cross section results were sent on request to Professor H. Feshbach - MIT, and Dr. B. Block - Princeton.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Fast Fuel Test Reactor

A series of nuclear design calculations for a Fast Fuel Test Reactor have been completed. This work was carried out jointly with members of the Design Analysis group. The calculations emphasized the physics statics aspects of the proposed design. Reactor control features were also briefly investigated.

Phoenix Fuel Program

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

In last month's report, a detailed thermal spectrum comparison for some typical Phoenix cores was described. This comparison was confined to the initial reactor condition. The comparisons now include the entire reactor life cycle. The behavior is quite similar to the beginning-of-life reactor condition. The Nelkin spectra still lie between the Wilkins and Wigner-Wilkins distributions. Early in the reactor life, when the Pu concentrations are high, the Nelkin spectrum is closely approximated by the Wigner-Wilkins distribution. As Pu burnup proceeds, the Wilkins and Wigner-Wilkins distributions approach each other, and the Nelkin distribution tends to lie almost midway between them.

2. Plutonium - B-10 Systems

Boron-10 is a commonly used burnable poison in U-235 reactors. If the B-10 is mixed intimately with the U-235 fuel, it is exposed to the same neutron flux. The time normalized poison concentration (P) and uranium concentration (U) are then related by $P = U^{\sigma_B/\sigma_U}$, where σ_B/σ_U is the effective born to uranium microscopic cross section ratio. If σ_B/σ_U is large, the boron burns out quite rapidly, an initially rising reactivity-time curve results, and the reactivity "mismatch" may be considerable. One approach to reduce this mismatch consists of lumping ("self-shielding") the boron so that the effective σ_B/σ_U ratio is reduced. Another possibility is to reduce the σ_B/σ_{Fuel} ratio by using plutonium as the fissile material. Since the plutonium cross section is large, the poison-to-fuel cross section ratio is reduced and the mismatch minimized. Calculations for Phoenix cores with B-10 have been carried out and favorable reactivity time curves were obtained. The Pu cross section is so large that other burnable poisons, e.g., Sm-149, might be even more promising than B-10. Calculations for Pu-Sm-149 have been started.

3. Medium-Enrichment Reactors

Analysis of medium enrichment cores ($\sim$ 30%) employing exposed plutonium as the fissile material has been started. Use of medium enriched fuel still permits relatively compact cores, while some advantage of Pu-239 grow-in can be taken. Only very preliminary, initial results are presently available.

4. Calculations for Proposed VSTR Experiments

Prior to carrying out the VSTR Pu critical experiments, some measurements on U-235 fuel might be conducted. Calculations for a U-235-fueled homogenized VSTR core have been performed with the PHYSICS CHAIN code. The composition of the available U-235 discs is 33.38 w/o of uranium in aluminum with the uranium being 93.51 w/o U-235. The U-Al plates are assumed to be stacked with polyethylene plates of various thicknesses to result several fuel-to-moderator ratios. The calculated k_∞'s for the homogenized cases are given in the following table:

V_{fuel}/V_{mod}	k_∞
0.20	1.8636
0.25	1.8726
0.40	1.8720
0.50	1.8639
0.60	1.8545
1.00	1.8190

The decrease in k_∞ at high V_{fuel}/V_{mod} values appears to be largely due to a rise in α^{235} in harder spectrum environments.

Approach-to-Critical Experiments - EBWR UO_2-PuO_2 Fuel

Subcritical measurements with 1.50 w/o PuO_2-UO_2 fuel elements moderated by H_2O have been completed for three of a total of six lattice spacings. The fuel core is made up of rods 0.372" in diameter and 48-1/2" long. The fuel is clad in 0.027" thick Zircaloy-2. Each rod contains approximately 826 gms of PuO_2-UO_2 of which there is 12.4 gms of PuO_2. The Pu contains 8 w/o Pu-240. The UO_2 is depleted to 0.22 w/o U-235. Both the uranium and plutonium make up 88% of its corresponding oxide. The following table summarizes the results of the experiment and compares them to calculated results.

Lattice Spacing	$\frac{V\ H_2O}{V\ \text{Fuel Core}}$	No. of Rods for Critical: Experiment	Calculated	rods/mk	Difference(b) mk
0.71	2.71	483 ± 5	465(a)	~ 3-1/2	+ 5
			410		+ 21
0.80	3.79	421 ± 5	415(a)	~ 2	+ 3
			360		+ 30
0.90	5.14	456 ± 5	435(a)	~ 3	+ 5
			375		+ 27

(a) Adjusted empirically to agree with Yankee criticals and Pu-Al subcritical experiments. (The numbers shown here may be inaccurate by ± 5 rods since they were interpolated from calculated results at other lattice spacings.)

(b) Between experimental and analytical results based on the computed number of rods per mk given in the table.

The results given in the table show that the k_{eff} of the mixed oxide loadings can be computed with the same accuracy as the Pu-Al loadings, namely, to within 2 or 3% in k.

Critical Experiments Using EBWR Fuel in Light Water

The PRCF has been loaded to critical with the EBWR UO_2-PuO_2 fuel. The experiment proceeded with subcritical evaluation of the strengths of the safety sheets and control rods.

Low Exposure PuO_2-UO_2 Lattice Studies

The fuel for these experiments is being fabricated by Ceramics Research and Development. During the last month, the core material was ground, 42 rods were vibrationally compacted and 22 were welded. If the present rate of production continues, the fuel will be delivered about the end of April.

Analysis of Nickel for Foil Detectors

Although nickel has several uses as a flux detector, it must be very pure for this purpose. It is difficult to obtain such high purity. Therefore, various nickel samples which have been obtained have been irradiated in the PCTR for 2,000 watt minutes. Data have been taken on these samples with the 256 channel analyzer and are now being studied.

Beta Ray Spectrum of Lu-176m

Since the beta ray spectrum of Lu-176m has been reported (Physics Research Quarterly Report, HW-76128, Oct.-Dec. 1962), two other reports have appeared in the literature on this subject. The results of both experiments corroborate the work at Hanford.

Code Development

RBU

Sections of the RBU Monte Carlo logic flow diagrams are being submitted to Graphics Operation for inclusion in Volume II of RBU documentation. The text portion of this volume is nearing completion and proofreading of the first few sections has begun.

Two significant inquiries concerning the cross section work now under way in Theoretical Physics were made during the month. R. J. Howerton at the Livermore site of UCRL expressed considerable interest in this work, and in the possibility of developing closer coordination between UCRL and Hanford in efforts of this kind in the future. M. Temme of Lockheed Missile and Space Division reported successful runs with the version of RBU supplied from Hanford and also indicated their plans for future RBU use. In addition, he requested the support of our Cross Section Library and systems for use at that installation. Further contact with both of these installations is anticipated.

Hanford (RBU) Basic Library

Review, evaluation, and updating of cross section data contained in the RBU Basic Library is continuing. Fifty additional isotopes have been reviewed and updating of the records has been completed. Plates of the cross section data contained in the RBU Basic Library have been made for 77 additional isotopes. The updated cross section information will be placed on a tape which will be referred to as the Hanford Basic Library tape to distinguish it from the information currently comprising the RBU Basic Library.

High Energy Inelastic Scattering

The statistical model code has been debugged and is now available to calculate the cumulative probabilities and energy losses of isotopes tabulated in the inelastic spectrum tables of RBU. Partial or excitation level cross sections as well as total inelastic cross sections are also calculated as a function of incident neutron energy. The calculated values of the total inelastic cross section will be used for those isotopes for which insufficient experimental data of their cross sections exist.

BARNS-II

Further revisions of BARNS-II, the code to process point or group cross sections from the Hanford Basic Library, are being made. The logic of the group averaging procedure was altered to a form which can be readily extended in the future to permit more accurate calculations of transfer cross sections. Considerable effort was spent in determining the contribution of the unresolved resonance region to the cross sections at any energy, and expressions for this contribution suitable for point or group values were obtained. These expressions contain the first order effects of Doppler broadening and of statistical uncertainty in resonance energy and resonance parameters, for either the complete or the background contribution of the unresolved resonances. These expressions, and provisions for optimal Doppler broadening of resolved resonances, remain to be coded.

ZODIAC CHAIN

A number of scheduled improvements to ZODIAC have been successfully completed: (1) All of the crippling size limitations were lifted, (2) unnecessary repetition of TEMPEST and GAM nonburnable regions was eliminated, (3) options were extended concerning action to be taken in case of unsatisfactory NORMOD cycle status, (4) incidence of tape failure was (theoretically) improved by a change in the FORTRAN tape error procedure, (5) the list of burnable isotopes was increased by eight, (6) the shielding calculation in SIGMA-3-H was completely reformulated, (7) GAM was revised to attain compatibility with any GAM library, regardless of the number of isotopes in the library, and (8) a power split calculation has been added.*

PHYSICS CHAIN

The SIGMA-3H calculation of the shielding factors for isotopes in the thermal group and for the Pu-240 isotope for the epithermal groups has been modified. The expression now used for calculating the shielding factors for the thermal group is:

$$f^{th} = \frac{\gamma_1 + \gamma_2 \Sigma_a^{th}}{1.0 + \gamma_3 \Sigma_a^{th} + \gamma_4 (\Sigma_a^{th})^2} .$$

The expression for calculating the shielding factor for the Pu-240 isotope for the epithermal groups is:

$$f_{240} = \frac{\gamma_5 + \gamma_6 N_{240}}{1.0 + 7 N_{240} + 8(N_{240})^2}$$

where N_{240} is the atomic density of Pu-240. All gammas for these equations are input parameters.

The Composite Library Tape has been updated to include four uranium fission products. The SIGMA-3 Library names and numbers, along with their corresponding TEMPEST and GAM Library numbers are as follows:

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* A new ZODIAC Chain Tape will be released early next month coincident with the new ZODIAC User's Manual, which is in rough draft form.

SIGMA-3 Name	SIGMA-3 No.	TEMPEST No.	GAM No.
UFPR-1	180	75	133
UFPR-2	181	76	134
UFPR-3	182	77	135
UFPR-4	183	78	136

Cross sections for these uranium fission products were taken from a document by E. A. Nephew, ORNL-2869, entitled, "Thermal and Resonance Absorption Cross Sections of the U-233, U-235, and Pu-239 Fission Products."

PHYSICS CHAIN Manual

A user's manual, HW-80968, for running HFN or CALX Physics Chain Tapes is about 95% completed. This manual contains information on capabilities and use of the PCT's along with input instructions for all programs used on the Chain.

Cross Sections

Cross sections for the lutetium isotopes (Lu-175 and Lu-176) have been taken from the RBU Basic Library and added to the GAM I and TEMPEST Cross Section Libraries. Some minor modifications were also made to the version of BARNS which is used to update the TEMPEST Library, so as to be able to obtain TEMPEST decimal library cards directly from BARNS.

RBU

The RBU input code was modified to handle the Δ6 cross section range. The Δ6 range allows a linear log-log representation of cross section vs. energy in the RBU Basic Library. The RBU pre-input code and the Library Editor code were checked and proved able to handle the Δ6 range without modification.

Data-Theory Correlation

The critical aqueous homogeneous plutonium sphere experiment 12-1 was run with the RBU Monte Carlo. The Monte Carlo indicated the system to be subcritical. The reason is believed to be the scattering model used. The hydrogen gas cross section used gives the neutron flux an apparently correct energy distribution, but the total scattering cross section for water is too low. A fictitious isotope with a mass of 238 and scattering cross sections equal to the difference between hydrogen in water and hydrogen gas scattering cross section was assembled. The total scattering cross section of water will be preserved by the use of this fictitious isotope. The input to experiment 12-1 has been modified to try this correction.

Methods Development

A method of constructing cell average cross sections, including the flux shape in the moderator, has been developed. Comparison to THERMOS results are currently being made. Initial results indicate the method reproduces the ratio of the average flux in the moderator to the flux at the surface of a fuel rod to about two percent. If the method survives quantitative comparisons, it will be incorporated into the HRG-SPECTRUM chain.

Theory-Experiment Correlation

Comparison of Cross Sections for Pu-239

In an effort to provide a test of basic thermal cross sections of Pu-239, the multiplication constant, k_{eff}, has been calculated for a series of plutonium aqueous homogeneous spherical critical experiments. The experiments covered a range of H/Pu ratios from 297 to 1018, and Pu concentrations from about 25 to 70 gms/liter. The changes introduced were thermal group parameters corresponding to the different 2200 meter/sec values given by Sher. The usual corrections which were H/Pu dependent were changed from experiment to experiment, such as effective values of $\sigma(1/v)$, $\sigma_{tr}(H_2O)$, etc. These were kept constant in a given case when the different cross section sets were used, as were the remaining epithermal group parameters.

The results showed that the differences from different cross section sets was about 3% in k_{eff} and the values of Sher providing $k_{eff} = 1.000$ within a half percent (.005) in almost all cases. Composition uncertainties will account for individual fluctuations. The difference between Wigner-Wilkins spectrum and that of Wilkins (HM) was not generally over a half percent in all cases.

Comparison of Thermalization Models

A series of solution experiments (uranium and plutonium) have been analyzed utilizing three different thermalization models (Wilkins, Wigner-Wilkins, and Nelkin). The experiments chosen covered the range of "hard to soft" spectra for both the U and Pu solutions. TEMPEST was used for calculating the Wilkins and the Wigner-Wilkins spectra and SPECTRUM with the Nelkin kernel. The results show the Wilkins matches the Nelkin spectrum better than the Wigner-Wilkins for the U experiments except at small H/U ratios ($\sim$ 25). For the Pu experiment the Nelkin spectra are better matched by the W-W spectra than by the W spectra, for the whole gamut of H/Pu ratios.

Burnup Experiments

The dissolution of irradiated fuel samples obtained from the destructive sampling of element 5103 marked the completion this month of the dissolution of the L_x Pu-Al fuels portion of the experimental fuel burnup program being done in the PRTR. Initial calculations using the data received from element 5103 indicates that percent burnup of the plutonium fuel will be extended from 35% to 50%. Element 5103 was not originally scheduled as part of the program. An additional outer rod from element 5095 has been cut and the samples dissolved.

Work is progressing with the gamma scan in the PRTR basin. To aid in positioning, the head design has been modified. Rods from element 5187 (.48 w/o PuO_2-UO_2 elements) are presently being scanned, with emphasis of the work on the reproducibility of the data.

Isotopic Analysis of PRTR Samples

Isotopic analyses were provided on 19 plutonium samples of PRTR-irradiated fuel elements in support of the Plutonium Recycle Program. Of these, 2 were macrodrill samples from UO_2 element No. 1101, 8 were macrodrill and 5 burnup samples from Al-Ni-Pu element No. 5111, and 4 were burnup samples from Al-Ni-Pu element No. 5103.

Isotopic analyses were completed and reported on the following elements:

UO_2 - No. 1006 - Burnup Samples (HW-80744), UO_2 - No. 1041 - Burnup Samples (HW-80679), UO_2 - No. 1101 - Burnup Samples (HW-80678), UO_2 - No. 1501 (HW-81046), and UO_2-PuO_2 Pellets (HW-81045).

Instrumentation and System Studies

Detailed planning continued regarding modifications to be developed to improve operational performance of the PRTR fuel rupture monitoring system. Emphasis will be placed on the development of an improved scanning system and on more reliable scintillation detectors.

Investigation was continued on improved methods to be incorporated in the present PRTR exhaust monitoring system. An improvement in both detection sensitivity and general system reliability is desired. Pertinent information is to be supplied by PRTR personnel.

All required engineering work was completed on the annunciator instrument developed for use with the PRTR reactor automatic controller. A final report was prepared.

A solid state relay control monitoring instrument was developed for use at PRTR. The instrument is used to monitor a cascade connection of relay contacts and one alternating current channel. A determination is made as to which circuit opens first. Tests are in progress at PRTR, and if the instrument proves to be satisfactory, gross replacement of considerable wiring in some long conduits can be avoided, with a resulting economic benefit.

The range change components for the PRTR controller power calculator were received, installed, and checked out. The reactor has been at operating power, and the power calculator and BTU compensator are performing satisfactorily. The last piece of equipment that must be checked out before the automatic controller can be used to bring the reactor up to power during startup is the period instrumentation. Tests on the period instrumentation are scheduled for August or later.

A Plutonium Recycle Critical Facility hazards analysis required information as to the severity of nuclear excursions under various conditions when light water is used as the moderator. A light water plutonium study was conducted on the EASE 1132 analog computer. The computer runs were made and all of the requested information was obtained and turned over to the customer.

HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

Although the fourth in a series of high temperature tests of materials had to be shut down after 75 hours of running time in 1200 C nitrogen because of a leak, nevertheless, some useful information was obtained about ceramic materials that are being considered for use in the HTLTR. Hot pressed cylindrical samples of boron nitride lost 4 to 7 mg cm^{-2} when in contact with G32 alumina insulating brick. Boron carbide and aluminum nitride samples held in graphite sleeves did not change significantly in weight. Powdered boron carbide reacted with the nickel capsule containing it. No apparent reaction was observed between UO_2 powder and a containment capsule of nickel. Nickel in contact with G32 alumina brick picked up alloying constituents which lowered the melting point sufficiently to cause slumping. A 1000 hour test at 1200 C in nitrogen has been started in which the behavior of six alloys and four ceramic materials held in contact with graphite will be observed. The insulating alumina brick will not be included in this test. Instead, the behavior of materials in contact with the brick or exposed to its effluent gases will be studied in the furnace in which the heater element tests are made.

A test of a heater element consisting of a graphite rod 1-5/8" in diameter and 4-1/2' long has shown that the low voltage heating system is feasible for HTLTR. The heating element itself showed very little deterioration after 500 hours at temperatures (at its mid-point) ranging from 900 to 1700 C. A poor contact at one end raised the temperature at that end well above this value--probably to about 1900 C since some of the alumina insulating brick in the neighborhood was melted. Another test has been started in which a graphite heater of 1/2" diameter and 40" length has been installed in the furnace. The reduced diameter will permit the potential across the heater to be raised to 20 volts. This is comparable with the voltage anticipated for the HTLTR heaters, and will thus provide a test of the performance of the tapered collet type electrical connections.

Future experiments that might be done with the oscillators scoped for the HTLTR were studied to find out what mechanical motions and experimental apparatus would be required. It was decided to restrict initial experimental capabilities to: (1) A heavy duty oscillator that will move the central cell up to 36" in a step function mode, (2) a light duty oscillator that moves poison samples up to 108" in a step function mode, and (3) provision to remove and exchange poison samples or a centrally located fuel element section while the reactor is at temperature.

The design criteria for the HTLTR calls for flat, blade-shaped vertical safety rods and for horizontal shutter-type control rods. The control rods have thin walled cylinders of fissile material which are slid over cylinders of fissile and poison material. Calculations were made to determine how much poison or fuel is required in different thicknesses of material in support of the mechanical design of the control and safety rods.

NEUTRON FLUX MONITORS

Data are being compiled for a report on plutonium regenerating detectors. Two plutonium and two uranium samples remain to be processed prior to analysis with a mass spectrometer. Data obtained to date on the analyzed samples indicate the experimental and calculational technique for optimizing the initial composition of in-core regenerating detectors will provide useful results. The accuracy of the technique appears to be limited by the spectral assumptions made in the cross sections. Information was received that data regarding neutron temperatures and spectra in water-cooled reactor test facilities had been declassified; this release will permit acceleration of an appropriate unclassified report on the work accomplished to date.

An unclassified report for possible journal publication was prepared describing the method of measuring integrated neutron exposure based on the change in isotopic ratio of U-236 to U-235. The technique was developed to provide accurate exposure data for use in the regenerating detector experiments.

Information was received from the Division of Research, AEC, that the required amount of U-234 will be released for use in coating the regenerating detector chambers now being fabricated offsite. The necessary transfer papers were completed.

Work continued on fabrication methods required to provide a mechanically stable beta current in-core flux monitor. Previous in-core experiments proved the concept feasibility; however, turbulent flow in the water-filled reactor test facility caused vibrations, which in turn caused the signal wires to short to the metal container. The new design should rectify the problem.

Work on microwave methods for reactor in-core neutron flux monitoring was accelerated. The power supplies and the 240 mW klystron were assembled together and successfully tested in the laboratory. The initial reactor experimental assembly, which will be used for examining the effects of neutron flux on the resonance of a tuned cavity, was prepared and moved to the KW Reactor test facility for in-core testing. In the laboratory tests, all developed control circuits functioned correctly and general operation appeared to be fully satisfactory. The actual reactor tests have now been rescheduled to early in April.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

The wiring of the laboratory model of the multiparameter tube tester was completed and evaluation tests were started. Tests are being made on artificial flaws in sections of N-Reactor process tubes. Initial tests indicate that the various circuits of the equipment are performing satisfactorily. Tests are being continued to determine the capability of the tester to separate test sample parameters. One parameter of special interest for the testing of N-Reactor process tubes is the electrical conductivity change due to the presence of hydride.

The assembly of the multiparameter tube tester included fabrication and wiring of new control panels, an external bridge circuit, a two-frequency bridge drive amplifier, and a temporary internal test probe assembly for use in the process tube samples. Incorporated in the control panels are four of the new "three degree of freedom" summing potentiometers.

First tests were made on test samples with holes drilled on the O.D. of the tube to simulate flaws. Results of these tests are tentatively interpreted to mean that this series of holes has a rather low dimensionality. If true, this is a favorable condition for the separation of this type of flaw from other types, e.g., holes on the inside tube surface. An additional test standard with drilled holes on both the inside and outside of the tube is being prepared by the shop.

A study is being made to determine what improvements can be made in the Model 1004 Tubing Tester to increase its range of application. This particular instrument was hurriedly designed last year to fill an urgent need to test N Reactor steam generator tubing. It is now desired to expand its use to detect smaller size defects and relatively small changes in electrical conductivity due to hydriding in Zircaloy-2 process tubes.

The construction of an attachment for use with a projector to display complex voltages on a screen is being considered. Initial experiments with pivoted mirrors projecting a spot on a wall looks very promising and the design of such an attachment seems feasible. The design of the electric drive for the mirrors and the specific means for damping the driver elements is still to be determined.

Fundamental Ultrasonic Studies

Transducers for shear wave velocity and attenuation measurements were received. Initial shear wave attenuation measurements on an aluminum sample were unsuccessful due to improper mechanical loading. Since the shear transducers function over a narrow frequency band, the transmitted pulse rings for a time comparable to the pulse transmission time through the sample. This causes interference between successive echos and results in improper attenuation measurements.

A transducer and sample holder is being designed with adequate backing to produce non-interfering, broadband pulses of short duration. This holder will incorporate a clamping technique to provide good coupling between the transducer and test sample. The holder will also be used to clamp and mechanically load a second transducer on the opposite surface of the test sample for insertion loss measurements.

The pulse frequency spectrum analysis work is continuing. The frequency components in four pulses of different shapes were measured and compared. Two of these were narrow band pulses produced with a pulsed oscillator and two were broadband pulses produced with a shock excited thyratron pulser. The approximate band width of the narrow band pulses was 0.3 and 1.25 Mc/sec and of the broadband pulses were 1.5 and 4.5 Mc/sec. This frequency spectra information will be used in the analysis of pulse shape changes due to diffraction and attenuation.

In order to obtain the forementioned spectra information it was necessary to design a gating circuit which rejects noise and echo signals but passes the desired signal. To achieve this, a circuit was designed which consists of a single transistor gated by a delayed square wave. Using this circuit, the noise and echo signals are suppressed to about 3% of the signal in the gate. The circuit works well for present applications and appears to have other applications where distortion free gating of R.F. pulses is desired.

The development of this gating circuit permitted further experimental studies on the propagation of broadband pulses. By gating the broadband pulse and suppressing the unwanted signals, the frequency spectrum at the beam center was compared to the spectrum at the beam edge. The measurements verified that the edge of the beam was dominant in high frequencies. For example, the ratio of the amplitude at a referenced frequency to the amplitude at twice that frequency was unity at the beam edge and five (5) in the center of the beam.

Test samples of aluminum, stainless steel and Zircaloy-2 having different amounts of cold work and different grain sizes are being obtained for attenuation and boundary wave studies. Due to the stringent requirements on parallelism of opposite sides, it was found necessary to surface grind the rough-cut samples.

Heat Transfer Testing

A new method of thermal testing based on basic heat flow theory is being investigated. Theoretical derivation of heat flow equations that correspond to Maxwell's electromagnetic equations for the one dimensional case have been completed. Although there are obvious differences in the behavior of a heat wave and an electromagnetic wave, enough similarity exists to allow application of the same analytical techniques in both cases. Due to the difference in physical nature, care must be exercised in the application of electromagnetic solutions to the heat flow case. The present approach is to carry out a complete solution for heat flow by using the electromagnetic solution as a guide.

Zircaloy-2 Hydride Detection

To couple the rotating eddy current hydride detector probe to the supporting equipment, a transformer was fabricated with a stationary primary and rotating secondary. The initial design gave promising results but needed mechanical improvement. A second laboratory prototype has been constructed for use in further evaluation studies. The new design avoids changes in the air gap spacing by using a one-piece ferrite core.

A newly designed transistorized eddy current circuit is being used to evaluate the hydride detector. This evaluation consists of noting the change in output of the device when the detector passes over the 500 ppm hydrided area in the standard.

Due to the temperature dependence of electrical resistivity, it will be necessary to compensate for temperature variations in the tube under test. An emissivity independent infrared method will be used for this purpose. It is anticipated that a metal strip bolometer or thermocouple type infrared detector will be contained in the hydride probe head. Manufacturers are being contacted to determine the availability of these detectors.

Nondestructive Test for Isotope Containment Cells

The barium titanate ceramic transducers for performing the higher temperature ultrasonic test on the isotope containment cells were received. Experiments were initiated to determine the types of backing members and techniques of fabrication to produce the best bonds between the crystal and backing. These experiments are designed to determine if the ultrasonic requirement of the transducer is satisfied. As these problems are solved, the radiation resistance requirements will be considered and those concerned with the project will be consulted.

As an alternate method of transducer fabrication, a new technique which completely eliminates the need for a backing member was developed. An external delay line is connected in parallel with the transducer and performs the damping function normally performed by the backing member. Preliminary experiments show this method holds promise. An invention disclosure describing this technique is being prepared.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Utilization of the prototype sheath tubing testing on a variety of testing problems continued. A problem arose where sheath tubing developed cracks on the I.D. surface during vibrational compaction of the ceramic fuel. These cracks may extend through the wall causing a leak in the sheath tube. The tester was set up to test these 25 mil wall loaded fuel elements. The I.D. surface on these tubes was abnormally noisy. Closer investigation indicated these I.D. signals were being received from large fuel particles (up to 1/16 inch diameter) in contact with the tube wall. Although the sheath tubing was loaded with fuel, zigzag shear waves were generated in the tube wall by the 3/16 inch diameter, spherically-focused transducers. The I.D., O.D., and inner wall signals are consequently separated in time due to the different lengths of metal path through which the ultrasound must travel. Since the return signals are separated, the electronic gate was positioned to turn on behind the I.D. surface noise from the incident wave and to turn off just before the I.D. noise reappeared from the reflected second wave in the zigzag pattern. This allowed inspection of the entire wall without presenting the unwanted I.D. surface signals on the test record. Substantial cracks were detected in the tube walls which correlated with destructive examination.

Final design changes and corrections of the sheath tubing test electronics were completed and the unit was put into operation. Stability tests are presently being conducted on the instrument while it is being utilized to test tubing. Minor changes were accomplished to improve system linearity and a new recorder output circuit to utilize a 100-cycle recorder was developed and installed. Fabrication of the final tube test mechanical system along with electrical controls is continuing on schedule with delivery anticipated within two weeks.

Notch depth versus amplitude data were obtained for a variety of entry angles in tubing ranging from 10 to 65 mils wall thickness of zirconium, aluminum, and stainless steel. Sufficient data were gathered to establish the general utility of the shear wave test. Using an eight mil diameter ultrasonic beam, the tester reliably located defects as small as 15 mils long by 1 mil in depth. Test parameters beyond which test results cannot be guaranteed were also established. The most predictable and linear notch depth amplitude curves were obtained when the ultrasonic beam from the transducer was symmetrical and the beam width was less than one-half the tubing wall thickness. If the transducer beam is not symmetrical, the linearity of the notch depth amplitude curves deteriorate somewhat and test results are more difficult to duplicate. For non-symmetrical transducer beams, the received amplitude is a function of the rotational position of the transducer. This indicates that test results could be duplicated in separate test stations (perhaps located at different tubing manufacturing sites) only if symmetrical transducers are used. Standardization of separate test stations would thus be simplified with symmetrical beam transducers.

Test results obtained by propagating shear energy down the axis of the tube to detect circumferential notches are equivalent to results obtained by propagating shear energy around the circumference of the tube to detect axial notches. As the ratio of tube radius-to-wall thickness decreases below 15, the axial notch test sees an apparent wall thinning due to the tube curvature. To compensate for this, the ultrasonic enetry angle must be decreased in order to maintain a linear notch depth amplitude test which sees equal depth I.D. and O.D. notches at the same amplitude. This angle compensation can only be accomplished until the critical angle for longitudinal energy generation within the tube wall is reached. Beyond this point, which is when the tube radius-to-thickness ratio decreases under five, return signals become confusing and would be very difficult to electronically gate. By utilizing a larger 20 mil diameter ultrasonic beam to detect axial flaws and an eight mil diameter ultrasonic beam to detect circumferential flaws, the test can be extended to inspect tubing with radius-to-thickness ratio of three for one mil depth defects. Tubing with a lower ratio than three must either be inspected for only larger defects or must be inspected at a new ultrasound entry angle where equal depth O.D. and I.D. flaws might not be detected at equal amplitudes.

Another limitation for the shear wave test was reached on the stainless steel tubing. Although the metal path through which the ultrasound must travel is very short and zigzag propagation is obtained, the tester could not detect one mil defects. The background signals obtained from reflections at the grain boundaries produced signals with amplitudes comparable to those from a one mil notch. This tubing is presently being analyzed for grain size. These grain boundaries or background noise signals were not observed on the Zircaloy or aluminum tubing standards.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Results of several months of effort toward summarization and analysis of data from all significant United States atmospheric dispersion experiments and detailed analysis of Hanford data have provided a basis for revision in the calculational methods for dispersion estimates. Peak exposure and standard deviations of crosswind exposure distributions at various travel times downwind of a continuous source were found to correlate well with the product of the standard deviation of the wind direction distribution and the mean wind speed. This product represents an estimate of the lateral component of turbulence. Prediction methods developed from these concepts permit extrapolation of results obtained from short period releases to much longer release periods. Good agreement between predicted and observed exposure distributions was obtained from these models, using independent data.

General acceptance of the improved methods for estimation of the dispersion of radioactive materials released to the atmosphere was furthered through presentation of these results at a meeting arranged by the USAEC, Division of Licensing and Regulation, in Bethesda, Maryland, on March 17, 1964. The methods described permit more complete utilization of on-site meteorological data for improved prediction of the lateral growth of the diffusing so as to include the effect of protracted releases as postulated in some reactor accident situations. In attendance at the meeting were representatives of the Advisory Committee on Reactor Safeguards, the Division of Licensing and Regulation, the Division of Production, the Division of Reactor Development, and the U. S. Weather Bureau.

In our continual audit of performance of laboratory equipment and procedures for assaying the quantity of zinc sulfide tracer material collected on filters exposured during field experiments, the "Rankin Counter" was shown to be maintaining its long-term stability. Twenty filters assayed in 1959 were reassayed during the month. The average value of the 20 filters was 100.6% of the 1959 value, indicating no significant change or drift.

Radiological Physics

New urine samples from Alaska indicate that further increases in Cs^{137} body burdens may be taking place there. Urinalyses indicated average body burdens of 210 nCi for males at Fort Yukon and 163 nCi at Point Hope. The values we obtained by whole body counting at these villages last summer were 34 and 39 nCi, respectively. Samples from Anaktuvuk Pass, taken after our visit there in January, indicate that the average burden may have increased significantly since then. We have no reason to believe that our urinalyses are in error, but we are investigating the accuracy carefully because we have so few ways to confirm the results. We have one good confirmation: The Hanford Whole Body Counter shows that the average Cs^{137} body burden has increased 50% in Richland since last summer. Our counts performed on pooled urine samples from Kadlec Methodist Hospital approximately confirm this.

The high Cs^{137} burdens of the Richland residents made it possible to make individual urine measurements to see if they showed the same correlation with burdens as the one we found among the Eskimos. We counted nineteen male subjects on our shadow shield whole body counter and obtained and counted two liters of urine from each one. The fraction of the body Cs^{137} excreted in the urine per gram of potassium excreted in the urine was about 20% less for these subjects than for the Eskimos. This may represent a real difference between Richlanders and Eskimos. Past evidence indicated that our urinalyses of the Kadlec samples could be interpreted all right with the Eskimo correlation. This month, however, it became apparent that the Hanford Whole Body Counter was giving low results for the body burdens of potassium. We consulted with them on this problem and found that the decrease resulted from a change in resolution of the scintillator used in the counter. We were able to restore part, but not all, of the resolution. Use of our new urine correlation with the Kadlec data gives better agreement with the Hanford data, corrected for the effects of the resolution change, than does the Eskimo correlation.

A field party left on the second trip to Anaktuvuk Pass for whole body counting of the Eskimos this year.

Biology, Radiological Chemistry, and we are conducting an experiment to determine what fraction of strontium in the body appears in growing hair. The goal is to be able to interpret results of Sr^{90} analysis of hair samples from Eskimos. A pig was injected with a known amount of Sr^{85} after its hair had been shaved off; hair samples will now be collected and measured.

Biology brought in one of the dogs that had been exposed to plutonium by inhalation to see how well it could be counted in our plutonium X-ray scintillation counter. It was possible to count it very well. In the most sensitive counting arrangement found, the counting rate from the dog was more than fifty times background. The plutonium body burden of the dog is not known, although it is certainly in the microcurie range. From now on till it dies all its excreta will be collected and measured and finally its body will be assayed for plutonium so that we will be able to relate the counting rate with the body burden. It is planned to do this with many more dogs, also.

The positive ion accelerator performed satisfactorily during the month.

Last summer we reported that we were not able to verify the neutron calibration (made by Bramblett, Ewing, and Bonner) of a set of polyethylene-sphere moderators for the Li^6I scintillation counter. This month we calibrated again with neutrons at six energies between 0.16 and 14 MeV. The results were closer this time. Our previous results had not been sufficiently corrected for scattering. Though the results are closer we feel that they are significantly different. This upsets attempts to use the sphere system for approximate neutron spectroscopy.

The noise in the Xe-CO_2 proportional counter for neutron spectroscopy proved it to be from associated instruments and was corrected. Tests of the pulse shape circuit for discrimination against gamma rays showed it to be working properly. When filled with A-CO_2, the He^3 proportional counter neutron spectrometer had good resolution but was microphonic and leaked. We believe we have corrected these faults.

We periodically compare the neutron emission of our standard Pu-Be source against a Pu^{238}-Be source to determine the rate of increase in emission of the former. Present data indicate a growth rate of 1.8% per year, but there is more variability in the data than there should be. We think this is due to mechanical instability of the Pu^{238}-Be source so we plan to mount it in a permanent, fixed position.

The calorimetric measurement of a Pm^{147} source completed last month gave a half-life of 2.678 ± 0.007 years.

Electricians completed the installation of a motor-generator set for the operation of the pulsed X-ray machine. A representative of the Linfield Research Institute then brought up the X-ray tube and put the system into operation. We then dismantled it so the laboratory space would be available for the U. of W. project.

The University of Washington graduate student in residence with us assembled the shield for their neutron generator and made initial tests and measurements of the shield and generator performance.

Instrumentation and System Studies

Detector experiments were conducted with several miniature GM tubes to determine the applicability for measuring animal thyroid dose, as caused by I-131 radionuclide uptake. The dose information will be telemetered as one channel of the animal physiological function telemetry system along with temperature, respiration rate, and blood pressure information. Promising test results were obtained for the GM tube method.

Development work was initiated to provide a method of measuring and controlling smoke inhalation by experimental animals. Several solid state circuits were designed and bench tested. Temperature detecting thermistors are being incorporated to provide control of a valve which will permit the animal to breathe either air or smoke at the desired rate.

Developmental planning was initiated regarding improvements for the dog counting system at the Biology Laboratory. The desired changes include both the detectors and the mechanical assembly, and the modifications are to improve the sensitivity and the scanning rate. It appears that considerable engineering effort will be required.

Laboratory assembly was started on the alpha energy analysis air monitor being developed for use during radionuclide inhalation experiments to be conducted at the Biology Laboratory.

Several chamber design changes were devised for the miniature recharging type gamma dosemeters to improve energy response and operational reliability. The changes will be incorporated in the field prototype instrument.

Valve modifications were incorporated in the radionuclide inhalation instrumentation system to match the available glove box requirements. The necessary mechanical linkage was also modified to provide better operation.

A number of solid state circuits, including a preamplifier, an amplifier, discriminator, a register driver, and a timer were developed for use in a combination alpha-beta-gamma hand and shoe counter. Conservative design was incorporated to assure proper performance and reliability. Extensive laboratory tests indicated that satisfactory performance would be achieved and detailed circuitry drawings were started.

Field tests indicate that correct operation is being achieved with the modified wind component analyzer instrument, used in atmospheric physics studies. A detailed topical report was prepared.

Solid state delay generator and a relay operating circuit were developed for use in radiological dosimetry experiments. Satisfactory operation was obtained.

Experiments continued with several scintillation detectors, including both two-inch and five-inch diameter multiplier phototubes, to determine the applicability to the detection of low energy betas, such as from Pm-147. Initial tests indicated that acceptable detection efficiency could be achieved.

Wind speed and direction information is being recorded from eight levels at the Meteorology Tower near the 200-W Area. It is desired to measure the deviations of short-time averages from a longer time average of the wind velocity vector. A previous study, made to determine the best method of measuring the desired mean square deviations, concluded that a digital approach was the best solution. At that time, however, the cost of obtaining such a digital system ranged from $75,000 to $100,000. The availability of new types of digital equipment which are less expensive can now fulfill the requirements of the meteorology data handling problem at a lower cost. Present efforts are being devoted to a more detailed evaluation of the capabilities of the newer systems.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples at a normal rate during the month.

Work has progressed on making operational the scintillation-type ion detector for the mass spectrometer. Instabilities were found and corrected in the power supply which controls the ion-mass settings of the spectrometer. A noise component was found on the output of the magnet-sweep amplifier and this has not yet been corrected. A different high-voltage power supply for the photomultiplier of the ion-detector was found to give an increase of a factor of ten in gain with a satisfactory noise level. The performance of the 10 MHz scaler is still unsatisfactory.

The alumina parts for a vacuum-lock sample changer for the mass spectrometer were received from the vendor but do not meet specifications and are not usable.

EXPERIMENTAL REACTOR PHYSICS FACILITIES

PCTR Operation

The PCTR was operated on an intermittent basis during the month. There was one unscheduled shutdown due to electronic failure. Four sets of foils were irradiated for normalization purposes.

TTR Operation

The TTR was operated one night for the University of Washington Graduate Center. One set of foils was irradiated for normalization purposes. There were no unscheduled shutdowns during the month.

Subcritical Facility

Critical approach experiments using EBWR fuel rods in water were continued. The 0.71" lattice spacing experiment was completed. Lattice spacings of 0.80" and 0.90" were initiated and completed. Exponential data using half-critical fuel loadings have been determined for the three lattices.

Preparatory work is in process to remeasure the 0.80" spacing of 2.0 w/o Pu-Al fuel rods, 16% Pu-240, in water.

VSTR Status

Preliminary plans have been prepared by Facilities Engineering showing modification to the 305-B Building to permit the installation of the VSTR. It is planned to submit the facility as a general plant project.

COMPUTER FACILITIES

The design drawings and final specification for the digital control computer van were completed and submitted to Purchasing. Delivery of the van is optimistically estimated in late May. A request for appropriation of a refrigerated air conditioning system was issued. This system is to supply temperature and humidity controlled air to the van. A compilation was made of the lengths of all interconnecting cables in the G.E. 412 computer. The list was forwarded to the G. E. Industry Control Department for their use in designing the equipment. A letter was written requesting a change in the original process computer feasibility study to permit purchase of the equipment instead of leasing it as first planned. This change will allow substantial savings in the over-all acquisition costs of the computer.

A digital data acquisition and control system was outlined for a new mass spectrometer. A small computer will adjust the accelerating voltage and magnetic field for the required mass, focus the ion beam, find the exact center of each peak, count the ions in each peak, compute the average and the deviation from the mean and type out the results. An appropriation request for this equipment has been submitted.

The view factor for radiant heat transfer of an open ended cylinder may be reduced to a complex equation involving a triple integral. Iterative analog computer techniques were employed to evaluate this triple integral. The procedure used to evaluate the triple integral was checked out by evaluating the volume of a sphere.

Iterative analog computer techniques were employed for optimization of the mathematical model for a C-column to physical data. Utilization of the new EASE 2133 analog computer allows rapid determination of the optimum parameters.

The report explaining how to use the MIDAS (Modified Integral Digital Analog Simulator) on Hanford's 7090 computer was completed.

The new analog computer is currently being used full time on day shift. The acceptance testing is essentially complete; however, a substantial number of items remain for the vendor to complete.

The analog computer utilization was as follows:

EASE 1132	EASE 2133	
128	135	Hours Up Time
22	22	Hours Scheduled Down Time
16	9	Hours Unscheduled Down Time
166	166	Hours Total

Problems considered during the month were:

1. PRCF Hazards.
2. Meteorology Problem.
3. Cold Water Injection (N Reactor).
4. N Reactor Secondary Loop.
5. N Reactor Primary Loop.
6. View Factor Problem.
7. C-Column Optimization.
8. Redox Specific Gravity Control Study.

CUSTOMER WORK

Weather Forecasting and Meteorological Services

Meteorological and climatological consultation services included 1) presentation of the N-Reactor study on the environmental consequences of a major plant failure to the USAEC, Division of Licensing and Regulation staff, and to members of the Advisory Committee on Reactor Safeguards, 2) compilation of the relative humidity data relating to equipment "moth balling" for IPD, 3) summarization of river temperature data relative to condenser operation for CPD, and 4) documentation of special weather observations relating to an accident investigation for C&AO.

A summary of the discharge and temperature of the Columbia River at Priest Rapids for 1963 and a Preliminary Forecast of these conditions for 1964 was distributed. In general, greater river flow rates and lower river temperatures are expected this year than last.

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	93	86.9
24-Hour General	62	86.6
Special	186	75.3

March was dry and windy. Precipitation totaled only 0.03 in. to bring the first quarter total to 0.41 in., the lowest of record for the period. The average wind speed of 10.7 mph was a record high for March.

Due mainly to an unseasonably cold wave on the 23rd and 24th, temperatures during the past month averaged slightly below normal.

Mass Spectrometry

Isotopic analyses were provided in support of the Enrichment Test II experiment of Testing Methods Operation. Analyses were made on 14 samples of standards used in this experiment.

A plutonium sample was received from the National Bureau of Standards. The results of isotopic abundances measured at Hanford Laboratories and several other mass spectrometer laboratories will be used to establish the isotopic content of this sample as a Pu standard for the NBS. These analyses are being carried out at the request of the NBS and the Division of Nuclear Materials Management - AEC.

Instrumentation and System Studies

Engineering improvements were completed on the timer circuits of a resistance welder for Advance Fuels Development, HL. Repeated tests indicated that satisfactory performance was achieved.

Detector development progressed for the mask monitoring system being designed for use at the HAPO Laundry Facility. The major problem is that of providing a soft beta scintillation detector head which matches the inside contours of the face mask. In addition, necessary solid state circuits are being developed.

Laboratory testing was continued on the completed U-235 fuel enrichment measuring instrument designed for Metal Fabrication Development, HL. Comparison tests, using samples of material with different U-235 content, gave encouraging results. The enrichment percentages ranged from 0.15% to 3%.

Considerable modifications to improve performance were incorporated in a solid state aural alpha-beta-gamma monitor for Calibrations, HL. The instrument, which uses a scintillation detector, is employed to monitor portable instrument surfaces for contamination. The modified instrument performed satisfactorily.

Engineering assistance was provided to Experimental Physics Research on the design of a phase shifting system for a motor control. Further work is contemplated on a feedback control system.

Equipment for logging the creep of metal test specimens during irradiation was placed on-line at the 100-KW in-reactor test facility and connected to three creep capsules. Electrical noise pickup by the in-reactor capsule signal leads has been causing difficulties. General reactor operations such as starting and stopping pumps, generators, etc., are causing line transients which may upset the data logger logic. Filtering and isolating techniques are being employed in an attempt to solve the problem.

An analog computer simulation of a chemical extraction column was conducted to determine the type of control scheme required to maintain acceptable control of column specific gravity. The process is characterized by 32 minutes of deadtime. A controller scheme employing deadtime compensation for linear predictor control was found to give the most satisfactory control. The range of optimum proportional and reset rate setting was determined through consideration of the system's response to step changes in controller setpoint.

Optics

Two horizontal traverses were run with the new Tramek process tube at B Area. Agreement between the two runs was excellent in that the greatest difference was 28 mils. Difficulty was encountered when removing the Tramek from the process tube at the end of each run. Modifications performed at the Optical Shop corrected this difficulty and a calibration run was successfully made before further process tube testing.

A new program has been devised for computing displacements which is especially suited to the data taken with the 2 FT Tramek. When an electron microscope is used to examine the surface of thick metal samples, the angle of view is quite large and the resultant picture is distorted by foreshortening in one dimension. It is desired to remove this distortion as if the picture were taken perpendicular to the metal surface. Two methods for accomplishing this feat have been demonstrated. One employs two crossed cylindrical lenses operating at conjugate magnifications. This has the effect of magnifying one dimension while reducing the other.

The second method used a slit camera which leaves one dimension unaltered while reducing the other. This second method was demonstrated by adapting the Photo Lab Slit Camera. Although both of these methods have satisfactorily been tested, the slit camera approach appears to require the least modification of existing equipment.

During the four-week period (February 16 to March 15) included in this report, the following shop work was performed:

1. Twenty polished stainless steel pieces were aluminized to provide a base for the Al_2O_3 protective coating to be formed by autoclaving. This work was performed for Advanced Fuels Development.
2. Twenty glass bearings were fabricated for CPD.
3. A quartz parallel-piped was fabricated for use in ultrasonic testing.
4. Four crane periscope heads were repaired for CPD.
5. The microscope and attached periscope of the microhardness tester used in the 327 Building were redesigned to conform to a new shielding arrangement. Necessary parts were fabricated to facilitate the modification.
6. One shutter and an Opton microscope were repaired for Radiometallurgy.
7. One laser crystal was reground for Ceramics Research.
8. A demonstration model pressure indicator was fabricated for the Waste Solidification Engineering Development program.
9. One sodium iodide crystal was ground, polished and aluminized for Testing Methods.
10. The Basin I underwater viewer at 105-C was repaired.
11. A filar micrometer eyepiece was repaired for Plutonium Metallurgy.

Physical Testing

Eddy current testing of selected tubes in the N-Reactor Steam Generators was conducted for the purpose of determining whether corrosion had propagated since its discovery in July 1963 and to obtain specific reference data from which reliable comparisons can be made during future inspections. Retesting of the tubes listed in the sequential sampling program conducted last summer, followed by a rather meticulous comparison of the two recorder traces, resulted in a positive conclusion that no detectable changes have occurred during this period.

Work continued on the scoping of a facility to inspect waste storage tanks in support of the Waste Solidification Engineering Development Program (WSED). A comprehensive description of the work that has been conducted was written and submitted to WSED as an interim status report (PT Memo 64-2). A prototype model of a pressure gage which can detect and read out both positive and negative pressures, without compromising storage tank integrity, was built and demonstrated to WSED personnel. The techniques of neutron radiography are being evaluated as a possible means of inspecting the highly radioactive contents of these tanks. Several neutron "shots" have been taken using the Van de Graaff accelerator, but the broad beam emitted by this unit is not ideal for radiographic purposes. As a result the radiographs of glass samples taken thus far have rather poor definition.

The Nuclear Division of Combustion Engineering has conducted tests to determine the effects of decomposed vapor phase inhibitor (VPI) on 304 stainless steel tubing which was removed from the N Reactor steam generators. Six tubing samples, which had been examined by the 1004 eddy current testers at Windsor, Connecticut, were used for these tests. Previous testing had shown no significant indications. After some four months of exposure to condensed VPI, the samples were returned to Hanford for retesting. This examination showed that two of the six samples (BTH-3 and BTH-4) had developed areas of severe intergranular attack, penetrating up to 100% of the wall in certain spots. The other four samples appeared unchanged. The attack originated on the inner walls, and both visual and metallographic tests were used to confirm the eddy current test findings.

Final assembly of the capacitor discharge equipment for rapidly heating metallic specimens by discharging electrical energy through them is near completion. An induction current probe was designed and fabricated. It consists of a small four-turn coil enclosed in a glass tube. By integrating the output of the coil, a representation of the current waveform during discharge can be displayed on an oscilloscope. From the decay of the current and the frequency of the wave, the circuit inductance, resistance, and the instantaneous values of current can be calculated. These values determine the amount of energy delivered to the specimen. An oscilloscope presentation will be used to view the temperature. A silicon photo voltaic cell will be used as a radiometer for temperature above 500°C.

Fluorescent penetrant work continued on K Reactor vertical control rods to detect fatigue cracks in the brittle Boron steel rods. Penetrant work on the Vanstone flanges on the rear face of 105-KW reactor was completed. This completes the Zircaloy retubing program in the K reactors.

A test program was started in support of fabrication of remote energy source fuel cells. Test cells are examined as they are fabricated to measure wall thickness variations resulting from fabrication procedures. Four cells have been tested to date.

Grain size determination was made of an N-Reactor steam pressurizer which was inadvertently overheated in excess of 1450 degrees F. Three areas were metallurgically prepared in the field, replicated and read for grain size determination, as per ASTM standards. The three areas compared favorably with ASTM standards, numbers 7 and 8.

Punched notches of various depths were made on the O.D. and I.D. surfaces of aluminum, zirconium and stainless steel tubing for use as ultrasonic test standards. The notches were microscopically examined, measured, and conditioned to maintain a depth tolerance within ± 10 percent of the total notch depth.

INSTRUMENT EVALUATION

In-field performance of the scintillation alpha-beta-gamma hand and shoe counters has been satisfactory. One instrument in 327 Building, operated next to several older hand and shoe counters, has detected several cases of personnel contamination that were undetected by the old units. The mainteance personnel training program for the new instruments is being continued on a part-time basis. Operating and maintenance procedures have been disseminated to all interested personnel.

Evaluation testing was performed on several developed scintillation detectors which were designed to detect low energy beta particles. Results of the scanning tests indicated a three percent counting efficiency was achieved for this isotope. In addition, a better process was devised for applying the aluminized Mylar light shield material which prevents undue stretching and thereby eliminates separation of the aluminum film.

Engineering assistance was rendered to maintenance personnel on a combined logarithmic and linear response area radiation monitor in use in the 325 Building.

Engineering assistance was given to Calibrations, HL, regarding preparation of purchase specifications for ten Victoreen Model 440 portable dose rate instruments. The purchase is based on the evaluation test results previously reported.

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RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

N-Reactor Fuel Overheating Study

An irradiated 24-in. NPR outer fuel element was heated to 1800 F at a programmed rate, then quenched with 150 F water. The experiment was performed to establish the integrity of the fuel during a delay in the operation of the back-up cooling system of the reactor. The element retained its integrity and no breaks in the cladding were discernible. The outer cladding showed many blister-like projections ranging from about 3/16-in. to 1/2-in. in diameter over all except the last inch of each end of the element. The inner cladding developed a few wrinkles. The experiment was designed to collect non-gaseous fission products; however, the cladding was not breached and fission product releases, if any, were below detectable limits. Continued experiments are underway using the inner element.

Electrodeposition of Nickel on Uranium from Non-Aqueous Solutions

A study has been started of the feasibility of plating nickel onto uranium metal surfaces from non-aqueous solutions. It is hoped that plating from such solutions will eliminate or decrease oxide and hydride formation at the uranium-nickel interface and produce a more adherent plate. Preliminary experiments with dimethyl sulfoxide and with the Divers solution prepared by condensing anyydrous ammonia over ammonium thiocyanate have shown that the solubility of the nickel(II) compounds tested thus far is not very great in either system. Anodization of nickel metal was used to produce a satisfactory nickel(II) Divers solution in cyclic voltametric measurements in the two systems, Ni/Ni(II) Redox curves were found to approach polarographic reversibility in the NH_4SCN-NH_3 Divers solution but not in dimethyl sulfoxide, where the behavior of the Ni/Ni(II) electrode reaction is very similar to that in aqueous solutions.

SEPARATIONS PROCESSES

High-Isotopic-Purity U-233 Studies

Production of the first sample of Hanford-produced "pure" U-233 (ca. 1 ppm U-232) was described last month. Hot cell and laboratory effort during the current month was directed at completing purification of the U-233 remaining in-cell in various processing fractions; laboratory studies on the selective leaching of uranium and protactinium from thoria; and initiation of work on protactinium and ruthenium absorbants, scavenging agents, etc., aimed at assuring adequate decontamination from these elements during Redox plant processing of irradiated thorium.

Several extractant systems have been scouted for possible use in U-233 separations in B-Cell. TBP (5-10 percent) - CCl_4 gave adequate uranium extraction and excellent phase separation behavior and was used in the work reported last month. It also had the important advantage of being completely fireproof and explosion-proof. Although satisfactory for processing of nitrate solutions, TBP did not extract uranium quantitatively from oxalate supernates. DBBP (dibutylbutylphosphonate) - CCl_4 on the other hand extracted uranium very efficiently even from oxalate solutions, and gave excellent decontamination from protactinium (which was apparently complexed by the oxalate). Extracted uranium was readily stripped with dilute sodium carbonate solution. DSBPP (di(sec)butylphenylphosphonate) in DIPB (di-isopropylbenzene) also extracted uranium from oxalate solutions and will be explored further. It was less effective when diluted with carbon tetrachloride. With all of these organic solvents, radiolysis is a serious problem, at least in the B-Cell environment.

The thoria leaching studies have continued to show very little promise-- even with high-surface-area, spray-calcined thoria. Best results were obtained with 7 M HCl which, however, removed less than 20 percent of the Pa-233.

Dissolution of Thoria

Studies are in progress on the dissolution behavior of ThO_2 prepared by Mallinckrodt, Inc. using a modified Sol-Gel process. The material used in these studies was dynapacked at Hanford to a density about 80 percent of theoretical. Complete dissolution of this material in boiling 10 M HNO_3 - 0.01 M HF - 0.03 M $Al(NO_3)_3$ to yield 0.8 to 1.1 M Th solution required 6-8 hours. It was also noted that 60-100 mesh size (Tyler Sieve Series) particles dissolved faster than 100-200 mesh size material. The smaller particles were more difficult to suspend so the lower dissolution rate may be attributed to poorer contact between solids and solvent.

Thorium Processing

During pneumatic impaction densification of thoria (by Advance Fuels Development) considerable difficulty was experienced due to excessive moisture content in off-site prepared thorium oxide fuel. Equipment was designed and installed in the Cold Semiworks and approximately 1300 pounds of ThO_2 fired to ca. 1000 C. Subsequent pneumatic impactions have been fully successful.

Po-210-Bismuth Separation Studies

Cerium Metal Scavenging - A tracer level test of the use of cerium metal as a scavenging agent for the removal of Po-210 from irradiated bismuth metal was reported in the November, 1963, monthly report. This work has now been repeated, using full-level Po-210 in irradiated bismuth. The scavenging was less complete in the full-level run (probably because of a shortened time cycle), with only 60.1 percent of the Po-210 concentrated in 8.3 percent of the bismuth, as compared with the tracer run results showing 81 percent of the Po-210 concentrated in 4.95 percent of the bismuth. In both runs the major portion of the Po-210 was found in the bismuth-cerium (Bi-Ce) "skin" held up in the bulb. Some segregation of Po-210 occurred in the Bi column as Ce-Bi floated to the top, although this effect was appreciable only in the tracer run where the time cycle was most favorable. In a practical process, this probably means that Po-210 would concentrate in a skin retained in a bottom-pour crucible. Overall, it is concluded that this concept represents a feasible means of achieving Po-210-bismuth separation.

Volatilization - A program to study the feasibility of separating Po-210 from irradiated bismuth by volatilization from bismuth oxide was started with an investigation of the volatility of Bi_2O_3. In preliminary work, it was found that induction heating of a sample of Bi_2O_3 in a platinum dish at 1070 C caused transport of Bi_2O_3 and alloying of the platinum with bismuth. A trial run with transpiration apparatus at 750 C gave evidence of the conversion of Bi_2O_3 to Bi_2O_5 by low concentrations of oxygen.

Solvent Extraction - Preliminary experiments on the solvent extraction of polonium from nitric acid solutions show greater promise for the success of a solvent extraction process than was first predicted. Of the solvents studied (DBBP-xylene, TOA-xylene, Aliquot-336-xylene, hexone, and 1,4-diethyoxybutane), the best results were obtained with hexone and 1,4-diethyoxybutane. Under conditions giving maximum Po-210 extraction from solutions containing tracer-level bismuth (E_a^o values of three to five), polonium-bismuth separation factors of about 30 were obtained with these two solvents.

Scrap Recovery Flowsheets

The current flowsheet under study for 234-5 Building processing of scrap containing plutonium, uranium and thorium involves co-extraction of uranium and thorium into 30 percent TBP-CCl_4 in the first column and rejection of plutonium as plutonium(III) to the aqueous raffinate. The adequacy of this flowsheet for recovering uranium and thorium was demonstrated in a mini mixer-settler run; unexpectedly, however, 75 percent of the plutonium also extracted. Subsequent batch contacts demonstrated plutonium extraction from

3.0 $\underline{M}$ HNO_3 - 0.5 g/l Pu - 0.45 $\underline{M}$ $Fe(HN_2SO_3)_2$ - $Al(NO_3)_3$ solution increases significantly as the $Al(NO_3)_3$ concentration increases from 0.0 to 1.0 $\underline{M}$.

In a further mixer-settler run the feed $Al(NO_3)_3$ concentration was decreased from the 1.0 $\underline{M}$ used in the first run to 0.5 $\underline{M}$. Plutonium extraction decreased from 75 percent to about 10 percent but the thorium concentration in the aqueous raffinate (0.2 g/l) was about 20-fold higher than desired. Current efforts are directed toward finding minimum salting requirements for thorium and uranium extraction and toward devising improved techniques for scrubbing plutonium from the organic phase.

<u>Alternate Plutonium Reductant for Purex</u>

Sodium formaldehyde sulfoxalate (SFS) is being studied as a replacement for ferrous sulfamate in the Purex plant 2D column. Laboratory tests of SFS stability and the effects of the presence of SFS degradation products in various process streams are being made.

Stock aqueous solutions of SFS oxidize slowly even in closed containers protected from light. The half-life of SFS in a 1.0 $\underline{M}$ aqueous solution is about 300 days at 25 C but only about six days at 50 C. Stability of SFS toward oxidation increases when stock solutions are also either 0.05 $\underline{M}$ NaOH or 0.6 $\underline{M}$ N_2H_4.

Elemental sulfur can form under certain conditions in acidified SFS solution through a mechanism apparently involving conversion of sulfoxalate to thiosulfate as an intermediate step. Available evidence indicates, however, that sulfur is not formed under normal Purex process flowsheet conditions. Sulfur formed from SFS is soluble in 30 percent TBP-Soltrol; retention of plutonium in the organic phase is not increased by the presence of sulfur in the solvent.

<u>Ion Exchange Contactor Development</u>

Testing of the modified folded-loop ion exchange contactor has been completed. This contactor, simulating an anticipated Purex redesign, showed minimum resin movement capabilities of 10 to 15 inches per minute with greatly reduced slip liquid movement as compared to the present Purex system. Strong acid slip liquid was displaced at simulated Purex rates from the stripping section in 3 to 5 minutes following the end of the resin moving period and the beginning of the process stream flows.

Investigation of the Redox ion exchange contactor has been resumed. This unit shows greatly enhanced resin movement ability using an air-vacuum pulsed standpipe for resin pumping. Resin pumping times will be decreased by at least 50 percent resulting in a major overall increase in contactor efficiency. The arrangement has the capability for moving relatively dry resin.

Dissolver Heat Transfer Studies

Studies have been undertaken to determine the heat transfer characteristics of plastic lined equipment, which is used in the Plutonium Reclamation Facility for handling highly corrosive process solutions. Initial investigations have been made with an available Teflon lined dissolver, heated externally with electric strip heaters. This system is quite inefficient; for a total power input of 16,000 Btu per hour, only 2700 Btu per hour was transferred to the process solution. A period of one-half hour was required to regain thermal equilibrium following a step-change in input from 3.0 to 3.8 kw. It is estimated that the warm-up period at start-up would be one and one-half hours. Because of these characteristics, external heating of Teflon lined vessels is not recommended. Designs for corrosion resistant internal heaters are now under study.

Disposal to Ground

Only minor changes are evident in the areal extent and concentrations of ground water radiocontaminants over the past several months. Strontium-90 concentrations above the routine detection limit, 7×10^{-8} µc/cc, exist in only two wells; both of these monitor the abandoned 216-S-1 and 2 Redox process condensate cribs, and have continued to show low, but detectable, concentrations of Sr-90 (10^{-7} - 10^{-6} µc/cc) for the past six years. The absence of this radionuclide in the many adjacent monitoring wells is in part attributable to the very slow migration rates of this isotope in the ground water zone.

Neutron Irradiation of Linde 13X Zeolite

Two four-gram samples of the sodium-based Linde synthetic zeolite 13X were irradiated to 3.16×10^{18} nvt in a KW reactor poison column as a preliminary step in ascertaining the usefulness of synthetic zeolites for "hot atom" or other reactions involving neutron irradiation. The samples were opened a week after irradiation was completed, and X-ray diffraction patterns of the 13X were obtained. No changes in the crystal structure of 13X were detected.

Duplication of a point on the sodium-cesium isotherm of the unirradiated 13X was attempted with the irradiated 13X. One of the implications of duplication of an equilibrium point would be that the total ion exchange capacity of the irradiated and unirradiated 13X was the same. Initial results with tracers were inconclusive due to extraneous radioisotopes in the equilibrium solution of the irradiated 13X. Preliminary flame photometer cesium results showed that there was no significant change in the irradiated 13X sodium-cesium equilibrium at that one point.

WASTE MANAGEMENT AND FISSION PRODUCT RECOVERY

Kilogram Scale Technetium Purification

Plans were described last month for completing purification and isolation of approximately one kilogram of Tc-99, which was recovered by the Chemical Processing Department from Purex waste tank supernate. The first phase has been completed. This consisted of passing the solution through a 10-liter bed of 8-10 mesh silica gel to remove Zr-Nb activity. The operation went very smoothly and required only about 10 hours to complete. Dose rate measurements indicated a gamma DF of about 20 (the dose rate decreased from ca. 20 r/hr on the half-inch diameter inlet line to 900 mr/hr on the outlet line).

The second phase has begun. The pH of the feed has been adjusted to 1.0 by in-cask addition of caustic (only a small amount of solids are formed at this pH) and the adjusted feed is being pulled through a 6-in. x 5-ft. ion exchange column packed with 40-50 mesh IRA 401 anion exchange resin. Following loading and washing, the technetium will be eluted and the product fractions transferred to the laboratory for evaporation, nitrate kill, technetium dioxide precipitation, and eventual hydrogen reduction to technetium metal.

Purex Tank Sludge

A second attempt, with redesigned equipment, to obtain a sample of sludge from a Purex waste storage tank was apparently successful. Approximately 700 grams of sludge was obtained from the 103A tank and unloaded and visually examined in B-Cell of the High Level Radiochemistry Facility (where it will be used in studies aimed at completing definition of the Waste Management flowsheet). The sludge consisted of two phases, one soft and the other extremely hard, suggesting that there are two distinct layers in the tank. Analyses of both are being sought. Process studies with the sludge will be delayed by lack of cell space until completion of the "clean" U-233 program.

CSREX Studies

If the CSREX process is selected for Waste Management application and a strontium product of purity necessary for isotopic power applications is desired, a second CSREX solvent extraction cycle may be required. An initial attempt to remove calcium from a concentrated strontium crude by using a modified CSREX process was unsuccessful. The procedure attempted to selectively strip strontium from the solvent with 1 M citric acid. The lack of any appreciable calcium DF is attributed to the low calcium distribution ratio obtained with citric acid. Partial neutralization of the citric acid or the use of an acid forming less stable complexes, such as formic or hydroxyacetic acid is expected to greatly improve the DF. The high concentration of strontium and calcium in the process (0.031 M each in the feed) did not adversely affect strontium recovery.

Strontium-Lead Sulfate Carrier Precipitation

The effect of making several successive lead sulfate strikes without precipitate removal continued to be investigated. It was discovered that precipitate from previous strikes had a significant remaining affinity for strontium in fresh feed with the result that as little as 0.01 M added lead (with 0.02 M in the initial strike) was sufficient to reduce the strontium concentration in the supernate by 20-fold. The total amount of lead required for six successive strikes was in the range of 26 to 33 moles per mole strontium, giving average losses of 3.4 and 5.2 percent in the two test runs. The old precipitate picked up strontium at a reasonably constant rate, reaching half of its ultimate equilibrium value in about 10 minutes, 95 percent in about 45 minutes. A 20-minute digestion period with fresh feed strontium consistently lowered the strontium concentration in the supernate 3-fold.

Metathesis of the well-digested precipitate from the above tests was very difficult when relying primarily on carbonate exchange with sulfate; however, addition of NaOH to a caustic/lead mole ratio of 15 provided excellent strontium recovery. The caustic ratio chosen is about equal to that required to dissolve lead oxide as sodium plumbite. The metathesis solution contained 0.5 M carbonate to prevent strontium loss; the strontium loss in the metathesis solution and in two subsequent 0.1 M Na_2CO_3 washes was only 1.5 percent. The bulk of the strontium was dissolved in 0.7 M HNO_3 to yield a solution containing 0.01 M strontium.

Cesium Removal from High Level Waste

A study is being made of methods to minimize the amount of ammonium ion in the scrub effluent from a column of Linde AW-500 zeolite loaded with cesium from alkaline waste supernatant solution. Low ammonium ion concentration is desired since the scrub effluent may be blended with other wastes and concentrated, releasing ammonia into the vent system. Based on current process flow sheets, eight column volumes of 0.2 M $(NH_4)_2CO_3$ are to be used to scrub sodium from the exchanger. This amount of ammonium ion is 80 percent in excess of the stoichiometric amount needed to replace exchangeable sodium ion in the AW-500 column. Without the scrub a minimum sodium to cesium ratio of 60 can be expected in the eluate.

The amount of scrub in current process flow sheets is based mostly on results with Linde AW-400 exchanger. Recent work with AW-500 indicates that a 20 percent excess may be sufficient. Initial results show a sodium to cesium ratio of 2.4 with a 20 percent excess of ammonium carbonate on an AW-500 exchanger. However, due to possible fluctuations in the ammonium carbonate concentration in the scrub it may be difficult to determine the point where an optimum amount of scrub has been used. A method of determining ammonium ion breakthrough by gamma monitoring of the column or effluent is presently being studied.

Experiments were carried out, at the request of CPD design engineers, to determine whether Linde AW-500 zeolite could be dissolved with concentrated acids or bases. The zeolite showed a 30 percent weight loss after exposure to 16 N HNO_3 at 90 C for one hour and a 46 percent weight loss after exposure to 16 N NaOH at 90 C for one hour. Weight loss appeared to be independent of particle size in the range 20 to 50 mesh. Nitric acid concentrations greater than 8 N resulted in only slightly greater dissolution. Treatment with concentrated caustic for longer time periods is expected to give nearly complete dissolution.

EQUIPMENT AND MATERIALS

Electrical Discharge in Radiation Shielding Windows

A sample of Penberthy Co. 3.8 density shielding glass received 1.3×10^7 r from exposure to radiation from calcined Purex 1WW at rates of about 10^4 r/hr without occurrence of electrical discharge. This sample was then exposed to a radio-cesium source (4×10^4 r/hr); discharge occurred during block movement after 160 hours when the total exposure was 1.9×10^7 r. Another sample of this glass is currently being exposed to irradiated reactor fuel in the 105 C basin. These samples are 5 x 5 x 6-in. blocks mounted in steel frames with plywood shims touching and supporting the glass.

Non-Metallic Materials

A sample of Ebolon (Chicago Gasket Co.), a filled Teflon bearing material, swelled and softened slightly when exposed to boiling $UO_2(NO_3)_2$ solution for 31 days. This material softens and embrittles under irradiation but still appears usable after a total exposure of 10^8 r.

A carbon-filled, cross-linked polyethylene marketed by G.L. Cabot Co. under the trade name "CAB-XL" was immersed 90 days in Purex HAX at 50 C. The sample did not swell and embrittled only slightly.

Corrosivity of Fluoride-Containing Purex 1WW Solution

Some carry-over of fluoride ion into the acid waste system would be expected if Zircaloy-clad fuels are reprocessed in the Purex plant. The corrosivity of simulated 1WW solution containing fluoride, with and without corrosion inhibitors, is currently under investigation. Curiously, the addition of fluoride (up to about 0.1 M) to 1WW solution decreases the corrosivity of the solution to both 304-L stainless steel and HAPO-20 alloy. At higher fluoride concentrations the corrosivity of both solution and vapor phases becomes excessive to 304 L. Most inhibition tests made to date have utilized boron (added as B_2O_3). Favorable results have been obtained. For example, 1WW solution containing 0.25 M fluoride and 0.25 M boron was less corrosive by a factor of three to 304 L in the liquid phase than 1WW solution containing no added boron or fluoride.

PROCESS CONTROL AND DEVELOPMENT

Ion Exchange Column Pulser Control

A control system for an ion exchange column pulser has been developed and tested. Because of the variable flow characteristics of the resin-water system a simple timed pressure and vacuum cycle is not workable. The initial approach to the problem utilized a timed sequence plus liquid-level over-ride. A simpler system has since been evolved in which a controlled rectifier is used as a memory device and the high and low level detectors serve as gate and main power supply switches. The mode of operation is one of continuously traversing between high and low limits, without attempting to control automatically the pulsing frequency.

Advanced Process Control Development

Programming of the GE-412 computer for on-line control of the experimental C-Column continued. Fifteen major sections of the program have been defined. These sections range from the executive control program which governs the time-sharing of the other functional programs, to the highly complex process optimization program which calculates controller set points using a mathematical model of the solvent extraction process. Seven of the sections are based on the use of the generalized monitor program provided by the vendor as part of the computer software package. All of the applicable monitor programs have been coded for the specific machine on order for Hanford Laboratories. Another program for which coding was completed provides for communication, via punched paper tape, between the data logging absorptiometer and the computer. This program, designated ABSDAT, enables determination of aqueous and organic uranium concentrations and extractant pH.

ANALYTICAL AND INSTRUMENTAL CHEMISTRY

Burn-Up Analysis

The previously discussed problems attendant to burn-up analysis seem to have been solved. Results obtained during the last month were internally

[redacted]

[redacted]

consistent. Even the earlier discrepancies between coulometric and α-counting analysis of Pu are no longer apparent, although the reason for the difference is not known. Attention is now being given to a procedure for burn-up analysis of mixed oxide fuel elements.

Thallium Determination by Flame Photometry

Thallium at the ppm level in nitrate solution was determined accurately by flame photometry. The metal was extracted from slightly basic solution by a 0.1 percent dithizone solution in chloroform. Directly aspirating the organic solution into the flame and observing thallium transmission at 378 mμ gave reproducible calibration curves representing 2 to 20 ppm, a much more sensitive and accurate thallium determination than standard analytical technology provides.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Chloride Contamination in UO_2 - It has been previously reported that the chloride contamination in UO_2 electrolytically deposited from LiCl-KCl melts under pilot plant conditions was suspected to result from the presence of certain metallic impurities in the melt. This observation aroused concern that the presence of fission products in the melt at normal operating concentrations might cause an increase in chloride contamination of the product UO_2. In an initial test, it was found that MoO_3 and Zr_2O_3 added to the melt in amounts proportional to those of an irradiated fuel (10,000 MWD/T) did not increase the chloride contamination in the UO_2.

Chloride Content of UO_2-ThO_2 Codeposited from Molten Chloride Salt Solutions - In the course of studying the recovery of uranium from 3 percent UO_2 - 97 percent ThO_2, deposits of UO_2-ThO_2 solid solutions have been obtained ranging in Th/U ratio from 1.8 to 0.64. The chloride impurity in the washed deposits varied from 1200 ppm to 36 ppm, the deposits containing less thorium generally having less chloride impurity. These results encourage further investigation of UO_2-ThO_2 recovery from molten chloride salt solutions.

U(V) in Fused LiCl-KCl Solutions of U(VI)-U(IV) - Spectrophotometric studies in molten LiCl-KCl containing 15 w/o U(VI) plus small amounts of U(IV) have resulted in the observation of a uranium species not previously seen in molten chloride salt solutions. From the information obtained to date, this species is tentatively concluded to contain singly-oxygenated uranium(V), e.g., $UOCl_x^{+3-x}$, produced via a reaction such as:

$$UO_2Cl_2 + UCl_4 = 2UOCl_3.$$

This tentative conclusion is based on the facts that:

(1) The species is not present in the absence of uranium(IV).

(2) The concentration of the species varies with the square root of the uranium(IV) concentration, as expected for an equilibrium as given above.

Hot Cell Runs - Dissolution of irradiated mixed plutonium-uranium oxides with HCl and chlorine gas at 600 C was continued. The three major volatile fission products were ruthenium, Zr-Nb-95, and Sb-125. Ruthenium volatilized while the melt was sparged with chlorine gas but stopped immediately when HCl-Cl_2 gas mixtures were introduced. Zirconium-niobium were detected only when [redacted] not present.

[redacted] DECLASSIFIED [redacted]

Three electrolyses were made with irradiated feed materials. The first, made with an HCl sparge, was designed to partition the uranium from the plutonium. The second and third electrolyses were codepositions made with 90 percent oxygen - 10 percent chlorine sparges. Complete data are not yet available on the results of these runs.

While dissolving for the next run the fused silica pot holding the molten salt was broken and the equipment was shut down. The cause of the pot breakage has not been determined, although the molten salt and fission products were confined within the metal susceptor for several hours until the salt was frozen.

<u>Cold Pilot Plant Runs</u> - Experiments using unirradiated uranium feed were conducted in support of the current hot cell effort with identical equipment. Efforts were directed toward improving the salt pot lid seal and uranium and plutonium dissolution using cerium as a stand-in for plutonium.

A graphite sleeve was used to line the fused silica pot in the area above the molten salt where the PuO_2 precipitate had been observed to adhere to the walls. Essentially complete dissolution of cerium added as $CeCl_3$ x $7H_2O$ was obtained with no CeO_2 precipitated on the pot wall. All of the cerium remained in solution over a period of 96 hours with no gas sparge in the molten salt.

The retaining lip on the graphite sleeve was eroded entirely away by air in-leakage. Additional severe general corrosion of the entire sleeve was noted. Additional gasketing is expected to reduce this problem.

Dissolution of 15 lb. of U_3O_8 was achieved in 7 hours at 600 C with a gas lift recirculator. During the first five hours, 6 ℓ/minute of Cl_2 was used. During the final two hours 2 ℓ/minute Cl_2 and 2 ℓ/minute HCl was used. The dissolution rate represents about a five-fold reduction in the time cycle previously experienced in the hot cell experiments.

<u>Chlorine Liquefaction Studies</u> - a 30-hour U_3O_8 dissolution run was made at 600 C with the off-gas passing through a low-temperature (-100 C) trap for removal of the excess chlorine by liquefaction. Prior to entering the low temperature trap, the gas was cooled to room temperature and dried by passing through a magnesium perchlorate packed column. Recovery was 93 percent corresponding to the chlorine vapor pressure at about -100 C. Recovery of chlorine evolved during electrolysis was demonstrated in a second run, but plugging by solid chlorine was encountered preventing a fully successful experiment.

Dissolution of PRTR UO_2-PuO_2 Fuels

Zirflex process decladding and core dissolution experiments were performed with 0.75-in. long sections cut from unirradiated PRTR rod CD-91. This swaged rod was incrementally loaded with PuO_2 and UO_2; the core contained 0.45 percent PuO_2.

Zircaloy cladding was readily removed by boiling two hours in a solution initially 2.0 M NH_4F-0.17 M NH_4NO_3. The core crumbled to a "mud" during decladding and a considerable amount of a green solid (UF_4?) formed. The unreacted UO_2 dissolved readily on subsequent treatment at 80 C with 11.1 M HNO_3-0.1 M NH_4F-0.0085 M NH_4NO_3 solution. The green residue dissolved only slowly, however, and a significant amount remained undissolved after seven hours at 80 C. At this time about 94 percent of the uranium and 30 percent of the plutonium were in solution.

In a second experiment, after exposure of the declad core for two hours at 80 C to 11.1 M HNO_3-0.1 M NH_4F-0.0085 M NH_4NO_3 solution, sufficient $Al(NO_3)_3$ was added to make the dissolver solution 0.3 M Al. The resulting solution was boiled one hour. This procedure resulted in complete core dissolution and is being studied further as a possible satisfactory dissolution scheme for irradiated PRTR fuel.

RADIOACTIVE RESIDUE PROCESSING DEVELOPMENT

Cold Semiworks Spray Calciner

Water and feed (Purex 1WW) capacity data were obtained for design of the Waste Solidification Engineering Prototype spray calciner. A 14-in. diameter, 6-ft. long reactor, and a 10-1/2-in. by 6-ft. draft tube, both fabricated from thin sheet metal, were inserted into the 18-in. spray calciner, with Spraying Systems Co. No. 72 internal-mix nozzle, 60 psig atomizing steam, and 700 C reactor wall temperature. The observed water capacity was 21 gph and the feed capacity was 14 gph. These agreed closely with predicted capacities of 22 gph and 16 gph, respectively.

Nozzle effects on spray calciner operations were studied by using an ultrasonic atomizing nozzle (Astrosonics, Inc. No. 1407) in the 18-in. calciner without a draft tube. The water capacity was in excess of 40 gph at a furnace temperature of 700 C. The highest water capacity previously observed was 25 gph, with an 8-ft. draft tube and a Spray Systems Co. No. 72 internal-mix nozzle. However, application of the ultrasonic nozzle to spray calciner work does not appear promising with Redox borate-phosphate feed and Purex 1WW mostly because of plugging of the 0.040-in. diameter nozzle orifices and cake buildup on the nozzle. Redox borate-phosphate product was delivered in one run but nozzle distribution was apparently poor. The product powder was coarse and contained many pea-size lumps. The water content was 7.3 w/o and the nitrate content was very low.

Calcine Melter

Inspection of the Inconel 600 melter used for about 135 hours in previous melter studies revealed a 1/8-in. diameter penetration through the wall near a weld seam at the normal melt surface.

Laboratory corrosion studies to date have shown 310 SS to be superior to Inconel 600 for several Redox-borate-phosphate melts. A new 310 SS melter is being designed and fabricated for direct coupling studies using the 18-in. spray calciner as the powder feeder.

Calcined Waste Melt Pot Materials

A 310 stainless steel crucible was exposed to calcined simulated Purex process waste with 50 percent phosphate addition at 900 C for 28 hours. The crucible wall was completely penetrated at several points corresponding to a penetration rate greater than one mil per hour. A type 310 stainless steel crucible exposed to a melt of simulated Redox process waste with phosphate and borax additions showed excellent corrosion resistance at both 750 and 900 C.

Measurement of Glass Viscosity

A technique has been developed for measurement of glass viscosity at elevated temperatures. The falling-ball method is used, in which the time for a platinum-iridium sphere to fall a given distance through the glass is measured electronically. The method yielded reproducible results during calibration runs; accuracy is expected to be within ± 10 percent. Viscosities of a phosphate glass from Brookhaven ranged from 27.5 to 10.4 poises over a temperature range of 1076 to 1173 C.

Available test equipment enables measurements up to about 1400 C over an estimated viscosity range of 2 to 2000 poise.

Radioactive Glass Preparation

Several samples of phosphate glasses, "doped" with Ce-144 to simulate power reactor power levels, were prepared in A-Cell by use of crucible techniques. These samples will be used to determine the effects of intense radiation and isotopic decay on storage stability and to help answer the question of whether microcrystalline solids or true glasses are the more suitable for long-term storage.

The cerium, as received from the Strontium Semiworks, contained a large quantity of sodium. It was further purified by precipitation as the oxalate, followed by filtration and dissolution in nitric acid. The solution was mixed with glass-forming ingredients, and the radioactive decay heat of the 13 kilocuries of cerium (25 grams) used to evaporate the mixture to dryness. The powder was then melted in crucibles in a small furnace. Gross appearance and dissolution rate of the samples will be observed over a period of time.

In supporting laboratory work, phosphate was found to be lost from glass formulations to a greater extent than expected. In one case, for example, a make-up composition of 55.4 weight percent P_2O_5 (intended to form metaphosphates of the metal ions) was reduced to 34 percent by spray drying and to 28 percent on subsequent melting. Further study will be required to determine whether the amount of phosphate initially included can be safely reduced and, also, whether there is further phosphate loss when a phosphate glass is held at high temperature for extended periods of time.

Hot Cell Glass Experiment

The BNL equipment for the hot-cell testing of the BNL continuous phosphate glass process was assembled during the month (with the assistance of George Weth of BNL) in the 325-A mock-up area for cold testing and "de-bugging." Several shakedown runs (made with a "cold" feed simulating the 1WW which will be used in the hot cell program) revealed a number of design and fabrication deficiencies which will require correction prior to hot operation; however, glass was made, and cesium leaching tests show it to be of fairly low solubility, ca. 10^{-3} g/cm^2 day. Major difficulties included a prolonged and difficult start-up (possibly because the cullet normally used in BNL start-up was not available); a concentrator which was unable to accommodate the scheduled feed rate, and which cracked in operation (a stainless steel coiled tube concentrator is being fabricated to replace this component); failure of the Vycor feeder head, which cracked on cooling; and fogging in the off-gas system, coupled with cross-flow between the nitric and sulfuric branches (resulting in carry-over of sulfuric acid into the nitric acid system).

Intermediate-Level Waste Treatment

Electrodeionization experiments were carried out in an electrodialysis cell to which mixed ion exchangers (Dowex 1 plus Dowex 50) were added. Purex acid condensate ($\sim$ 0.01 N HNO_3) was passed through the cell at a flow rate of 0.2 column volumes per minute. Ruthenium decontamination factors greater than 25 were obtained after passage of 875 column volumes of condensate, whereas decontamination factors of 2.5 were obtained after only 25 column volumes with ion exchange alone.

Columbia River Sediment Study

Radionuclide analysis of sized fractions of Columbia River bottom sediments indicates that the clay particles (less than 5 μ diameter) contain a much higher concentration than the coarser fractions. The coarsest fraction does contain a relatively high concentration, however. An experiment in which the organic matter in the sediment was destroyed with hydrogen peroxide before particle size separation indicated that the radionuclides in the coarsest fraction were almost completely associated with organic matter. Particle size separation was much easier after hydrogen peroxide treatment, indicating that the particles are cemented with organic matter.

CONTAINMENT SYSTEMS EXPERIMENT

Provision of Facility

The bid packages for the containment vessel and the simulator (long-term procurement items) were completed during the month. The packages reflect suggestions and ideas of several Hanford Laboratories organizations as well as those of the Atomic Energy Commission. The criteria for containment concepts such as the multiple barrier and ducted pressure relief are not provided at this time; however, provisions were made in the containment vessel so the necessary equipment can be installed at a later date.

Work was started also on the instrumentation required for the measurement of flow rates and the stresses that will occur in the system. Two problems are evident. One is the measurement of the mass flow rate from a nozzle. The other is the attachment of devices within the simulator vessel in a manner that they will not be destroyed during each blowdown but yet give accurate readings.

Test Program Planning

Work on a detailed test program was started. Initial attention is being given to the engineering tests which are necessary to shake down and characterize the facility. Attention will then be given to the experiments involving fission products or tracers.

Fission Product Simulation

Experiments are being planned to compare the deposition behavior of real fission products with simulants. Equipment will be assembled and readied so that tests can start by about July 1, UO_2 fuels irradiated to low levels will be heated to generate the real fission product aerosol. Several methods of simulating a fission product aerosol will be tested, including stable iodine incorporated in a UO_2 matrix.

Iodine Generation

Tests to date on the injection of elemental iodine vapors into a vessel show that the passage of 0.5 cfm of 130 C air over iodine crystals can vaporize

$\sim$ 2 grams per minute. This iodine-laden air (about 10 percent of saturation) was passed through 20 feet of 3/4-in. SS tubing at 150 C to scrubbers. Deposition at this temperature in the tube was not more than 18 percent of the total and probably much less. Some localized discoloration was noted in the tube. No serious attack of the metal was evident.

The transport of high concentrations of iodine vapors in air seems feasible although it may prove difficult to obtain high percentages of saturation in a simple manner.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

Comparisons of numerically-determined flow paths with the mathematically closed-form solution gave very good agreement. The improved numerical scheme uses two integrals evaluated twice, rather than iteration with the original pair of difference equations. In addition to the improved accuracy achieved, less computer time is required for the new integral determination of flow paths. Initial results scaled up to field situations indicate maximum deviations in flow paths, due to numerical approximation, of from 0.5 to 3 feet per mile of flow length. Work is continuing on getting more results on this error, as well as the surface fitting error, utilizing the GENORO program and the permeability integration errors. These results will be the basis for a complete error analysis of the permeability determination method.

The program for calculating transient, unsaturated, horizontal (one-dimensional) flow in soils was completed. The method used, called "Phillips Iterative Procedure," requires a moisture content vs. capillary pressure relationship as input. Results include the moisture content distribution for a series of input times and the cumulative inflow for the various times. A series of 27 cases was run for comparison with actual soil column data. The cases included varying the boundary conditions for several of the columns. Expansion of the program into one that will analyze vertical flow is underway.

A boundary value problem, patterned on the 101-SX tank as a model, was solved for the steady-state heat flow from a salt cake. The problem was solved using a modification of the steady-state fluid flow program. Results are being checked to determine their agreement with the model.

The laboratory soil column density-moisture measurement equipment was assembled and tested. Fabrication of the column support and source-detector slide was completed. The system uses gamma photon attenuation to measure soil density or moisture content and consists of a soil column support, Am-241 source and holder, photomultiplier tube and holder, and associated counting equipment. The system should permit measurement of soil density and moisture content with an accuracy of about ± 1 percent.

Water level data from piezometers installed in wells in the south and southeastern part of the Hanford Works area disclose that the upper Beverly interbed (contaminated by Separations areas wastes north of 300 Area) evidently is naturally recharged by the Yakima River near Horn

Rapids and from the Rattlesnake Hills in the Dry Creek Valley area. Recharge also takes place at the site of the Purex cooling water groundwater mound. The flow path of contaminants in the confined interbed will be dependent upon the amounts of water entering the confined aquifer in the three sites and the respective gradients.

Long-term records of water levels in wells tapping the upper Beverly interbed were examined. Piezometric heads so measured record a rise of about 0.2 feet per year for up to 15 years. Close to the Columbia River, in wells 699-10-E12, 699-15-15 and 699-20-E12, the rise as measured through piezometers installed in 1963 is at a rate of about one foot per year. This probably is due dominantly to recharge at the Purex swamp site. However, well 199-H4-2 shows a comparable rise since 1958, following an earlier drop from completion in 1952, that corresponds to rises in the piezometric heads in wells east of the Columbia River in the Irrigation Project area. The rises in wells 699-10-E12 and 699-20-E12 may in part be the response to rises of 100 to 200 feet in the Ringold Coulee area (as reported by the U.S. Geological Survey) and recharge there of the upper Beverly interbed. Under such conditions, flow beneath the Columbia River is a possibility with consequent effect on flow paths of contaminants in the aquifer.

RADIOLOGICAL AND HEALTH CHEMISTRY

Uranium Ore Inhalation Studies

Analyses of uranium and thorium content of over 50 rat lungs have been completed in a joint study with the Pharmacology Operation. Lungs from rats which were exposed three times at weekly intervals to uranium ore dust and subsequently sacrificed at various times up to eight weeks show thorium to uranium activity ratios which vary from 1.5 to 2.7. Since the ore used was in radioactive equilibrium, with a thorium to uranium ratio of unity, this indicates a significantly greater retention of thorium than of uranium. Analyses are now in progress on the kidneys and bone from these rats to determine the distribution ratios in these tissues.

Analytical Procedures

In the development of a procedure for the determination of Sr-90-Y-90 in hair it has been found that oxidation of hair in the Tracerlab Low Temperature Asher converts the sulfur in the hair to the sulfate form. This necessitates a sodium carbonate fusion to remove the sulfate and to put the strontium in an acid soluble form. Addition of oxalate at pH 3-4 precipitates the strontium and calcium, leaving the complexed iron in solution. The oxalate precipitates are dried, ignited to the oxides, dissolved in dilute acid, and finally precipitated with ammonium carbonate. A yield of 83 percent is obtained.

An improvement was made in the activation analysis procedure for the determination of thorium in biological materials. The procedure formerly used di-isobutyl carbinol to extract the Pa-233. By extracting with a tertiary amine in xylene a simpler, shorter and more selective procedure is possible.

Radioanalytical Procedure Evaluation

The Health and Safety Laboratory fluorometric procedure for the determination of uranium in urine was evaluated and found to be completely satisfactory. A detection limit of 2 μg/l was estimated. It was found that some improvement was gained by transferring the flux bead to a blackened platinum dish which provided a reduction of blank fluorescence by a factor of two.

Tracers for Atmospheric Dispersion Tests

Rhodamine B dye was found to have suitable fluorescence emission properties to be easily excited and measured in the presence of fluorescein dye. These two dyes may therefore be useful in the dual tracer tests of the Atmospheric Physics Operation. By exciting at 578 mμ and measuring the 590 mμ emission of Rhodamine B a sensitivity of 1×10^{-10} g was obtained without affecting the 5×10^{-11} g sensitivity of the fluorescein analysis. The system has not yet been field tested.

Radiation Chemistry

A study was made to compare the effect of ionizing radiation on solutes in liquid aqueous solution with the effect in frozen aqueous solution. The effect was determined by measuring the bleaching of erioglaucine dye in liquid solution at 25 C and in solid frozen solution at -7 C. Comparison of the initial slopes of the bleaching curves indicates that the dye is 150 times more sensitive in the liquid solution state than in the frozen state. Lower temperatures thus might be useful in depressing the radiation sensitivity of important constituents.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Aerosol Sampling Study

The inaccuracy in sampling particles in a wind of 2.7 miles per hour when sampling rates much below isokinetic are used was measured for various particle sizes. Sampling rates of 9 percent, 60 percent and 80 percent of isokinetic flow were used. The data obtained along with earlier data taken at 16 percent and 30 percent of isokinetic cover the anticipated range of subisokinetic flows for 2.7 mph windspeed.

Five- and ten-micron particles are collected as progressively higher fractions of the isokinetic sample as the sampling rate approaches the isokinetic. For 15-micron and larger particles a smaller fraction of the isokinetic sample is collected at 2.7 cfm than at 2 cfm, which is an anomaly yet to be explained.

Recounting and sizing a greater fraction of particles on the filters is being undertaken to resolve some of the anomalous values obtained to date. Initial results of these redeterminations show that improved statistics will improve the coherence and consistency of the data.

Fallout Studies

Sodium-22, which was indicated to have a fallout origin in the February monthly report, seems to originate mainly as a result of cosmic ray spallation reactions on stratospheric argon. Study of its distribution and its relationships to Cs-137 may give information regarding atmospheric circulation behavior. Measurements of Na-22 and Cs-137 in local beef and in Alaskan caribou compared with the concentrations in urine determined in February show that the Na-22 concentration is 30 times higher in urine than in meat relative to the Cs-137 concentrations. Thus meat is not the only, and probably not the major, source of Na-22. Air concentrations indicate that inhalation is an insignificant source. Milk is suspected and is being tested as a probable major source.

ISOTOPES DEVELOPMENT - 08 PROGRAM

Transmitting Wattmeter for Calorimetry Studies

A transmitting wattmeter was developed to indicate and control the electric power input to a simulated isotopic heat source for use in calorimetry measurements. The wattmeter employs a Hall-effect device, which generates an output voltage proportional to the product of input current and magnetic flux. Input current was obtained from the heater voltage and input magnetic flux from the heater current passed through a coil. Open circuit Hall output voltage was slightly non-linear when plotted against heater power, but was made linear within 2 percent by the addition of a resistor across the output terminals. Fabrication of additional test equipment for calorimetry of isotopic power sources continued.

Acid-Side Promethium Purification Processes

Four runs, two with DTPA and two with HEDTA, were completed during the month. This completes investigation of the effects of concentration and pH. Following a few runs in which temperature and length of resin bed are varied, a report will be issued.

[redacted]

Isotopic Source Development

No new compactions were performed during the month, but several made last month were cut open and inspected. In addition, receipt of a high temperature furnace will make possible production of neodymium and samarium metals (stand-ins for promethium metal) by reduction of the respective fluorides with calcium. Preparation of several potentially interesting compounds will also be facilitated.

Fission Product Computations

A computer code (ISOGEN, for isotopes generation) has been written which will greatly facilitate the computation of the quantity and activity of nuclides in decay chains up to ten nuclides long and with up to 30 different time steps. Accuracy of the program is within 0.1 percent for all calculations checked thus far. Major advantages of the new program are that it will yield accurate data even if (1) two or more of the nuclides have identical decay constants, (2) branching may take place at any place in the chain, and (3) input and read-out data are in familiar units (seconds, minutes, hours, days, and years) rather than all being in seconds. Planned modifications to include use of the Westcott notation for cross sections and provision for constant fission rate calculations with weighted average cross sections will make the program even more versatile.

Another program (BREMRAD, for bremsstrahlung radiation) has been written (with the assistance of the Electronic Data Processing Operation) for the computation of bremsstrahlung radiation. It consists of two parts: BREMEX (for external bremsstrahlung) and BREMIN (for internal bremsstrahlung).

M. T. Walling, Jr.
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Chemical Laboratory

MT Walling:cf

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BIOLOGY LABORATORY

A. ORGANIZATION AND PERSONNEL

No significant changes occurred.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Reactor Effluent Monitoring

The 1964 reactor effluent monitoring at KE with chinook salmon will start in April. For the test about 5,000 chinook fry were transported here from the new Washington State Department of Fisheries salmon hatchery at Priest Rapids. These fish closely represent the local fall chinook stock which spawn in the Hanford Reservation. Aside from the usual monitoring aspect, the 1964 test is designed to provide some measure on the interaction of temperature and chemical toxicity and to test the biological effects, if any, of the new zirconium tubes of KE reactor.

Powdex Ion Exchange Resin

Powdex anion resin (hydroxide form) and powdex cation resin (ammonium form) scheduled for use at NPR, both produced 100% mortality in chinook salmon fingerling at 1000 ppm within 24 hours. No mortality occurred in either resin at 560 ppm in 72 hours. At 750 ppm the anionic resin killed 10% of the test fish and the cationic resin killed 70% during the 72-hour test.

Columnaris

In two fish which had gone through the course of infection by and recovery from columnaris, there was recurrence of the infection cycle within two weeks of the time of apparent end of the first cycle. The second cycle was initiated by placing these fish in the same trough with other fish known to be infected.

The importance of duration of low temperature to the wintering over of columnaris in trout is emphasized by the rapidity with which heavy infection was re-established in fish held at normal river temperatures during early winter and then elevated temperatures during the later part of the winter. In these year-old fish the deaths which occur appear to be due to the fish disease Furunculosis, not to columnaris, in spite of the obvious presence of the latter.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY AND TRANSFER OF RADIOACTIVE MATERIALS

Zinc

The relatively high Zn^{65} activity in the GI tract of trout after a single oral dose of 200 μc persists. Five fish killed on 299 days post-administration showed on the average a body burden of 7 μc of which 3.9 μc is found in the GI tract. The concentration of the partitions of the GI tract in descending order of activity were 0.39 μc, 0.28 μc, 0.23 μc and 0.16 μc/g for mid-gut, pyloric caeca, hind-gut and stomach, respectively. Eleven weeks of chronic feeding of Zn^{65} to sibling rainbow trout have not produced any readily observable effects. At the end of 11 weeks each fish in the three treatment levels of 1.0, 0.1, and 0.01 μc Zn^{65}/g fish received a total of 8.8, 0.85, and 0.09 mc, respectively. Three fish receiving 0.1 μc/day that were sampled during the fifth week had body burdens of about 93 μc Zn^{65} or 42% of the administered dose.

Strontium

A three-year-old miniature pig from the 125 μc Sr^{90}/day level (with a skeletal Sr^{90} burden of ~1200 μc or 600 times the body burden MPL for industrial workers) developed a clinical picture and was shown to have gross pathological lesions suggestive of a neoplastic process. The animal had normal hematological values and appeared clinically normal approximately three months before death. Two weeks prior to death, the animal lost its desire for food; however, the animal's blood retained near normal values except for a moderate neutropenia and thrombocytopenia and the appearance of a number of bizarre-looking mononuclear cells. During the last two weeks of life, the animal developed a severe anemia, and platelet numbers decreased to very low levels. The bizarre mononuclear cells continued to be present in moderate numbers.

In view of the animal's deteriorating condition and anticipated death, she was killed. The most significant gross change was an enlargement of the spleen and lymph nodes, with many of the latter having areas of hemorrhage. There was no evidence of any massive hemorrhage to account for the severe anemia.

On the basis of the hematological changes and gross pathology, the condition was tentatively diagnosed as a lymphoma. Final diagnosis will have to await the microscopic examination of tissues now being processed. Similar cases have not been observed in our control animals; however, considering this is only one case, a causal relationship to the Sr^{90} burden would be presumptive.

In cooperation with the Radiological Chemistry Operation, a single miniature swine was administered Sr^{85} intravenously in an attempt to produce an appreciable skeletal burden of Sr^{85} and, more importantly, to result in a measurable quantity of Sr^{85} in the animal's hair. Hair samples from this animal were given to the Radiological Chemistry Operation, for use in standardization of their procedure for detecting Sr in hair.

Iodine

Three young miniature swine were exposed to I^{131} vapor. Initial skin deposition varied from 30 to 70 μc on an area of 155-160 cm^2. Thyroid uptake, which is barely detectable, is being monitored daily and evaluated. Skin thickness at the exposure sites is being measured.

Neptunium

The sixth group of sheep in the Np^{237} toxicity study has been completed, and we are awaiting results of radioanalysis and histopathology. We plan to test one more control group in the near future, thereby completing this study in which progressive biochemical, histopathological and clinical changes are simultaneously evaluated in an animal administered a liver toxin.

The incorporation of DL-leucine into liver protein by control and Np^{237}-treated female rats was investigated using leucine. Preliminary results indicate that Np^{237} increases the amount of leucine incorporated into liver protein.

Plutonium

Intact or splenectomized rats showed the same reaction to acutely toxic doses of either Pu^{239} or Pu^{238}. In the case of Pu^{239}, all animals died within 38 days following injection of 71-75 $\mu c/kg$. In the case of Pu^{238}, there were no deaths within 38 days following injection of from 95-100 $\mu c/kg$. This would seem to eliminate the spleen as a highly significant factor in explaining the differences in acute toxicity of Pu^{238} and Pu^{239}.

Studies with DTPA-perfused gut segments suggest that a certain minimum concentration of chelating agent must be maintained before increased biliary secretion of plutonium will occur. Thus, DTPA administered to the lower third of the jejunum or to the stomach is ineffective at 200 mg doses, but effective at 400 mg doses. This may, in part, explain the usually unsatisfactory results obtained from oral therapy in humans, where dose levels have always been low.

Inhalation Studies

The incorporation of C^{14}-labeled acetate in fatty acids was studied in intracellular fractions of rat lung homogenates. Preliminary data show that in the absence of added carbonate, long-chain fatty acids are synthesized almost exclusively by the mitochondrial fraction. Additional experiments are planned to substantiate these findings and to explore the role of various intermediates known to be involved in the synthesis of fatty acids in lung tissue. These studies are expected to contribute to our knowledge of the basic biochemical mechanisms in the lung which is necessary to understand the behavior and biological effects of radioactive particles deposited in the lung.

A dog sacrificed one and one-half years after deposition of 90 µc $Ce^{144}O_2$ showed only slight to moderate fibrosis in the lungs. These changes are considerably less than those seen in dogs after deposition of similar quantities of $Pu^{239}O_2$. Analyses of tissues for $Ce-Pr^{144}$ are not complete.

One dog died three and three-fourth years after inhalation of $Pu^{239}O_2$. The body burden just prior to death was estimated at 2 µc using the counting equipment under development by Radiological Physics Operation. Clinical signs included increased respiration rate, lymphopenia, body weight loss, anorexia, decreased arterial blood oxygen, and hemoglobin saturation. Thoracic X rays showed cardiac enlargement and increased opacity of the lungs, especially the anterior dorsal part of the right diaphragmatic lobe. At necropsy the lungs were congested, edematous, and fibrotic. The anterior dorsal part of the right diaphragmatic lobe was consolidated. The bronchial and mediastinal lymph nodes were larger and less dense than in dogs that died earlier and contained pin-point white nodules in the sub-surface. Lesions were seen in the heart, kidney, aorta, and brain which were probably agenal in nature at time of death.

Six of the remaining 11 dogs on this experiment (about four years post-exposure) show increased respiratory rates, radiographic changes and lymphopenia. Three dogs show only lymphopenia, and two appear normal.

All dogs on the long-term plutonium study are being caged periodically for collection of urine and feces which are being analyzed for Pu^{239} by Internal Dosimetry Operation. Levels of Pu^{239} in the urine and feces of these dogs are so low as to be undetectable by our method of analysis. All of these dogs will also be monitored in the Pu^{239} counter being develcped by Radiological Physics Operation in the 300 Area.

Because of the high rate of pulmonary lesions in rats we are investigating the possible use of hamsters for inhalation studies. They are apparently free of chronic respiratory disease and, unlike rats, maintain a body size that will facilitate their being exposed in our aerosol exposure apparatus throughout their two and one-half year life span.

Due to crowded conditions in the colony, one long-term plutonium dog was accidently bred and delivered five puppies. One pup will be sacrificed for plutonium analysis and histological study.

To be assured of experimental animals during fiscal year 1966, the beagle dog breeding program was revised despite the hold-up in construction of facilities. Sixty puppies are expected by July 1, 1964. To house the expanded colony temporary runs and dog houses are being constructed on the concrete pad to the east of 144-F building.

Co-carcinogenesis Studies

Histological data are being evaluated from the study of the effects of internal emitters on hepatic tumor induction by dimethylaminoazobenzene (DMAB) in rats. There seems to be a rather clear indication that Pu^{238} and Ce^{144}, at sufficiently high deposition levels, significantly decrease the incidence of tumor induction.

Gastrointestinal Radiation Injury

Three days after a 1,500 r X-ray dose to the abdomen of rats, the absorption of C^{14}-labeled taurocholate (a bile salt) from the intestinal tract was reduced by 55% as compared with unirradiated controls. This reduction was evident during the first hour, but was subsequently masked by continuing absorption during later periods. It would seem that active transport of bile salts may be blocked by irradiation but passive absorption may continue at a reduced rate. This may explain the enhanced effect of bile salts in producing the diarrhea following irradiation of the intestinal tract.

Intraduodenal administration of neomycin and phthalylsulfathiazole for three days prior to and following abdominal irradiation of 1,000 and 1,400 r failed to prevent diarrhea and death, indicating that bacteria sensitive to this mixture of antibiotics are not involved in the intestinal radiation syndrome.

Effects of X rays on Cichlids

To provide more information on the response of adult cichlids, Aequidens portalegrensis, to X rays, three groups with 25 fish each were exposed in early March to 3,000, 2,000, and 1,000 r. The 3,000 r group suffered initial mortality on the ninth day post-exposure; 50% were dead on the 14th day; and all fish were dead on the 18th day. The 2,000 r group had one mortality on the 20th day. Eye damage was apparent in both the 3,000 and 2,000 r groups. Excessive melanophores of the choroid appeared in one or both eyes. Anorexia appeared early in the 3,000 r group and the 2,000 r group indicate some growth depression. Observations continue, but no apparent difference exists between the 1,000 r and control groups.

RNA as a Radiation Protective Agent

Further experiments were performed to test the protective effect of "highly polymerized" RNA. The results were inconclusive due to unexpectedly high survival of controls, small numbers of animals involved, etc. There was, however, a clear indication that RNA plus bone marrow cells were decidedly more effective than RNA alone.

Clinical Studies

Two ewes manifesting symptoms of ketosis were subjected to the rose bengal I^{131} liver function test. A definite decrease in plasma clearance rate was seen compared to normal appearing ewes of about the same age and stage of gestation.

Microbiology

In a Neurospora strain having a genetic block in tryptophan synthesis at a point prior to the tryptophan cycle, there was a fivefold increase in protein synthesis after disappearance of all external tryptophan. This increase was accompanied by a fourfold decrease in specific activity of the C^{14}-labeled tryptophan in the protein. Such loss of activity was the consequence of cycling of the tryptophan through a metabolic conversion sequence, known as the tryptophan cycle, and re-entering the tryptophan pool as metabolic rather than externally derived tryptophan. In these tests carried out at 22 C, the specific activity of tryptophan in the cell pool remained much higher than the specific activity of the tryptophan being synthesized into protein. This difference in specific activities is distinct evidence of differential between a metabolically derived molecule and the same molecule introduced into the cell without benefit of any metabolic conversions. The extent of this differentiation, or channeling, is much greater at 22 C than at 32 C, which was the temperature used in previous tests.

By observing rates of growth and of induction of the enzyme tryptophanase during changes in temperature with E. coli, it was found that differential enzyme induction rate was very high when temperature was changed from 37 to 25 C, but virtually ceased upon the reverse temperature change. A working hypothesis is that growth rate affects the availability of enzyme inducer.

Plant Nutrition

Pea roots grown in various concentrations of iodide were examined for frequency of division figures. The mitotic index rises slightly as iodide increases from 0 to 0.6 µg/ml. This may be related to the previous observation of increased percent of iodide taken up by plants as substrate iodide increases.

Attempts to amplify an autoradiographic image by "controlled Townsend avalanche" have as yet been unsuccessful.

Columbia River Limnology

Periphyton production on standard substrates increased three- to fourfold over February in spite of a lack of temperature rise. These data support the hypothesis that effective light penetration is of more importance to growth of Columbia River algae than is water temperature.

Terrestrial Ecology

Depth of moisture penetration for the 1963-64 season is 0.5 meters, or about half that observed in the previous year.

Counting data from the year's sampling in Alaska became available and is being compiled. It appears that caribou from Anaktuvuk Pass have a slightly lower concentration (100 pc/g dry weight) of Cs^{137} than those taken from the Fairbanks area (140 pc/g dry weight). Highest Cs^{137} concentration (89 pc/g dry weight) observed in plants was from water lilies taken in the Kobuk River area. Concentrations in lichens ranged from 15 to 50 pc/g dry weight.

Relating of observed concentrations with environmental conditions and food base will continue for some time in order to assess the meaning of these data.

HA Kornberg
Manager
BIOLOGY LABORATORY

HA Kornberg:es

TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORY

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposia

Hanson, W. C. The accumulation of fallout cesium-137 in northern Alaskan natives. Wildlife Conference. Las Vegas, Nevada. March 9, 1964.

Eberhardt, L. L. Sampling ecosystems for fallout radionuclides. Wildlife Conference. Las Vegas, Nevada. March 9, 1964.

Kornberg, H. A. A comparison of approaches to radiation and general toxicology. Society of Toxicology. Williamsburg, Va. March 10, 1964.

Mahlum, D. D. Effect of Np^{237} on lipid metabolism. Society of Toxicology. Williamsburg, Va. March 10, 1964.

McClellan, R. O. Acute toxicity of Sr^{90} in miniature swine. Society of Toxicology. Williamsburg, Va. March 10, 1964.

Stuart, B. O. Distribution and excretion of inhaled and intravenously administered Pm^{147}. March 10, 1964.

Dean, J. M. Instrumentation in Biology at Hanford. Pacific Fishery Biologists Annual Meeting, Ocean Shores, Washington. March 25-27, 1964.

Nakatani, R. E. Chairman of Panel on Instrumentation in Fisheries. Pacific Fishery Biologists Annual Meeting, Ocean Shores, Washington.

b. Seminars (Off-Site and Local)

Park, J. F. Toxicity of inhaled plutonium. 6516 U.S. Army Reserve and Development Unit, Pasco, Washington. March 2, 1964.

O'Brien, R. T. Physiological effects of heavy water on microorganisms. Exchange Seminar Program. Washington State University, Pullman, Washington. March 6, 1964.

Dean, J. M. Effects of temperature acclimation on some aspects of carbohydrate metabolism of Crustacea. Exchange Seminar Program. Washington State University, Pullman, Washington. March 23, 1964.

McClellan, R. O. Some measurements of damage in miniature swine ingesting Sr^{90}. Industrial Physicians Meeting, March 19, 1964 - Richland, Washington.

McClellan, R. O. Some hematological effects of daily ingestion of Sr^{90} in miniature swine. Department of Medicine, University of Washington, Seattle, Wash. March 18, 1964.

c. Seminars (Biology)

Clarke, W. J. Bronchiolo-alveolar tumors of the canine lung following plutonium particle inhalation. March 4, 1964.

Sullivan, M. F. Biological effects of neutron irradiation. March 11, 1964.

Hecht, A., Department of Botany, Washington State University, Pullman, Washington. Experimental modification of genetic incompatibility. March 13, 1964.

Cushing, C. E. Plankton and water chemistry. March 18, 1964.

Ragan, H. A. Effect of plutonium on swine skin. March 18, 1964.

Vanderbeek, J. W. New production reactor. March 19, 1964. (100-N Department)

Hungate, F. P. Transmutation effects in biological systems. March 25, 1964.

Price, K.R. Root distribution in *Artemesia* as measured by I^{131} uptake. March 25, 1964.

d. Miscellaneous

McClellan, R. O. Anatomy of the pig. 5th grade classes, Jason Lee School, Richland, Washington.

II. Articles Published

a. HW Documents

None

b. Open Literature

Bair, W. J., B. O. Stuart, J. F. Park, and W. J. Clarke. 1964. Factors affecting retention, translocation and excretion of radioactive particles, p. 253-274. In Radiological Health and Safety in Mining and Milling of Nuclear Materials, Vol. 1. IAEA, Vienna, Austria.

McClellan, R. O., Glenda S. Vogt, R. E. Kane, and F. F. Hahn. 1964. Endotoxin-induced neutrophil response in miniature pigs ingesting strontium-90 daily. Nature 201:721-722.

III. Visits and Visitors

a. Visits to Hanford

Frank Conte, Oregon State University, Corvallis, discussed research with R. E. Nakatani. March 3, 1964.

W. H. Lawrence, Weyerhauser, Centralia, Wash. Discuss research with L. L. Eberhardt. March 3, 1964.

M. M. Sigel, Children's Variety Research Foundation, Coral Gables, Fla. Discuss research with R. E. Nakatani. March 11, 1964.

Dr. R. L. Mathewson, Lerner Marine Lab, Bimini, Bahamas, discuss research with Dr. Nakatani. March 11, 1964.

Adolph Hecht, Ray Bills, Sudhir Kuhmar, Washington State University, Pullman, present seminar and tour facilities. March 13, 1964.

Dr. M. Suzuki, IAEA, Vienna (en route to Japan). Discuss research with W. J. Bair. March 19, 1964.

J. A. DeMoss, University of California, San Diego, California. Discuss research with J. M. Dean and W. H. Matchett. March 18, 1964.

12 prospective AEC employees from various parts of the country toured the facilities with R. F. Palmer. March 27, 1964.

T. Mahoney, Pathologist, Kadlec Hospital, attended seminar and discussed research with M. F. Sullivan. March 25, 1964.

b. Visits Off-Site

2/28-3/7 - W. H. Matchett attended a symposium at Houston, Texas.
3/6 - R. T. O'Brien presented a seminar at Washington State University, Pullman, Wash.
3/7 - L. K. Bustad worked on the ionizing radiation study at the University of Washington, Seattle.
3/6-15 - R. O. McClellan presented a paper at the Society of Toxicology in Williamsburg, Va. and discussed research at the Deaconess Hospital in Boston and the University of Rochester.
3/4-12 - D. D. Mahlum presented a paper at the Society of Toxicology in Williamsburg, Va.
3/6-13 - H. A. Kornberg presented a paper at the Society of Toxicology and discussed the symposium with the AEC, Washington, D.C.
3/9-11 - B. O. Stuart presented a paper at the Society of Toxicology in Williamsburg, Va.
3/8-14 - L. L. Eberhardt and W. C. Hanson presented papers at the North American Wildlife Conference in Las Vegas, Nevada, and visited the Denver Research Center, Denver, Colo.
3/9-10 - C. E. Cushing, W. H. Rickard, and D. G. Watson collected fallout samples at the Blue Mountains near Dayton, Washington.

b. Visits Off-Site (continued)

3/17-18 - D. G. Watson attended the annual review of salmon research at the University of Washington, Seattle.

3/18 - R. O. McClellan presented at seminar at the University of Washington, Seattle.

3/18 & 26 - V. G. Horstman inspected feed at Hermiston, Oregon.

3/19-20 - W. H. Rickard, and L. L. Eberhardt collected fallout samples at Packwood, Washington.

3/24-4/7 - W. C. Hanson, Anaktuvuk Pass, Alaska, for whole-body counting of Eskimos.

3/24-27 - D. G. Watson, J. M. Dean, and R. E. Nakatani attend Pacific Fishery Biologists Meeting at Ocean Shores, Washington. J. M. Dean presented a talk and R. E. Nakatani was chairman of one of the panels.

3/23 - J. M. Dean presented a seminar at Washington State University, Pullman, Washington.

APPLIED MATHEMATICS OPERATION

MONTHLY REPORT - MARCH, 1964

ORGANIZATION AND PERSONNEL

Gauin C. Moore transferred from Reactor and Fuels Laboratory to Operations Research Operation as a Technologist, Operations Research on March 16, 1964.

ACTIVITIES FOR OTHER HAPO COMPONENTS

N-Reactor

The development of an experimental layout and procedure for a proposed in-reactor fuel corrosion test was completed.

A set of data relative to correlations between speed of welding of end caps, depth of weld penetration, amount of weld overhang, and extent of melting of internal cladding was analyzed.

Consultation services were provided on mathematical methods applicable to safety circuit analyses and reliability studies.

An analysis was made of data results from an opinion survey on the subject of safety.

Third and fourth degree polynomials were fitted to data generated from theoretical equations relating the percent Pu-240, percent U-235, change in gms/ton of Pu and change in gms/ton of U-235 individually to the exposure in MWD/T. The polynomials fitted so well that several errors in the data from the theoretical equations were discovered.

Other

A program to assess the significance of orthogonal polynomial effects up to the fourth degree for nonequally spaced independent variables has been completed. It is being adapted to handle other situations too as, for example, in assessing the effect of orthogonal sin θ, sin 2θ, and sin 3θ, effects when the θ are not equally spaced and/or do not cover a full period.

Irradiation Processing Department

The design of a test to assess the effects of carbon, iron and silicon, each at three levels, in the fabrication of uranium on the pre- and post-irradiation characteristics of fuel elements was submitted to interested personnel.

Assistance was given in determining a suitable experimental procedure for determining optimum submerge and preheat times in the manufacture of hot-die-sized fuel elements.

The results of an experiment to characterize fuel element dimensional distortion behavior as a function of temperature and the number of thermal cycles by out-of-reactor tests are being analyzed. The objective is to associate a given set of out-of-reactor conditions with a comparable set of reactor exposure conditions.

The analysis of a test to assess the effects of can and core annuli thicknesses on the quality of the bond cladding was completed and submitted to interested personnel. The report included equations for the response surfaces for different yield variables as functions of the can and core annuli thicknesses as well as graphs depicting the same relationships.

Work is continuing on the analysis of a 5 x 5 factorial design to determine the best lead and preheat times in the manufacture of HDS fuel elements.

The analysis of a tubing test in connection with the manufacture of HDS fuel elements was completed. The report indicates by formulae and graphs how the core and can annuli thicknesses affect the bond integrity measurements.

Studies comparing the Altrex 1097-LF and Oakite NST cleaning and degreasing processes indicated no significant differences between them and the normal F process on the basis of bond integrity measurements.

A report on the effects of various heat treatment variables in the manufacture of uranium on the preirradiation warp of fuel elements was completed and submitted to interested personnel. A considerable number of statistically and practically significant effects were found among the core size, quenching medium, delay time, and quench temperature variables.

The data processing program to go directly from outlet temperature data positioned on the tube circumference to R values has been completed and the program is being used on the large amount of temperature data which has accumulated.

Work is continuing on the analysis of PT-572 to characterize weight loss and dimensional distortion effects according to the support heights of self-supported fuel elements. The relationships of the frequency, area, and severity of hot spots and weight loss are being studied.

A study was completed which compared dimensional characteristics of HDS fuel elements canned by six different variations in the process at five stages of the canning.

Results were submitted to interested personnel on the effects of different welding heads and cleaning processes on the bond strength values of fuel elements.

At the request of interested personnel a table of expected mean squares were calculated to explain why the use of a certain model almost inevitably led to a low F ratio.

Work continued on the 100-H Reactor outage information and control system and the 100-B preventive maintenance information and control system. A report describing the 100-B system is now being prepared. A request has been received to have the preventive maintenance information and control system extended and adapted to the 300 Area fuel processes.

Chemical Processing Department

Appropriate control limits were recommended for weekly analytical measurement error, process variability and process average for minimum Pu content of C-line material to permit compliance with new LRL specification.

For A-line components, a review of data indicated that apparent change in yield strengths was more likely a matter of measurement changes than of metallurgical characteristics. A change in product specifications was recommended.

Business Planning and Transfer Operation

The first phase was completed of a study of retail trade potential in the Tri-City Area. Data estimates were related to 1962 as a base period. The next phase will carry projections through 1967.

Work was begun on Tri-City Area transportation problems. This included a preliminary analysis of local airline service as a connecting link to long-distance service, and local highway needs for future growth and development.

ACTIVITIES WITHIN HANFORD LABORATORIES

2000 Program

C-Column Computer Control

Statistical analyses of gamma absorptiometer data were completed and write-ups are in process in connection with the revision of the paper, "A Data Logging Absorptiometer for Routine Uranium Analysis" for publication in Talanta. Investigation continued on the numerical problems associated with the calculation of the solution of the nonlinear system of differential equations which explain mass transfer in the C-Column. An experiment was designed to calibrate the twin filter photometer for direct quantitative analysis of organic uranium solutions.

A number of 412 computer programs and subprograms must be written for control of the C-Column. Specifications for various of these programs are beginning to take form and the first is over half coded. This program will decode the rotary switches of the manual input handle, carry out operations such as ranging high and low limits, scan points, turning on other functional programs and writing out internal values and newly inserted values on the alarm typewriter.

Ground Water Models

Work continued on the "shoe-box" model for ground water flow. Various cases have been run on EDPM equipment, and the output is being subjected to an error analysis. Additional discussions have been held on the "streak-function" concept.

3000 Program

Numerical Control of δ-ω Lathes

A test tape produced for the experimental δ-ω lathe exhibited unacceptable bit-spacing characteristics that have never been encountered before. Extensive checks of the EDPM program CUPID which produces these tapes indicate the trouble arises from a seldom experienced case of machine round-off error induced by a specific geometry. An appropriate test and double precision subroutine is being written for insertion into CUPID to prevent this situation from reoccurring.

Numerical Control of Rotary Contour Gauges

Work continued on a program to predict and quantify the types of errors which could occur on the rotary contour gauge because of part misalignments.

4000 Program

Radiation Damage to Fissionable Materials

Work was completed on the major sections of the FORTRAN language "Bubbles" program for the quantitative analysis of metallographic data collected on the Ziess particle size analyzer. Debugging of the program continues and sections pertaining to special situations will be added in the near future.

Sonic Transmission in Visco-elastic Media

A series of meetings were held to discuss appropriate experiments -- both theoretical and in the laboratory -- that could be devised to test the applicability of the recently developed "linear attenuation" model for wave transmission.

Particle Packing Studies

Calculations were made to determine optimum screen sizes and proportionate mix factors for the fabrication of an annular-shaped vibratory compacted fuel element.

General

Work continued on assembling all the appropriate chemical reaction relationships which are needed as input to the EDPM program that simulates carbon burnout in a nuclear reactor.

5000 Program

Actinide Element Research

Work continued satisfactorily on crystal indexing routines. The ORTHO program was essentially completed with final details awaiting experimental data.

General

A closed form solution was obtained to a two-dimensional mathematical model for the propagation of a small crack in an elastic body. Further studies were made on methods of applying conformal mapping techniques on a similar problem.

An analysis was made and an EDPM program written to aid in a study to determine the number of changes in energy level which will result when metallic materials are subjected to sudden changes in temperature.

Work is proceeding on the problem of estimating the total amounts of various isotopes in a channel based on a harmonic-type regression model and estimates of the curies per foot at different locations within the tube.

Density distributions of particle sizes fitted to experimental data for 18 different sets of data were completed. In connection with the study, 36 two-way contingency tables are being tested for interaction between the frequency of particle sizes and sampling location.

Derivation was completed of a model of propagated errors associated with percentages of constituent gases as obtained from a mass spectrometer. The 7090 computer is to be used to obtain mappings from this error model for an extensive range of gases.

Estimates were obtained (iteratively) of constants in a previously derived model relating stress in a metal to the elapsed time following discontinuance of a mechanically applied strain.

Computation and Statistical Analysis

Work continued on the calculation of the power function of the Poisson index statistic used to check stability of counting instruments. The FORTRAN language program for performing power calculations 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 was debugged and is currently being used to construct tables of the power function. The current library FORTRAN program for calculating the incomplete gamma function is being revised to extend its calculation range to handle chi-square distributions with degrees of freedom greater than 68.

Radiochemical Analysis

A new main calculation program for the IRA system was put into production service during the month and recalculation of the old RCA data started. A new change and error report has been specified for the Program IRA-325. The present report is excessively long, time consuming to produce, and difficult to read. The new report is an improvement in all these areas.

Two new IRA passes were specified during the month, which deal with the handling of inactive analyses retired from IRA-325 program file. Retirement of data from IRA-325 reduces significantly the expense of operating the IRA system.

<u>6000 Program</u>

The first major assignment was started on the economic-biological consequences of a nuclear attack. This assignment is to develop ideas concerning a basic approach to evaluating economic interactions with the Nation's biological system.

Ecological Models

A series of meetings were held with various plant representatives to discuss the present state of the art on nuclide transport prediction. The posture of HAPO concerning mathematical models used in such prediction was ascertained.

Biology

Statistical analysis of data from a promethium-147 study on beagle dogs was completed. The excretion data of promethium in urine and feces were fitted to power functions of time and the parameters of these functions for inhaled and intravenously administered Pm-147 were compared.

A computer program was written to calculate the amount of DNA in a nucleus. The input data for the program are the diameter of the nucleus and percent light transmittance through the nucleus which are measured with an ultra-microspectrophotometer.

A statistical analysis of crab data designed to investigate the differences in the amount of blood glucose found in the crab at various stages of development and for various species was completed. The stages of development were mature male, immature female, early spring female with no eggs, female with early eggs, and female with old eggs.

The statistical analysis of data from the study to investigate the effect of treatment with DTPA on the clearance of N_p-239 and P_u-239 from various rat tissues was completed.

<u>Other</u>

A FORTRAN language program ELSA (Elementary Spectral Analysis) was written and debugged during the month. This program will analyze a set of time series data by blocking in groups of 2^n. In each block iterative sums and differences of adjacent pairs of observations are taken. Sums and differences tables are constructed and mean sums of squares computed. The program can also be used to analyze replicated 2^n factorial experiments and factorial experiments where all factors but one appear at two levels.

A task force was formed to investigate possible machine configurations which would satisfy the present and future digital computer needs of the Hanford Laboratories. The initial phase of the investigation, to define the functional computing need of the Laboratories, has been completed and a formal report is in progress.

R Y Dean

R. Y. Dean
Acting for C. A. Bennett
Manager, Applied Mathematics

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RYD:dgl

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PROGRAMMING

REACTOR PROGRAM - 04 PROGRAM

PLUTONIUM UTILIZATION PROGRAM

Crossed Progeny Fuel

As reported last month, the use of U^{233} to enrich high fast effect reactors involves irradiation of U^{233} in hard neutron spectrums; thus, the resonance parameters of U^{233} become of key interest. This has prompted an intense review of the literature which, so far, has produced about 60 pertinent references. These data are to be carefully analyzed and evaluated.

Successive Plutonium Recycle

The analysis of Recycle of Plutonium in Thermal Reactors is now being completed and an initial draft is being prepared. In the course of completing this work, successive recycle is being analyzed (1) with assumed limited control in the reactor, (2) with limited natural resources, (3) with termination of uranium cascade availability, (4) with different amounts of plutonium available for recycle, and (5) with several other minor variations. These cases are set up for calculation as computer time permits.

Production of Transuranic and Transuranium Isotopes in Power Reactors

The first rough draft in final form of the study of Curium and Plutonium^{238} Production in Power Reactors is nearing completion. Some special runs will be made to complete the data. Among these data are calculations of the indifference values of Am^{241} which is a precursor to Pu^{238} by way of Cm^{242} decay. Americium^{241} is formed by beta decay of Pu^{241} (13-year half life). Curium^{242} is formed by neutron absorption in Am^{241}. However, the fact that americium^{241} is a decay heat source in its own right also affects the relative values of these isotopes.

Small Production Reactor Simulation

Recent fuel cycle analyses have been made to evaluate alternate by-product power recovery fueling schemes for the Hanford small production reactors. It appears that if an oxide fuel were to be used in the existing reactor lattice to obtain long exposures, a fuel consisting of UO_2-ThO_2 with plutonium as a driver would yield the lowest fuel costs. Fueling systems consisting of uranium metal have lower fuel costs, but the exposure to which this fuel will run may tend to eliminate this advantage.

Tube-In-Tube Fuel Element in Small Production Reactor Simulation

A tube-in-tube fuel element was introduced into a simulation of the Hanford small production reactor lattice to determine the effect of varying the water annulus between the inner and outer fuel regions on the resonance escape probability, p. The JASON-MELEAGER chain was used for this investigation, to examine cases for both UO_2 and uranium metal. As the thickness of the water annulus was decreased, the probability of resonance escape decreased for both fueling systems. At the position of the largest water annulus, p was a maximum for the UO_2 system (see Figure 1). The value of p for any size water annulus is larger for the UO_2 system because there is less nuclear material present in which a resonance absorption event can occur.

At the position of maximum p, the value of k_{∞}* was a minimum. This result was due to the small value obtained for f.** The values of the other terms in the four-factor formula varied slightly as the water annulus was changed but the variation of f was the dominant term in producing changes in k_{∞}. Thus, it appears that there is an optimum size of water annulus for a tube-in-tube fuel element that will produce values of p and f in the small production reactors that will result in fewer resonance absorptions in U^{238}, and thus, will tend to minimize the production of plutonium isotopes. The liberty to select this optimum is sensitive to the criteria of reactivity transients as the coolant density changes. Because p changes far more slowly than f with water annulus thickness, one cannot have as stable an operating point with thick annuli as with thin annuli. This situation may be reversed with plutonium enrichment, and is now being checked.

CODE DEVELOPMENT

Uranium Conservation Studies (YUKON)

The YUKON code was changed by the addition of a special curve fitting subroutine, LIME. This subroutine converts data from the MELEAGER CHAIN computations into fitted curves adapted to the YUKON code. The calculation of economic perspectives was added to the YUKON code as were several improvements to the output.

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* K_{∞} = the neutron multiplication factor for a reactor of infinite size.

** f = thermal utilization, or the ratio of thermal neutrons absorbed in the fuel to the total thermal neutrons absorbed in fuel, moderator, etc.

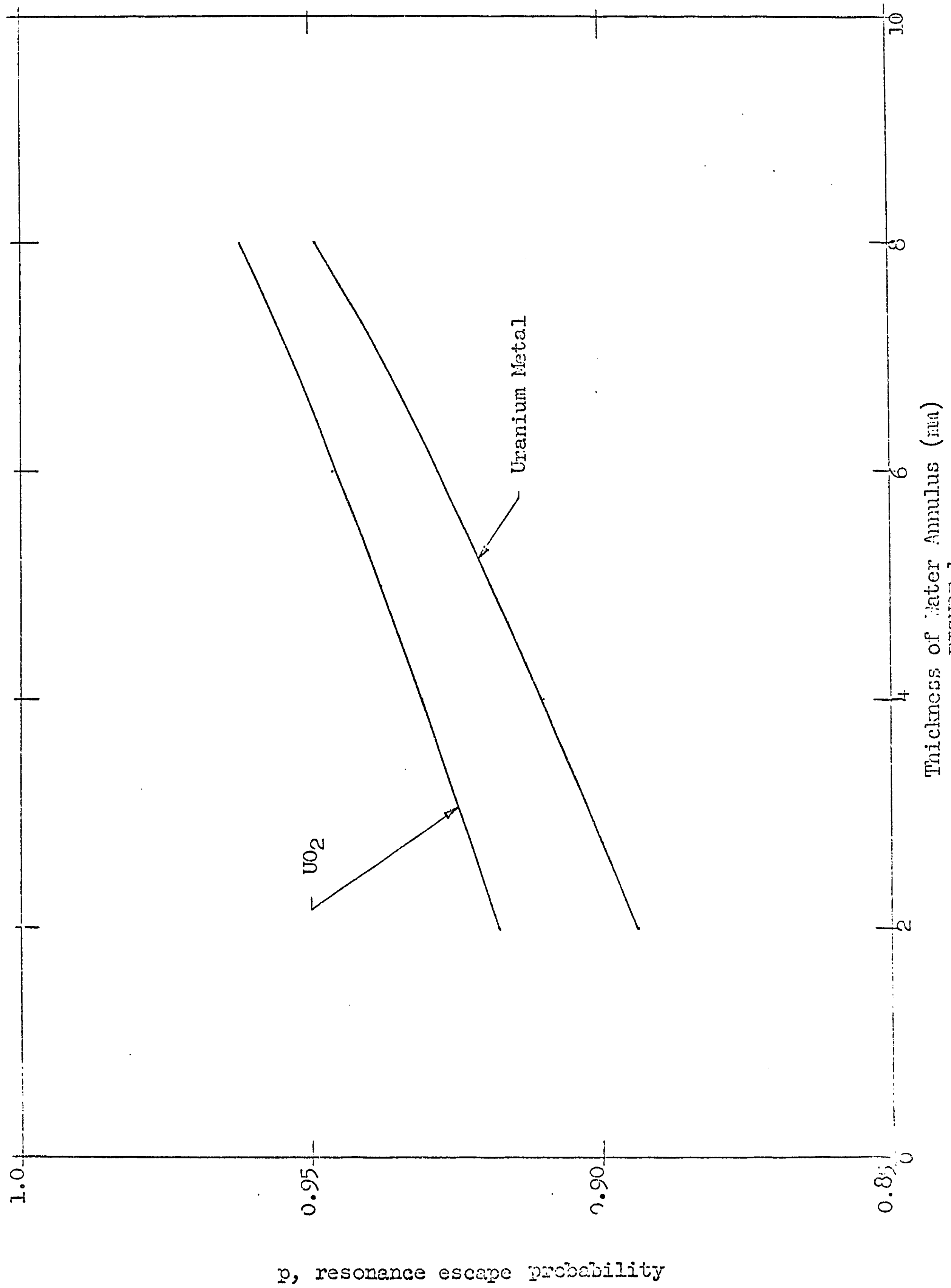

FIGURE 1
RESONANCE ESCAPE PROBABILITY FOR A TUBE-IN-TUBE FUEL ELEMENT
IN SIMULATION OF SMALL PRODUCTION REACTOR LATTICE

Work was begun to calculate additional recycle data for the BWR simulation at various price schedules so that the uranium conservation problem can be studied in detail using this type of reactor. Further work in the conservation study includes (1) the revision of the YUKON code to match reactor operation with recycle of plutonium to reactor operation without recycle of plutonium, (2) the addition of the capability of considering two or more reactors simultaneously within each of several hypothesized economies, and (3) the study of various rates of breeder reactor introduction into the economies hypothesized.

ALTHAEA

ALTHAEA has been revised to provide 100 cells in common for communication with QUICK, OPTIMIZER, and other codes on the ECONOMICS CHAIN System. This was the first simultaneous recompiliation of all subroutines since 1962, so this opportunity was taken to clean up and consolidate all the minor patches and revisions incorporated in the past 18 months. Provision for the influence of fuel temperature on resonance shielding (Doppler broadening) was also incorporated. An effective resonance integral is computed from the dilute resonance integral as follows:

$$RI = \frac{RI_O}{\sqrt{1 + \frac{(Y)\ (Z)}{SCA}}}$$

Where

RI = effective resonance integral
RI_O = dilute resonance integral
Y = concentration
Z = empirical value
SCA = potential scattering of the fuel plus the effective hydraulic radius of the fuel.

The magnitude of Z is usually on the order of the maximum cross section in the first resonance and a value may be selected from BNL-325, if effective integrals as a function of concentration or reaction rate data are not available. At the higher operating temperatures of the fuel, the proper values of Z are less than at room temperature. The decrease will be a function of the temperature and the width of the resonance. It will now be possible to develop and use variable Z's as a function of fuel temperature.

Efforts related to the seed-blanket studies consisted of lengthy hand calculations to obtain assurance that all cost factors are being properly handled in the QUICK code.

SUPERCASE GENERATOR

This program is used to generate cases for the MELEAGER burn up code to analyze, and has been altered so that the changes it reads as input can now be read either from Tape 2, which would include data for MELEAGER submitted as input by the programmer, or from Tape 10, which would contain more accurate MELEAGER data written to it by JASON, a code which has the ability to calculate the MELEAGER parameters from geometry data.

There are some programs in existence at Hanford that cause the IBM 7090 to write information onto tape in 132-column configurations. If it becomes necessary to print such a tape, it can be printed on one of the two IBM 1401 computers at Hanford. The 1401's, however, are scheduled to be removed soon, after which the only available input-output equipment will be the Anelex printer and the G.E. 225 computer, both of which print out a maximum of 120 columns. Accordingly, a program has been written which will read 132-column information from tape and print it out in 120-column arrays. The program is named REDUCE.

The QUICK reactor economics code was originally designed to calculate fuel costs based on use charge and depletion payments every six months as required by the AEC Use Charge Regulation. It was also set up so that the initial burn up period started at the beginning of the first six-month payment period. Most multizoned reactors will require that fueling costs be calculated for cases which cannot logically begin burn up at the beginning of a payment period. For these reasons QUICK was modified such that cost calculations are no longer dependent on the time at which the first burn up period begins. A second modification was made to allow for payment periods other than the six-month period used for AEC Use Charge payments. The latter change makes the code more suited to studies of privately owned reactors, as any payment schedule can be used, as well as a rigid six-month payment period.

Nuclear Safety Activities

The theory, experimental results, local meteorological basis, and application of variability in wind speed and direction to reactor accident analyses were reviewed in detail with AEC-Washington personnel, primarily from the Divisions of Production and Licensing and Regulation. Similar information specific to the N Reactor was briefly summarized in a meeting with a subcommittee of the Advisory Committee on Reactor Safeguards. No basic objections were raised during these discussions to the use of the new evaluation method.

F. W. Woodfield
Manager-Programming

FW Woodfield:ch

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF MARCH 1964

A. ORGANIZATION AND PERSONNEL

A. M. Valentine joined Radiation Monitoring as an Engineer effective March 23. Gordon A. Little formerly of Radiation Monitoring resigned on March 20. Betty W. Compton, Secretary, CDS&R, transferred to Programming effective March 23, and Betty B. Haller filled the vacancy. C. E. Rosenberry, Radiation Monitor Trainee, terminated March 27, and W. L. Coles, Radiation Monitor Journeyman, terminated March 6. R. C. Lawrence, RMO, transferred into Environmental Studies and Evaluation, effective March 2.

B. ACTIVITIES

Occupational Exposure Experience

There were four new plutonium cases confirmed by special bioassay analysis during the month. Three cases were estimated to be less than 10% of the maximum permissible body burden (MPBB, plutonium with bone as reference, is 0.04 μc), and the fourth case was estimated as less than 20% of the MPBB. All four cases were detected by routine bioassay sampling.

The total number of individuals who have received internal plutonium deposition at Hanford is 342. With the termination of three employees having confirmed plutonium deposition, there are currently 246 employed.

In March, there were eleven incidents involving fourteen employees which required special bioassay sampling for plutonium analysis. The following is a brief description of the more significant incidents:

A CPD operator received skin contamination of 10,000 d/m while working in the Weapons Manufacturing Building (234-5). The employee was transferring a carton of radioactive waste when it collapsed. Nasal smears indicated ≈ 500 d/m in nasal passages. Decontamination efforts were successful.

A CPD laboratory employee received a contaminated injury while decontaminating equipment in the Weapons Manufacturing Bldg. (234-5). Dichromate solution splashed on the employee's face causing visible acid burns and facial contamination to 40,00 d/m. The contamination was reduced to approximately 500 d/m.

A CPD employee was exposed to high air-borne concentrations of plutonium while dissolving 200 mg of plutonium metal in a hood in the Weapons Manufacturing Bldg. (234-5). Nasal contamination, indicated by nasal smears of 1050 d/m and 600 d/m, and skin contamination of 2000 d/m were reduced to non-detectable levels.

A CPD employee received a possible contaminated injury while working the Redox Laboratory (222-S). A glass vial, contaminated to 5000 d/m plutonium nitrate, broke, and penetrated the right hand in two places. A tourniquet was applied and the wound flushed. An examination, using the plutonium wound counter, showed less than the detection limit of 1×10^{-4} μc.

A CPD operator received a plutonium oxide contaminated injury on 3-26-64 while working at the Weapons Manufacturing Bldg. (234-5). The employee was moving an empty pallet from a conveyor to a hood when a sharp object penetrated his right index finger. The wound did not bleed; however, a small black object could be seen in the finger. Examination of the wound, using the plutonium wound counter, indicated 1.2×10^{-3} μc plutonium at the wound site. After excision at GE First Aid Station, a recount indicated less than the 1×10^{-4} μc detection limit of the wound counter. The employee is not a previous deposition case.

Two HL employees were involved in an accident involving plutonium metal while working in the Plutonium Metallurgy Facility (231-Z). When a can containing plutonium metal was opened in the storage vault, smoke was detected. The first employee involved replaced the plastic clad sample which he had removed from the can and placed the can on the floor. A second employee placed the small can into a larger one and moved them to a nearby hood. Hand contamination of 10,000 d/m and nasal contamination, indicated by smears of 850 d/m, were quickly removed. A routine air sample obtained in the storage vault indicated an air-borne concentration of 5.3×10^{-9} μc Pu/cc.

In addition to the incidents involving plutonium, there were two incidents that required evaluation for possible intake of fission products. Neither incident resulted in significant deposition of radioactive material in any of the employees involved.

There were two incidents of special interest involving exposure of employees from external sources.

An HL millwright received a localized exposure in excess of the 13-week operational control, to his right thumb while working in the storage basin at the Plutonium Recycle Test Reactor (309 Bldg.) on 3-11-64. The exposure was incurred during the removal of rollers from a fuel inspection tray when a small piece of irradiated zirconium wire penetrated the employee's glove and was in contact with the thumb for an estimated three minutes. The estimated dose to the employee's thumb was 57 rems including 42 roentgens.

An HL employee received a dose to the skin of a major portion of his body estimated at 2.3 rads and a whole body penetrating dose of 0.25 R while working in the Radiometallurgy Facility (327 Bldg.) on March 11. The employee was working on top of B cell in a position where his back was exposed to a dose rate estimated at 25 rads/hour including 5.4 R/hour.

Environmental Experience

Concentrations of fallout materials in the air of the Pacific Northwest averaged 1.0 pc β/m^3 during March, slightly higher than the average value of 0.8 pc β/m^3 noted during February. The weekly samples of particulate concentrations in air in the vicinity of the plant ranged from 0.2 to 2.2 pc β/m^3 in air.

The monthly average emission rate of H^3 from the PRTR stack was estimated at 7 curies per day during March. This is the highest monthly average noted to date. The previous maximum was 5 curies per day during the severe fuel element rupture of August, 1962. The current emission rates are well below any level of concern, but represent an increasing trend due to long-term buildup of H^3 in the heavy water. The primary system has been the principal source of heavy water and tritium leakage to the ventilation exhaust system.

Results of tritium analyses of river water were above detection limits for the first time. The concentrations of 4 pc/ml and 3 pc/ml measured at the 300 Area in January and February samples, respectively, indicate a rate of transport of about 600 curies/day. Analysis of samples taken upriver from Hanford is being initiated to confirm that the H^3 is from fallout. Information from others indicates that tritium in rainfall from weapons tests is at an all time high and still climbing. No significant exposure is implied from these concentrations.

Analysis of routine sagebrush samples collected from the 200 Areas for Mn^{54} was instituted recently and values of around 3 to 4 pc/g (wet) have been found. Similar concentrations were found on sagebrush sampled from Irrigon, Oregon, confirming that the source of this nuclide was fallout rather than the Hanford operations.

Studies and Improvements

The neutron spectrometer system was calibrated on the energy scale used for the measurements at the C-2 test hole, K-West reactor. This particular energy scale was not used before, but was convenient since this spectrum was somewhat degraded. After incorporating the detector material cross-section corrections, the incident neutron spectrum was determined. The neutron spectrum has a well-defined peak between 0.5 and 0.6 Mev and tails off gradually to about 3.0 Mev. An unshielded fission spectrum has a peak at about 0.8 Mev with about half of the total number of neutrons in the measured spectrum below 0.8 Mev.

Spectra were accumulated from a PuBe source with and without He^3 in the neutron detector. Since the average neutron energy from PuBe sources is about 4.5 Mev, a significant amount of silicon detector activation events are present in both the neutron and background accumulations. The data were taken through 2 inches of lead to help reduce the gamma radiation interference. To complete the data analysis, the He^3 (n,p)T cross section from 4.2 to 11.0 Mev was assumed to decrease at the same rate as from 3.0 to 4.2 Mev. The final results agree quite well with the PuBe spectrum measured by Tochilin at the U. S. Naval Radiological Defense Laboratory using nuclear track plates.

A pair of 4.5 square centimeter area surface barrier diode detectors were received for use in alpha energy analysis of aerosol filters. The larger surface area caused an increase of about 3.4 in counting rates with satisfactory resolution of the alpha energies in a thorium oxide aerosol filter when compared to the one square centimeter detector previously used. The alpha particle energies from a single source containing plutonium, neptunium, and americium were also detected with satisfactory resolution. This study is providing a method to evaluate routine air samples when mixtures of alpha emitting radionuclides are present.

The exchange period for film badge dosimeters was changed from once each four weeks to once a month resulting in twelve exchange periods per year instead of thirteen. Each calendar quarter was divided into two four-week exchange periods, and one five-week period. The dosimeter exchange dates are now regularly scheduled for the last Friday of each month.

Evaluation of the characteristics of samples of Columbia River water taken at various points on a cross section above the 300 Area showed that one particular sampling point gave results which were representative of the entire cross section for temperature, Cr^{+6}, and the longer-lived nuclides Zn^{65} and Cr^{51}. Gross activity measurements and the shorter-lived nuclides, Na^{24} and Cu^{64}, did not conform to this pattern. The conclusion is tentatively drawn that the highest equilibrium effluent concentrations occur at one point, but the effluent transport rate is greatest elsewhere in the section. Obtaining a representative grab sample in the event of an unusually high activity release of short duration is, therefore, very difficult, but possible with adequate information on the characteristic flow pattern at a specified location.

Individual radiation exposure summaries, with accompanying pamphlets, were distributed to more than 8,500 HAPO employees. This annual summary listed for each individual both the whole body dose received during 1963 from penetrating radiation and the total whole body penetrating dose received as a result of his employment at Hanford. The accompanying pamphlet was provided to assist the employee in understanding the dose units used and in making a comparison of his dose with recommended permissible exposure limits.

Research Studies

Effect of Reactor Effluent on the Quality of Columbia River Water (02)

Work progressed on defining the concentration of reactor effluent in cross sections of the river and the variations in water temperatures in the cross sections. The work was inhibited by lack of an adequate boat for carrying out dye studies (a replacement for the large boat is on order), and a maintenance outage of the rapid response thermometer.

Outlet temperature data was obtained from a 107-K basin during the month, the last of the areas to be monitored. Results to date have shown the same significant loss of heat as from the basins observed in other areas.

Mechanisms of Environmental Exposure

The Zn^{65} in reactor effluent water in combination with whole body counting appears to provide an excellent indication of the extent of use of Columbia River water by people in the environs. Body burdens of Zn^{65} in most residents of Pasco and Richland can be used to estimate the quantity of water consumed, the burdens in local fisherman to estimate fish consumption, and the burdens in rural residents to estimate the intake of milk and beef from irrigated pastures. One of the highest body burdens of Zn^{65} yet measured (100 nc) was found in a 300 Area employee that has been consistently eating local beef raised on pasture irrigated with Columbia River water. The local game protector that has assisted in the Creel survey is optimistic that he can interest several retired men who spend a comparatively large amount of time fishing in visiting the whole body counter.

Nuclear Facilities Monitoring Guide

A visit by Harold Bernard, AEC-DRD, provided an opportunity to discuss the intent of the Guide with a representative of the sponsoring AEC division. It was learned from the discussion that the sponsors hope that the Guide will not only describe monitoring methods which enable nuclear facility operators to comply with current regulations, but also will present the principles of monitoring effluent discharges, and thus provide a base from which current regulations can be improved.

C. RELATIONS

Safety meetings were held throughout the Section during the month. A safety questionnaire entitled, "Poison Proof Your Home" was prepared by J. W. Spoonemore and distributed to RMO personnel.

Eight new suggestions were received during the month. Six suggestions were evaluated; five suggestions rejected; twelve suggestions are outstanding; and one suggestion was adopted for an award of $145 to H. N. Larson and J. R. Berry for suggesting the use of an improved form for routine surveys.

One security violation was incurred.

Preliminary training of 321 Building personnel to improve radiation protection control practices was started. This training was initiated because of thorium use in the facility and to acquaint personnel with the more rigid practices which will have to be observed when they transfer into the 324 Building.

All four sessions of the seventh round, and session one of the eighth round in the Radiation Monitoring Refresher Course were held. Total attendance was 53, of whom three were CPD employees. The sessions in the seventh round dealt with monitoring techniques and practices.

Sessions five through nine of the Radiation Protection Training Course for Exempt Personnel were held. Average attendance has been about 25. The comments of the participants continue to be very favorable.

A one-hour training session covering the biological hazards of thorium, uranium, plutonium, and beryllium was presented for employees of the Technical Shops. Personnel were very interested, and presented several job-related situations for discussion. Total attendance was about 25.

D. SIGNIFICANT REPORTS

HW-80892-2 - "Radiological Status of the Hanford Environs for February, 1964" by R. F. Foster.

HW-81295-1 - "Radioactive Liquid Waste Disposal for January, 1964" by R. H. Wilson.

	March	1964
External Exposure Above Permissible Limits		
Whole Body Penetrating	0	0
Whole Body Skin	1	1
Extremity	1	1
Hanford Pocket Dosimeters		
Dosimeters Processed	5278	11,054
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	10,151	29,672
Results - 100-300 mrads	195	470
Results - 300-500 mrads	27	57
Results - Over 500 mrads	7	13
Lost Results	0	37
Average Dose per Film Packet - mrad (ow)	6.7	6.8
- mr (s)	37.9	36.05
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	229	3110
Results - 50-100 mrem	0	24
Results - 100-300 mrem	14	29
Results - Over 300 mrem	1	5
Lost Results	0	5
Fast Neutron		
Film Processed	845	1541
Results - 50-100 mrem	5	32
Results - 100-300 mrem	11	111
Results - Over 300 mrem	0	2
Lost Results	0	3
Hand Checks		
Checks Taken - Alpha	38,761	108,607
Beta-Gamma	52,946	170,639
Skin Contamination		
Plutonium	32	53
Fission Products	90	174
Uranium	4	5
Tritium	0	0
Thorium	0	0

Whole Body Counter

Subject	747-A WBC	1964	Mobile WBC	1964
	Number of Examinations			
GE Employees				
Regular	34	153	104	429
Incident Cases	7	32	2	3
Terminations	9	17	0	0
New Hires	32	76	0	0
Special Studies	3	15	0	10
Non-Employees	1	2	0	0
Total	86	295	106	442

Bioassay

Analysis	Current Reporting Limit	Results Above Reporting Limit Mar.	Results Above Reporting Limit 1964	Samples Assayed Mar.	Samples Assayed 1964
Plutonium	2.2×10^{-8} μc/sample	150	270	834	1627
Fission Prod.	3.1×10^{-5} μc/sample	0	0	0	0
Strontium	3.1×10^{-5} μc/sample	11	11	57	57
Tritium	5.0 c/l	135	272	205	530
Uranium	0.14 ugm/l	0	0	184	530
Special Studies		0	0	30	90

Calibrations

	Number of Units Calibrated March	1964
Portable Instruments		
CP Meter	1093	3175
Juno	259	769
GM	531	1589
Other	221	686
Audits	102	303
Total	2206	6522
Personnel Meters		
Beta-Gamma Film	648	1778
Rings	75	270
Other Film	244	702
Total	967	2750
Miscellaneous Special Services	524	715
Total Number of Calibrations	3697	9987

Environmental Monitoring

Samples	March	1964
Air		
Filters	304	1039
Scrubbers	213	684
Water		
Raw	73	147
Sanitary	77	228
Process	28	73
Vegetation	80	240
Test Well	231	806
Fish	73	353
Food Products	40	135
Beef Thyroids	40	103
Waterfowl	0	77

Measurements

Control Plots	53	123
Aerial Monitoring	2	4
Ionization Chambers	203	637

for C.E. Newton Jr

Manager
RADIATION PROTECTION

AR Keene:ald

FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

The operating cost control budget was adjusted in March to reflect the changes in funding contained in Financial Plan No. 6 issued by RLOO-AEC. In summary, the authorized funds for the nonproduction programs assigned to Hanford Laboratories were adjusted as follows:

	Increase
04 Program	$ 31 000
05 Program	35 000
08 Program	50 000
Total	$116 000

Additional equipment and miscellaneous capital work order funds were allocated to Hanford Laboratories during the month. The $288,000 added equipment funds raised the FY 1964 allocation to $2,430,000. The miscellaneous capital work order authorization was increased $151,000 for a total FY 1964 authorization of $347,000.

An increase of $35,000 in the Project Whitney authorization for FY 1964 was received from RLOO-AEC. The annual authorization now totals $185,000 for this work sponsored by Livermore Radiation Laboratory.

Work is continuing on the Budget for FY 1966 and Revision of Budget for FY 1965. The Research and Development proposals for all programs have been prepared and are currently being reviewed by management. The 04 Program proposals were submitted to Contract Accounting. The irradiation unit requirements for the two years were prepared and submitted to Contract Accounting.

Special accounting codes were established during the month for the activities described below:

- .3Z - Fuel Development Studies for NASA. A $900,000 authorization has been received for this work extending into FY 1965.

- .5A - Nondestructive Test Development for High Temperature Nuclear Components for NASA. $75,000 has been authorized for this work extending into FY 1965.

.3N - Basic Cermet Studies for NASA. An additional $110,000 has been received in support of this work, bringing the total authorization to $210,000.

.2B - Oxide compaction for Appliance Park, Louisville, Kentucky. $2,500 has been authorized to perform some vibratory compaction work on oxide for calrod heating elements.

.1A - Plutonium Inhalation Studies for the U. S. Air Force. $57,500 has been authorized for the continuation of this program to November 1, 1964.

Organization code 7459 was established for the Testing Methods Engineering function, which was transferred from N Reactor Department, effective March 1, 1964.

General Accounting

The following new or revised OPGs were issued during the month of March:

OPG No.	Title
1.13	Participation in Hazardous Business
22.1.6	Applied Mathematics Operation
22.1.7	Finance and Administration Operation
22.1.11	Programming Operation
3.1.2	Holidays
3.4.4	Inquiries Relative to Employees
3.4.18	Hanford Nuclear Program
3.6.1	Communication Regarding Radiation Incidents, Accidents or Emergencies at HAPO
3.6.2	Communication Regarding Radiation Incidents at HAPO (canceled)
3.7.4	Announcements of Organization Changes
4.3	Use of Federal Standard #186
5.2 Supp.	Authorizations for Construction or Acquisition of Plant and Equipment (Supplement covering Automatic Data Processing (ADP) Equipment)
*10.1	Commitments to Customers and New Contractors

*New OPG

Hanford Laboratories' net material investment at March 1, 1964 totaled $26.0 million as detailed:

			(In thousands)
SS Material			$ 24 358
Reactor and Other Special Materials			1 350
Spare Parts			364
Yttrium			26
Subtotal			26 098
Reserve:	Spare Parts	$77	
	Yttrium	26	(103)
Net Inventory Investment			$ 25 995

The cumulative value of nuclear material consumed in research by Hanford Laboratories during FY 1964 (at March 1, 1964) is as follows:

02 Program	$ 18 813
03 Program	279 375
04 Program	445 782
Total	$743 970

Responsibility for accounting for Reactor and Other Special Materials (General Ledger Account 0577) will be transferred to Hanford Laboratories from Contract and Accounting Operation effective April 1, 1964. HL Property Accounting will provide C&AO information required for preparing the consolidated HAPO report. Hanford Laboratories will continue to maintain a separate General Ledger Account 0571 - Zirconium and Heavy Water until the end of the fiscal year, at which time the two accounts will be consolidated into account 0577.

Certification inventory reports for the quarter ending March 31, 1964 were prepared and forwarded to Hanford Laboratories' holders of Other Special Materials for completion and reconciliation with Hanford Laboratories' Property Accounting records. A report of results will be issued in April 1964.

The heavy water inventory at the end of March 1964 showed a loss of 1,792 pounds valued at $24,801 for the PRTR. Heavy water scrap generated during the month amounted to 3,063 pounds, resulting in a $4,319 charge to operating cost. Heavy water accumulated at March 31, for return to SROO amounted to 17,359 pounds valued at $215,810.

Laboratory Storage Pool activity is summarized as follows:

	Current Month		FY to Date	
	Quantity	Value	Quantity	Value
Beginning Balance	1 903	$1 422 828	1 480	$ 811 520
Items Received	290	152 439	1 649	1 173 756
Items Reclaimed by Custodians	(19)	(6 657)	(179)	(88 220)
Equipment Transfers	(26)	(17 425)	(296)	(126 612)
Items Disposed by PDR			(146)	(11 389)
Items Disposed by Excessing	(177)	(4 208)	(537)	(212 078)
Ending Balance	1 971	$1 546 977	1 971	$1 546 977-1)

(1- Includes 164 items valued at $115,187 on loan at March 31, 1964.

During the month, 163 items valued at $75,407 were loaned and/or transferred in lieu of purchases. A total of 983 items valued at $396,938 has been redirected to useful purposes this fiscal year. Operating cost for FY 1964 (at March 1, 1964) was $12,052.

Total value of equipment and material in custody of the Laboratory Storage Pool at March 31, 1964 was $2.5 million, including Reactor and Other Special Materials, $310,999; SS Material, $154,800; and Other Materials valued at $474,152.

The following contracts were processed during March:

CA-431	J. L. Powell
CA-433	P. L. Walker
SA-332	Union Carbide
SA-333	Great Lakes Carbon
SA-341	Dr. R. S. Layton
MRO- 73	Beckman Instruments

Physical completion notices have been processed on Project CAH-922. Total General Electric costs were $115,365. Total project costs including accruals are estimated at $273,865.

An additional interim authorization of $10,000 was received on Project CAH-100, bringing the General Electric authorization to $20,000.

Personnel Accounting

C. P. Hawthorne retired April 1, 1964.

Personnel statistics follow:

Employee Changes	Total	Exempt	Nonexempt
Employees at beginning of month	1 806	800	1 006
Additions and transfers in	33	19	14
Removals and transfers out	37	16	21
Employees on payroll at end of month	1 802	803	999

Overtime Payments During Month	March	February
Exempt	$ 4 204	$ 5 539
Nonexempt	13 342	10 406
Total	$ 17 546	$ 15 945

Gross Payroll Paid During Month		
Exempt	$ 804 463	$ 780 596
Nonexempt	697 885	554 205
Total	$1 502 348	$1 334 801

Participation in Employee Benefit Plans at Month End	March		February	
	Number	Percent	Number	Percent
Pension	1 614	99.5	1 602	99.5
Insurance Plan - Personal	402		408	
- Dependent	1 396	99.9	1 394	99.9
U. S. Savings Bonds				
Stock Bonus Plan	150	36.9	154	38.2
Savings Plan	67	3.7	64	3.5
Savings and Security Plan	1 249	89.5	1 248	89.0
Good Neighbor Fund	1 294	71.8	1 298	71.9

Insurance Claims	Number	Amount	Number	Amount
Employee Benefits				
Life Insurance	1	$10 139	0	$ -0-
Weekly Sickness & Accident	17	1 733	14	1 186
Comprehensive Medical	80	5 032	89	5 202
Dependent Benefits				
Comprehensive Medical	170	14 132	158	12 974
Total	268	$31 036	261	$19 362

TECHNICAL ADMINISTRATION

Requisitions for 11 nonexempt employees were filled during March; four requisitions remain to be filled at month end.

Suggestion Plan activity:

Suggestions received	40
Suggestions adopted	25
Suggestions rejected	22
Suggestions in process	124

Visitors Center activity:

March attendance	1 312	
Average attendance per day open	50	
Cumulative attendance since 6-13-62	63 969	
Conducted groups	15	(totaling 270 people)

Plant Tour activity:

	Number	Total
General Public Relations Tours	6	259
Special Tours	3	14

Overall recruiting results for March are as follows:

Offers extended	17
Offers accepted	6
Offers rejected	15
Added to roll	7

Advanced Degree - Twelve Ph.D. applicants visited HAPO for employment interviews. Six offers were extended; two acceptances and one rejection were received. Five offers are currently open.

BS/MS (Direct Placement) - Three offers were extended. No acceptances and no rejections were received. There are three offers currently open.

BS/MS (Program) - Eight offers were extended. Four offers were accepted and 14 rejected. Forty-seven offers are currently open.

Technical Graduate Program - Six Technical Graduates were placed on permanent assignment. Three new members were added to the roll and one terminated. Current Program members total 58.

FACILITIES ENGINEERING

At month end, Facilities Engineering Operation was responsible for eight active projects having total authorized funds in the amount of $7,543,500. The total estimated cost of these projects is $10,550,000. Expenditures on them through February 29, 1964 were $2,605,000.

The following summarizes project activity in March:

Authorized projects at month end -------------------------- 8

New projects authorized ----------------------------------- 0

Projects completed -- 0

New projects submitted to the AEC ------------------------- 1
CAH-123, Laboratory Fire Protection System

New projects awaiting AEC approval ------------------------ 5
CAH-114, Critical Mass Laboratory Addition
CAH-123, Laboratory Fire Protection System
CAH-126, Waste Transport System
CAH-128, Heat Transfer Apparatus for Model Studies
CAH-137, Temporary Physical Sciences Center (formerly submitted as CAH-131, Temporary Physics & Mathematics Center)

Project proposals being prepared -------------------------- 5
CAH-136, 327 Building Services Addition
Atmospheric Physics Building
Geological and Hydrological Wells - FY 1964
308 Building Addition
329 Building Addition

The status of active projects follows:

CAH-916 - Fuels Recycle Pilot Plant - Construction is 45 percent complete compared to a scheduled 31 percent. A new design change was instituted changing 56 carbon steel sleeves in the Low Bay Cell to stainless steel. Roof panels, structural steel and concrete block walls are being installed. Potential obstacles to continued progress are delivery of viewing window frames and pipe sleeves to be buried in the cell walls.

CAH-922 - Burst Test Facility for Irradiated Zirconium Tubes - The Physical Completion Notice was issued by the Company. Control instrumentation for the heating and ventilating equipment cannot be calibrated. Correction by the vendor is being sought. The prototype vessel has been partially decontaminated. It must still be modified and installed in the test basin.

CAH-962 - Low Level Radiochemistry Building - Construction bids were opened on March 4, 1964. The successful low bidder was Emmett Nelson of Spokane, Washington, with a low bid of $1,059,540. The Government Fair Cost Estimate was $941,000. The Richland Operations Office is seeking additional funds from Washington before a contract can be awarded.

CAH-977 - Facilities for Radioactive Inhalation Studies - Design is complete. No action has been taken by the Commission during the past two months.

CAH-982 - Addition to Radionuclide Facilities - 141-C Building - Upon completion of design the Architect-Engineer increased his estimate of the direct construction cost from $96,000 to $131,000. The Commission has not issued a project proposal requesting construction funds.

CGH-999 - Plutonium Recycle Critical Facility Conversion to Light Water - The Commission has deferred action on the project proposal requesting an extension of time and authorization to change the scope to permit handling of irradiated fuel. Advice is being sought from Washington.

CAH-100 - High Temperature Lattice Test Reactor - Detailed design is 35 percent complete. The Commission has not issued a design schedule. Vitro Engineering Company is supposed to submit a schedule for Commission approval this month. The Commission failed to approve two previous submissions.

It now appears that cooling water requirements for HTLTR will exceed the capacity of the process sewer system. This will probably necessitate provision of a special process drain system in the project.

At a meeting on March 13, 1964, Vitro presented a list of dates when design information being developed by the Company will be required. These dates are being reviewed to determine the feasibility of meeting them with current research and development funds. The Company's request for authorization of an additional $50,000 for prototype development has not been approved. A Work Authority modification authorized the Company an increase from $10,000 to $20,000 for project activity.

CAH-114 - Critical Mass Laboratory Addition - The project proposal requesting authorization of design funds is still awaiting authorization by Washington-AEC.

CAH-116 - PRTR Decontamination and D_2O Cleanup - Title I design was completed by Vitro Engineering on March 5, 1964. The Company's comments were submitted to the Commission on March 8, 1964. Detailed design was scheduled to start on March 15, 1964.

CAH-119 - PRTR Storage Basin and Experimental Facilities Modification - Detailed design by Vitro Engineering is 20 percent complete compared to a scheduled 44 percent.

CAH-123 - Laboratory Fire Protection System - Action on the project proposal has been deferred while the Commission determines the availability of funds to authorize the project.

CAH-126 - Waste Transport System - The Commission approved the design criteria but has deferred action on the project proposal. The project proposal requests authorization of $315,000 from General Plant Projects funds for performance of the construction portion of the work and $235,000 from budget item 0290 for purchase of the rolling stock. This split was established at the Commission's request.

CAH-128 - Heat Transfer Apparatus for Model Studies - 185-D Building - Commission action on the project proposal was deferred. The Division of Production felt that it could not justify the expenditure. However, an inquiry is being made to determine if the Division of Reactor Development is interested in the project and has funds available to finance it.

CAH-136 - Services Addition - 327 Building - The project proposal is routing through Contract and Accounting. An informal meeting was held with the Commission to discuss the proposal and design criteria. The Commissions' reaction was favorable.

CAH-137 - Temporary Physical Sciences Center - (Formerly CAH-131, Temporary Physics and Mathematics Center) - The Commission deferred action on this project pending a clarification and resolution of security problems.

A new proposal was submitted to resolve problems. It is currently planned to relocate Programming into the 5201 Building instead of Applied Mathematics. The title and minor parts of the text were altered to reflect the change. The scope and cost of work to be performed are not changed from those described in Project Proposal CAH-131.

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 test with irradiated fuels; (4) review of A-E preliminary design on CAH-100, HTLTR, and consulting engineering on heater element design; and (5) preparation of scope and criteria for 308 Building Addition.

Budget studies were completed 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.

Vendor information drawings were completed for the major vessels for the Containment Testing Facilities and were released to J. A. Jones for purchase when AEC comments are received. Design criteria are being prepared for design of the facilities and equipment by Vitro Engineering.

Engineering and consulting work was provided to research and development personnel as requested. Major work items included: (1) engineering assistance on experimental neutron spectrometer 105-KE building; (2) analysis of 325 building and layout sketch for a chemical fume canopy; (3) engineering assistance on fast reactor concept study; (4) engineering assistance on critical mass experiments; (5) engineering assistance to CPD on Project CAC-880, CAC-121, instrument design, and dry-air system, 234-5; (6) development of engineering information for purchase of computer trailer; (7) engineering for installation of waste flow meters; (8) engineering on liquid sodium loop; (9) consulting instrument engineering assistance to IPD on retention basin sampling equipment; (10) preparation of layout of service piping for room 11, 324 building; (11) engineering for extension of hydrogen gas system, 306; (12) engineering analysis of Biology Laboratory breathing air system; (13) engineering of high-temperature gas loop, 314 building; (14) preparation of standard specification for high efficiency filters; and (15) analysis and recommendation for correction of power disturbances on 325 building instrumentation.

Pressure Systems

Engineering of the PRTR corrosion and film studies loop was reviewed and recommendations made to the user. The loop will be fabricated of ASTM grade A-312 pipe and will require a deviation request.

Welding problems on the high-temperature helium loop were analyzed and corrective measures indicated.

Engineering assistance was provided on the geometry of the sodium loop to obtain acceptable thermal stresses.

Engineering is being provided for installation of a high pressure (2500 psi) hydrogen gas purifier.

A code variance form was prepared for approval of the repairs to the PRTR flash tank.

Facilities Operation

The following table summarizes Waste Disposal Operation:

Item	January	February
Concrete waste barrels disposed to 300-Wye burial ground	12	4
Concrete waste barrels disposed to 200-W plutonium burial ground	4	5
Loadluggers of dry waste disposed to 300-Wye burial ground from 300 Area sites other than the 325 building	24	19
Loadluggers of dry waste disposed to 300-Wye burial from the 325 building	4	2
Loadluggers of dry waste disposed to 200-W plutonium burial ground from 300 Area sites	10	7
Crib Waste Volume, gal.	395,000	310,000
Total Beta Conc., µc/gal.	402.0	131.0
Total Beta, Curies	158.8	40.6
Total Alpha Conc., µc/gal.	487×10^{-3}	86.3×10^{-3}
Total Alpha, mc	192.4	26.8
Pu Concentration, µg/gal.	2.38	0.215
Amt. Pu. g.	0.94	0.067
U Concentration, µg/gal.	4470.0	1510.0
Amt. U. g.	1777.5	468.1

A contamination spread occurred on March 6, 1964 at the 300-Wye burial ground when waste from 325-A was dumped into a vertical burial pipe previously used to dispose 327 building waste. The 325-A waste was in a container ~12" in diameter by 36" long. The exact cause of the spread is not certain. The ground area immediately surrounding the pipe was contaminated to 40,000 c/m. Other ground area was contaminated to a lesser amount. There were several skin and clothing contamination cases. Personnel were decontaminated with no problems. Clothing has either been reimbursed for or is being decontaminated. Vehicles which were contaminated have been decontaminated and are back in service. Ground contamination has been covered.

Several control measures have been tried. All have been apparently successful. Water spray in the pipe while dumping and backfilling is being used routinely. Foam was tried but the equipment is quite elaborate. Fabrication on plant of a foam generator is being studied. Plastic bags have been used to catch the cans. They prevent ground contamination but have caused contamination to blow up into the cask. Control measures will be tried until a successful and economical system is worked out.

On March 3 and 4, there were four retention basins that had activity levels ranging from 1.0×10^{-5} to 1.2×10^{-5} μc-β/ml (Cr^{51} & Np^{239}). On March 9, basin #2 had an activity level of 1.1×10^{-5} μc-β/ml. On March 20, basin #3 had an activity level of 1.41×10^{-5} μc-β/ml. No basin exceeded 1.0×10^{-7} uc-α/ml during the month.

Tanker #5472 was flushed by CPD personnel. Maximum tanker reading is now 900 mr/hr @ 2".

About 50 casks were decontaminated this month in the wash tank.

Drafting

The equivalent of 181 drawings were produced during the month for an average of 24 man-hours per drawing.

Major jobs in progress are: (1) plutonium powder processing line; (2) fast super pressure power reactor concept; (3) capacitor discharge test apparatus; (4) 309 building piping service drawings; (5) PRTR corrosion and film loop; (6) floor plans, HAPO buildings; (7) waste disposal cask; (8) thoria fuel production equipment; (9) automatic densitometer; and (10) safety and control rods HTLTR.

Construction

Major nonproject jobs in progress during the month were: (1) construct swine farrowing bar and pens, 141-F; (2) modify heating and ventilating system, 144-F; (3) install hoods and exhaust duct work, 1705-F; (4) construct new dog runs and modify existing dog runs, 144-F; (5) install intercom system, 144-F, (6) revise electrical system, 146-FR, (7) install fire detection, 100-F area, (8) construct pasture and sprinkler system at meteorological tower; (9) relocate small dynapak and install new dynapak, renovate room 125, install platform stairway and monorail, 308; (10) construct office addition and building services, and construct HTLTR mock-up, 314; (11) construct thoria laboratory, room 417, install emergency generator, modify room 520, install floor tile in ceramics laboratory, install filter boxes and enlarge men's restroom, 325; (12) install lighting, fire detectors and construct roof, 3718-A/B; (13) soundproof rupture loop annex, construct manhole and retention waste by-pass line, install snubbers in high pressure lines and install jib crane and hoist; and (14) 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 has been removed by plant forces, construction of the barrier wall is under way, gallery modifications are started, and the access opening to the canyon is 50 percent complete.

Fabrication work continues on the waste solidification program equipment for 324 building.

Contractor bids to J. A. Jones for repair and modification of ten wells as a trial to determine construction methods substantially exceeded estimates. New bids are being obtained on a revised specification.

DS Parsley

Acting Manager
Finance and Administration

DS Parsley:KKL:whm

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output for March was 1302 MWD, for an experimental time efficiency of 72% and a plant efficiency of 60%. There were nine operating periods during the month, seven of which were terminated manually and two were terminated by scrams. The longest sustained critical period in PRTR history was realized this month. A summary of the fuel irradiation program as of March 31, 1964, follows:

	Al-Pu		UO_2		PuO_2-UO_2		Other		Program Totals	
	No.	MWD	No.	MWD	No.	MWD	No.	MWD	No.	MWD
In-Core	0		7		77	10783.0			84	12149.8
Maximum				277.6		259.8				
Average				195.3		140.0				
In-Basin	7	569.9	25	2686.1	31	2096.5			63	5352.5
Buried							1	7.3	1	7.3
Chem. Process.	68	5465.8	35	1965.8	—	—	—	—	103	7431.6
Program Totals	75	6035.7	67	6018.7	108	12879.5	1	7.3	251	24941.2

Note: (MWD/Element) X 20 $\sim$ MWD/TU for UO_2 and PuO_2-UO_2.

Heavy water and indicated helium losses for March were 1791 pounds and 118,521 scf., respectively.

Equipment Experience

A total of 76 reactor outage hours were charged to repair work. Main items were:

Primary ion exchanger replacement	19 hours
Shim Rods	14 hours
Leak repairs (heavy water and helium)	15 hours
Weld repairs (primary pump seal coolant line)	8 hours

Preventive maint nance for the month required 206 hours or about 5.5% of the total mainter ce effort.

Two repaired and fully operable shim rods and one partially operable shim rod were installed in the reactor.

Installation of a jib crane in the M&M basement access pit, which would be similar to that now installed in the 309 service building basement access pit, was started during March. It is scheduled for completion in May.

Improvement Work Status (significant items)

Work Completed

HX-5 Bypass
Power Calculator Modification
Installation of Trouble Monitor on the Shutdown Valve
Primary pH Cell Modification
Peroxide Pump
Tritium Detector
Core Blanket Moat Rupture Discs with Vacuum Backup Plates
Holdup Tank - High Level Alarm

Work Partially Completed

Storage Basin Work Platforms
Jib Crane at M&M Basement Access Well
In-line Gas Sampling
Process Tubes Level Indicator
Inlet Gas Seal Replacement
Shim Rod Shroud to Top Cap Modification
Improved RTD Connector Sealant and Bracing
Instrument Power Transfer System
Installation of New Alarm Annunciator
Thermistor Probe Installation in FEEF
Flow Monitor Tubing Snubber Installation
Indication of DC Solenoid Failure
DT-1 Storage Tank Sight Gauge
Battery Power for Galvanometer Light
Vibration Snubbers for Earthquake Protection
Process Sewer Bypass Manhole #2A
Process Tube Examination Equipment Installation

Design Work Completed

Core Level Check Valve Modification
Warehouse 3718-C Addition
Mark III Shim Rods
Gas Moisture Detection System Blower Exchange
Modifications to Level Measurement on HX-1
Shim Rod Electric Brake
Containment Valve Additions

Design Work Completed (Continued)

Removal of S-64 Valve
Improved Reliability - Compressed Air, Breathing and Instrument Air
Valve P-11 Removal
Primary Water Sample Station
Inspection Manholes for Containment Penetrations

Design Work Partially Completed

Additional Fuel Storage and Examination Facility
Decontamination Building and D_2O Cleanup Facility
PRTR Steam Utilization
Flux Wire Scanning System
Supplemental Emergency Water Addition
Permanent Installation of Closed Circuit TV
Rupture Monitoring System Modifications

Process Engineering and Reactor Physics

PRTR Test No. 77, Supplement 1, was completed. In this test, it was established that the automatic controller will hold the reactor power level constant during the most severe shim moves possible.

PRTR Test No. 87 was completed. As a result of the data obtained, vacuum backup plates were authorized for each of the moat rupture discs. The backup plates were installed early in the month and have survived two reactor scrams so far.

Primary coolant radioiodine samples still indicate the presence of about six milligrams of plutonium contamination in the flux zone (assumed average flux of 3×10^{-13} mv) of the reactor. There has been no noticable reduction in background radioiodine coolant activity since the 1962 $MgO\text{-}PuO_2$ rupture decontamination.

Procedures

Revised Operating Procedures issued	4
Revised Operating Standards issued	10
Temporary Deviations to Operating Standards issued	5
Revised Process Specifications accepted for use	5
Maintenance Procedures issued	1
Equipment Standards issued	1

	March	Total
Drawing As-Built Status:		
Approved for As-Built	38	1 312
In Drafting		25
In Approval		4
Deleted or Voided		87
		1 428
Scheduled for Review		172
Total		1 600

Personnel Training:	Manhours
Qualification Subjects	143
Specifications, Standards, Procedures	48
Emergency Procedures	23
FERTF and PRCF	68
Maintenance Procedures	28
	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 March 1964, is shown below. Those elements which had reached their assigned goal exposure or had been permanently discharged for other reasons prior to March 1, 1964, have been deleted from the table.

Test No.	Channel Location	Number	Description	Date Initial Charge	Date Dis-charged	Approximate Accumulated MWD
14	1956	5097	Moxtyl-Swaged	4/2/62	--	163.1 repad
14	1352	5098	Moxtyl-Vipac	5/8/62	--	251.8 repad
14	1758	5099	Moxtyl-Vipac	5/8/62	--	177.2 repad
48	1156	5150	Moxtyl ($\frac{1}{2}$" x $\frac{1}{2}$" pads)	8/1/62	--	172.2
54	1542	5116	Moxtyl (clip-on pads)	5/8/62	--	179.6
54	1554	5118	Moxtyl (clip-on pads)	5/8/62	--	259.8
61	1249	5185	Moxtyl-Physics	5/28/63	--	160.6
61	1354	5186	Moxtyl-Physics	5/28/63	--	152.0
61	1445	5192	Moxtyl-Physics	6/13/63	--	163.1

67	1152	5119	Moxtyl (Repaired wire)	10/20/63	--	88.5
67	1457	5117	Moxtyl (Repaired wire)	10/20/63	--	130.9
80	1544	5214	Moxtyl (1% PuO_2, Swaged)	11/18/63	--	88.6
85	1847	5230	Moxtyl (1% PuO_2, Vipac)	1/30/64	--	39.4
37	1548	1098	UO_2-Physics	5/12/62	--	152.2
37	1550	1097	UO_2-Physics	5/12/62	--	162.0
37	1451	1100	UO_2-Physics	5/12/62	--	131.8

Process Tube 6079 was removed from Process Channel 1354 and sections obtained for physical property measurements in the Radiometallurgy Laboratory.

Nineteen fuel elements were examined in the basin during the month. Broken wire wraps on one fuel element were repaired. Fuel element 5187 was disassembled; 14 rods were shipped to Radiometallurgy for sectioning.

Plutonium Recycle Critical Facility

Modifications were completed to correct several operational problems encountered in placing the PRCF light water EBWR core into service. These included: (1) installing electrical inductance suppressors in the DC circuit to correct spurious nuclear instrument and DC ground indications, (2) correcting control rod mechanical snubbers to prevent excessive de-acceleration forces, and (3) strengthening top support plate structure to prevent vibration from dislodging additional rods during partial rod drop tests.

EBWR fuel was received, inspected, and loading began on March 17, 1964. Criticality was achieved on March 27, 1964. Rod calibration experiments were in progress at month-end.

Fuel Element Rupture Test Facility

A leak developed at a weld in the cold inlet section of the Rupture Loop. Repairs are complicated by the location of the piping section which is behind heavy shielding. Repairs were initiated during reactor operation by discharging the fuel element and providing minimal flow for process tube cooling. Repair efforts continued through month-end.

Processing of Spent PRTR Fuels

The chemical processing of 35 Al-Pu fuel elements was begun at Redox.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 17,932 hours. This includes 13,390 hours performed in Technical Shops, 3,795 hours assigned to J. A. Jones Company, 743 hours assigned to offsite vendors, and 4 hours to other project shops. Total shop backlog is 13,961 hours, of which 90% is required in the current month with the remaining hours distributed over a three-month period. Overtime worked during the month totaled 118 hours or 0.6% of the total available hours.

Distribution of time was as follows:

	Manhours	% of Total
N-Reactor Department	3 002	11.2
Irradiation Processing Department	3 542	19.7
Chemical Processing Department	664	3.7
Hanford Laboratories	11 724	65.4
Hanford Utilities and Purchasing Operation	0	0

LABORATORY MAINTENANCE OPERATION

Total productive time was 18,800 hours of 20,300 hours potentially available. Of the total productive time, 74.2% was expended in support of Hanford Laboratories components, with the remaining 25.8% 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 March was as follows:

A. Shop Work		1 700
B. Maintenance		8 900*
1. Preventive Maintenance	2 500	
2. Emergency or Unscheduled Maintenance	1 200	
3. Normal Scheduled Maintenance	5 200	
C. R&D Assistance		8 200

*Includes 2,700 man hours loaned to IPD, NRD, and CPD.

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
F. G. Dawson, P. L. Hofmann, J. J. Regimbal	Reactor Core Design (HWIR-1712)
E. D. Clayton, L. L. Carter	Anti-Criticality from Pu^{240} Fueled Reactors (HWIR-1710)
W. L. Bunch, G. F. Garlick	The Detection of Wood Deterioration (HWIR-1709)
W. L. Bunch	Reactor Fuel Element Failure Detection (HWIR-1708)
D. P. Brown, G. F. Garlick, N. S. Porter	Detection of D_2O in Water (HWIR-1714)
O. H. Koski	Continuous Analysis of Organic Solutions for Uranyl Nitrate
E. A. Eschbach	Patent Disclosure: Preliminary Consideration of Milk Bottle Reactors Utilizing Excited Mercury as Primary Heat Transfer Device

H M Parker

Manager, Hanford Laboratories

END

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