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HANFORD LABORATORIES OPERATION MONTHLY ACTIVITIES REPORT

MARCH, 1962

APRIL 16, 1962

**HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON**

GENERAL ELECTRIC

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This document consists
of 212 pages.

HANFORD LABORATORIES OPERATION

MONTHLY ACTIVITIES REPORT

MARCH, 1962

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By Authority of CG-PR-2

RM Iken 8-6-92

J. Tancy 8-18-92

DJ Kwohac 8/20/92

Compiled by
Operation Managers

April 16, 1962

CLASSIFICATION REVIEW
DECLASSIFICATION
UNCHANGED

By J. P. R. R.
Date 5/15/23
Reg. Division of Classification

HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

PRELIMINARY REPORT

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

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TABLE I. HLO FORCE REPORT

	DATE: March 30, 1962			Total
	At beginning of month		At close of month	
	Exempt	Salaried	Exempt	Salaried
Chemical R & D	132	115	131	123
Reactor & Fuels R & D	174	152	176	154
Physics & Instrument R & D	88	61	92	59
Biology	36	51	36	53
Operations Res. & Syn.	17	4	17	4
Radiation Protection	41	90	40	89
Finance and Administration	116	102	107	99
Programming	17	3	17	3
General	3	4	3	4
Test Reactor & Auxiliaries	46	182	49	182
TOTAL	670	764	668	770
				231
				1,438

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BUDGETS AND COSTS

March operating costs totaled \$2,320,000, an increase of \$123,000 from the previous month; fiscal year-to-date costs are \$19,910,000 or 72% of the \$27,784,000 control budget. Hanford Laboratories' research and development costs for March, compared with last month and the control budget are as follows:

(Dollars in Thousands)	C O S T			Budget	% Spent
	Current Month	Previous Month	FY To-Date		
HLO Programs					
02 Program	\$ 49	\$ 39	\$ 397	\$ 605	66
03 Program	23	4	27	175	15
04 Program	949	894	8 188	10 955	75
05 Program	121	98	716	1 088	66
06 Program	237	194	1 804	2 637	68
	<u>1 379</u>	<u>1 229</u>	<u>11 132</u>	<u>15 460</u>	<u>72</u>
FPD Sponsored	115	96	1 005	1 400	72
IPD Sponsored	138	124	1 017	1 348	75
CPD Sponsored	<u>157</u>	<u>125</u>	<u>1 224</u>	<u>1 609</u>	<u>76</u>
Total	<u>\$ 1 789</u>	<u>\$ 1 574</u>	<u>\$14 378</u>	<u>\$19 817</u>	<u>73</u>

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

An experimental single tube, dual-enriched, metallic uranium fuel element successfully completed one cycle of irradiation in the ETR.

Examination of the NPR tube-and-tube fuel element after an additional cycle of irradiation at the ETR revealed that one of the "buggy spring" supports on the inner tube failed by fatigue after one cycle of irradiation.

Studies of heat treatment of cast uranium ingots preceding NPR billet fabrication indicate that an alpha soak followed by a single beta heat treatment may be superior to the present triple beta heat treatment.

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To test the hypothesis that substituting extrusion for forging as the primary hot working method of Zircaloy-2 ingots will improve N-fuel clad uniformity, an ingot was immersion copper plated and extruded at 1300 F. The extrusion was machined into five billets for fabrication into co-extrusion components.

Examination of the graphite samples irradiated in the GETR to a maximum exposure equivalent to 5200 MWD/AT in N-Reactor has revealed that the samples are in excellent condition. All samples and flux monitors were successfully recovered. A new capsule was installed in the GETR and the thermocouples indicate that the capsule is functioning properly. The wear and fatigue problems have apparently been corrected.

Several modifications were made to the N-charging machine to improve performance. Design testing continued with charging of fuel elements from a magazine, through a simulated nozzle, and into a simulated process tube. Mechanical damage to numerous fuel element support feet was observed. A modification to the adapter between the nozzle and magazine apparently cured this difficulty.

Automatic measurement of chloride and phosphate concentration in once through reactor coolant was successfully demonstrated at an accuracy of ± 5 ppb in the 0-100 ppb range.

Heat transfer experiments simulating rupture of the inlet piping of a C-Reactor process tube indicate that rear header pressures of about 60 psi will assure sufficient cooling water from the rear header to prevent melting of aluminum jackets.

Visual studies of electrically heated models of production reactor fuel elements indicate that local boiling is rather easily initiated under the supports of self-supported fuel pieces.

In laboratory studies of flow mixing it was found that the mixers presently used in the production reactors efficiently interchange the hole and annulus flow but that a new helical mixer under study does a better job of inter-mixing the annular flow.

A comment draft of the Fuel Element Rupture Loop Final Safeguards Analysis was issued. The analyses conducted lead to the conclusion that

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the presence of the rupture loop in the PRTR will not materially increase the hazard of operating the reactor. Operational safety reviews of the PRTR and the Plutonium Recycle Critical Facility were conducted by the General Electric Technological Hazards Council.

Irradiation of plutonium-aluminum fuel elements in the PRTR is limited to those amounts and compositions of plutonium where the reactivity value is less than that provided by 330 grams of Pu-239 in the cold-clean condition. The variation of reactivity of Pu-Al fuels containing up to 29.6 per cent Pu-240 was investigated. No increase in reactivity with increasing exposure was indicated for any of these cases under the assumed PRTR neutron spectra. Details of this investigation are presented in HW-73107.

The EDEL-I flow loop for testing the rupture loop in-reactor test section has achieved operational status. Design of special equipment and tools for removing ruptured fuel elements from the rupture loop continues. Most of the necessary material has been ordered, and fabrication of some items is in progress.

A proposal has been submitted for irradiation testing in the PRTR of one or more liquid poison solutions planned for use in liquid shim assemblies. Design work continued on second generation PRTR mechanical shim rods with emphasis on solving known problems on the present rods.

Seven PRTR pressure tubes were inspected during the month. The minimum gas gaps in two channels were found to be about 35 mils. Application of mechanical force increased the minimum gaps to 100 mils, but the gap returned to about 35 to 40 mils when the force was removed. The process tube in one channel was rotated 180 degrees which increased the gap to about 90 mils.

Wear corrosion marks at contact areas with the PRTR fuel element end brackets measured in five of the tubes inspected varied from about 0 to 6 mils deep. Contact with the outer spiral wire wrap caused grooves up to 4 mils deep. A new tube installed in October, 1961, had grooves up to 2 mils deep.

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An initial group of 120 cold-swaged UO_2 - PuO_2 fuel rods for PRTR use have been fabricated. A method for preventing weld area contamination was developed. The information obtained from the fabrication of this initial group of rods has resulted in several process modifications which will permit an increased fabrication rate and reduced over-all costs. Fabrication of vibrationally compacted UO_2 - PuO_2 fuel rods has been started.

A 42-inch long UO_2 - PuO_2 cosine-enriched seven-rod cluster element has successfully completed 27.3 days of operation in the ETR 3x3 high temperature loop, generating 30 per cent more power than a uniformly enriched fuel with the same maximum core temperature.

Autoradiographs of irradiated incrementally loaded mixed oxide fuel rods operating at a power of 345,000 Btu/hr-sq ft. and a maximum core temperature of 1850 C show no evidence of longitudinal relocation of the plutonium content from the plutonium-rich increments.

Eight MgO - PuO_2 and eight ZrO_2 - PuO_2 fuel capsules have successfully completed their first cycle of irradiation in the ETR.

Phoenix fuel experiments continued to accumulate exposure in the MTR. One composition has completed four cycles, another three cycles, and a third composition is starting its second cycle. Four aluminum-plutonium-boron poison standard samples have been made for calibration of the ARMF.

Investigation of the stoichiometry of plutonium carbides continued. Low carbon PuC was found to disintegrate in dry hydrogen at room temperature due to grain boundary attack. High carbon samples were more resistant to attack.

Plutonium nitride synthesized by arc melting plutonium in a nitrogen atmosphere did not melt at 2650 C in helium although volatilization occurred.

Variation of plutonium concentration by a factor of two within a radial distance of 0.050 inch was observed in a sintered UO_2 pellet irradiated to 900 MWD/ T_U . Nearly an order of magnitude variation of Cs-137 concentration was observed in the same pellet.

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The first examination of an irradiated (5000 MWD/T) UO_2 single crystal by reflection electron microscopy revealed fracture surfaces only slightly different than those observed in single crystals of non-irradiated UO_2 .

Extended duration optical microscopy of molten UO_2 was made possible by increasing the ambient helium pressure to 1500 psig.

Irradiation of four controlled-temperature swelling capsules is continuing. An irradiated uranium-uranium diffusion couple revealed no evidence of diffusion of fission products into the depleted uranium after annealing at 650 C for 100 hours.

In-reactor creep activation energy measurements were completed on a Zircaloy-2 specimen in a creep capsule. Measurements were continued until a low ductility fracture of the specimen occurred at approximately five per cent total strain.

Tensile tests were conducted on unirradiated Zircaloy-2 at 300 C and at various strain rates to determine the breaking stress, notch strength, and crack toughness. Strain aging was observed at 300 C at rapid loading rates.

Burst tests conducted on Zr-2 tubing showed that notches deeper than 20 per cent of the wall thickness markedly reduced uniform elongation.

Experiments at temperatures as low as 150 C confirm that the oxidant level of the gas environment is the primary factor controlling the hydriding of zirconium. Temperature, hydrogen pressure, metal surface roughness, and prior ZrO_2 films all affect hydriding rates but are not determining factors as to whether zirconium will or will not hydride.

The suppression of graphite combustion by chlorine was tested in the EGCR burning rig. With graphite burning in an air stream at 775 C, approximately 6 w/o of chlorine was required to stop the runaway combustion.

The first in-reactor graphite creep irradiations in the Hanford reactors at 150 and 300 psi (1.0×10^7 and 2.1×10^7 dynes/cm²) indicate that after an exposure of 1500 MWD/AT the creep rate at about 650 C is in

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the range 9×10^{-12} to 3×10^{-11} per dyne/cm² per 1000 MWD/AT. Approximately half of the rate is attributable to radiation-induced creep and half to thermal creep.

An improved method for handling NaK from irradiated fuel elements has been introduced at the Radiometallurgy facility. Elements are opened under kerosene and cellusolve (ethylene-glycol-mono-ethyl-ether) is added to react with the NaK. Reaction products are soluble, leaving a clear liquid, easy to handle and discard. Fire hazard is almost eliminated by this process.

Conceptual design activity during the month was focused on a liquid metal-cooled solid fuel reactor for generation of electric power on space vehicles.

2. Physics and Instrument Research and Development

Recent theoretical studies predicted that the neutron temperature in a 10.5" graphite moderated lattice is 44°C higher than the graphite physical temperature. This agrees well with corresponding experimental values ranging from 38°C at the cell edge to 62°C at the fuel-moderator boundary.

The exponential pile mockup of the NPR lattice is being rebuilt to correct an error in control rod hole spacing. This change will alter the control rod strength results but will not affect the previously completed buckling and flooding effect measurements.

Experiments at the Critical Mass Laboratory showed that a cadmium shell around a 14" diameter vessel increases the critical volume 10-13 per cent for plutonium concentrations ranging from 33 to 321 grams per liter. Other measurements were completed on the effects of 3.5" and 1.0" paraffin and 4" concrete reflectors.

Theoretical advances in critical mass physics included: Development of methods for correcting experimental criticality data on partially filled spheres, explanation of reactivity effects observed upon insertion of a control rod into a criticality vessel, and encouraging results from calculations of interaction effects between subcritical assemblies using two-group diffusion theory in the HFN code.

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NPR diffusion length measurements and graphite stack nuclear purity tests were successfully completed March 3 and 4 with the instrumentation system we developed and provided for IPD.

Tests were started on the first commercially fabricated model of the gamma energy analysis system we developed for the NPR Fuel Failure Monitor. Recommendations were also made regarding design changes to improve the reliability of the KER Fuel Failure Monitor, and design of a new monitor for a KE test hole experiment.

Technical planning for the NPR simulator included a visit to Bailey Meter Company to obtain the dynamic characteristics of instruments and control valves to be used in the NPR plant, a visit to obtain information about the Fermi Reactor Simulator, and the collection of information needed to evaluate digital methods of simulation for comparison with the analog approach which has been developed in considerable detail.

Further technical analyses and consultations were provided IPD regarding the NPR Flow Monitor and Zone Temperature Monitor systems.

A contribution was made to improve understanding of the NPR heat dissipation system through completion of an analog analysis of the stability of the secondary loop. Additional analog reactor speed-of-control studies for the old reactors were completed to aid in predictions of reactor behavior following postulated positive reactivity disturbances.

Analysis of the "fuel reuse" concept where a fuel element from a fast reactor blanket, low in Pu-240, is reused in a thermal reactor indicates that a single interchange might reduce the fuel cost by 0.3 to 1.0 mill/kwh depending on the circumstances involved.

On the basis of nuclear analysis only, weight savings of 20-35 per cent may be possible in a reactor for space propulsion by using Pu-239 carbide rather than U-235 carbide.

Buckling and critical mass studies have now been completed for half-length PRTR fuel rods in light water moderated lattices of four different spacings. These studies and fuel irradiations at the PRTR for burnup experiments are continuing.

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In reactor physics computer code development, MELEAGER was modified to handle up to four independent self-shielded materials in addition to the previously available self-shielding in the fuel, and to extend the range of usefulness of the "r" value calculation.

Fabrication of the prototype liquid effluent activity monitor for the PRTR containment trip circuit is nearly completed; the completed portions have tested satisfactorily.

In the Neutron Flux Monitor program, calculations showed that Nuclear Magnetic Resonance and Electron Paramagnetic Resonance are not practical methods for detecting neutron-induced changes in a gas irradiated at normal power reactor flux levels; but that Microwave Spectroscopy does offer possibilities for flux values above 5×10^{12} nv.

Changes have been devised for the developmental multiple-parameter eddy current nondestructive test system to reduce undesirable inter-channel effects observed during attempts to separate three and four test parameters.

An invention disclosure was submitted on a method for providing indicators of maximum temperature for use, for example, in the heat treatment of weldments for stress relief. Simple measurements of the widths of X-ray diffraction lines from sandblasted metal coupons are used.

In the development of improved methods for ultrasonic testing of thin-walled fuel sheath tubing, excellent agreement was obtained between the experimental response to varying test parameters and the theoretical analysis in terms of Lamb wave propagation.

Accelerated development continued on several types of pocket-size dosimeters which both indicate the accumulated dose and audibly alarm at a pre-selected dose level. Encouragingly low bids were received for off-site fabrication of one model of modified pencil ionization chamber detector. A miniature scale-of-32 circuit was also developed to replace a larger electromechanical register.

A new transistor amplifier was developed to permit use of more rugged and reliable panel meters in certain radiation monitoring instruments.

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The atmospheric diffusion programs at Cape Canaveral and Vandenberg Air Force Base was resumed after major delays at both sites. The off-site Cape Canaveral work was completed on March 31 with a total of 76 experiments conducted there. At Vandenberg, the extensive damage done by weather was completely repaired and the second series of experiments completed on March 31. A third series will be conducted at VAFB upon completion of a weather information network and display system.

Studies of precipitation-scavenging processes in the atmosphere progressed with reduction of several raindrop spectra; these are in flux density form and as such may be used directly in scavenging predictions for the light winter rains at this site.

Improvements are being designed for the Whole Body Counter to halve the counting time. A shadow shield is also being designed for a mobile counter to be mounted on a truck.

Corrections for the radioactive decay of Sb-Be neutron sources will be improved through completion of a very accurate measurement of the half-life of Sb-124 (60.20 ± 0.03 days at the 95% confidence level).

A large version of the recently developed process tube distortion traversing gage has been designed for use to determine the alignment of lateral castings to be installed under the high level waste tanks in the 200 Areas.

3. Chemical Research and Development

Laboratory studies show that the effectiveness of silicate in reducing phosphorus-32 formation in reactor effluent water could be as much as 50 per cent greater in reactors operating at pH 7 than in those at a pH of 6.6, depending on the silicate concentration in the process water.

The alum addition rates at the two K-Area reactors have been reduced from 18 to 9 ppm to optimize the removal efficiency of radioisotope parents. Shortened filter runs are also being used to keep the turbidity of the water to 0.003 ppm or less.

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In support of RMC-Line, 234-5 Building Operations, an anion exchange flowsheet was developed in the laboratory as a way to simplify the recovery of plutonium from plutonium oxalate supernatant solutions.

Engineering studies have failed thus far to pinpoint the cause(s) of the recent poor operating performance of the Purex 2D and 2E solvent extraction columns.

Sampling of the contaminated sediments in the bottom of the 216-Z-9 Recuplex crib by core-drilling is planned. Tests of equipment and procedures are being made in uncontaminated soil adjacent to the crib.

A flowsheet for the removal and purification of cesium-137 from Purex formaldehyde treated waste was developed. The process includes absorption of Cs-137 on clinoptilolite, removal of sodium which was retained by the clinoptilolite with dilute nitric acid, elution of the cesium with 2 M ammonium acetate, and volatilization of the ammonium acetate by boiling to leave a solid residue of a cesium salt.

Hot cell studies with actual Purex process streams disclosed the cause (excessive amounts of iron) and the remedy (sulfate precipitation at a lower pH) of difficulties which had hampered recent Purex strontium-90 recovery runs.

Tritium was detected in a water sample from a well located eight miles north of the 300 Area and one-half mile from the Columbia River. Ground water temperature measurements along with tritium and gross beta analyses from wells intervening toward 200 East Area substantiate this as one of the preferential paths for the 200 East Area contaminated ground water movement.

A Purex plant test to evaluate the effectiveness of uranium(IV) as a plutonium reductant is planned. To obviate column modifications, the uranium(IV) may be added via an organic, vice aqueous, feed stream. A procedure was developed for preparing the required uranium(IV)-bearing organic stream.

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Engineering studies show promise that superimposition of an alternating current on the normal direct current used to prepare electrolytic UO_2 from molten salt baths may provide means of internal heating of the melt.

Irradiated Pu-Al alloy shows activation and dissolution rates in nitric acid-mercuric nitrate systems which are substantially less than those obtained with irradiated alloy. Activation and dissolution can be effected at lower nitric acid concentrations but with the penalty of increased hydrogen gas evolution.

4. Biology

A study of reactor effluent toxicity was started at K Area in which chinook salmon fingerlings will be exposed to 3, 5, and 7 per cent reactor effluent.

It was observed that EDTA in the gut markedly reduces Ca absorption, but has little effect on Sr absorption. This further bears out predictions made on the existence of two or more acceptors for absorption of these elements. In other studies it was found that gut absorption of Np and Pu nitrates differ by a factor of 10, with Np absorption being greater.

DTPA aerosol inhaled by rats after their exposure to $\text{Ce}^{144}\text{O}_2$ decreased their lung burdens to 7 per cent of controls. This may be a major breakthrough for the treatment of particulate contamination.

Other programs on study of biological effects of radioisotopes in animals continued, and work on the small animal metabolism of Tc-99 was started. A miniature beta probe was used with some promise to determine dose rates on surfaces of bone sections from animals fed Sr-90. Thyroid carcinoma was found in a seven-year-old ewe fed 1.5 μc I-131/day since birth.

5. Programming

Studies of the irradiation of mixed plutonium-uranium oxide fuel in PRTR indicate that the reactivity losses during irradiation are caused primarily by depletion of U-235, and that recycle of plutonium into the

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same amount of natural uranium will restore reactivity to its initial value essentially independent of initial plutonium composition or of the amount of irradiation (for low values of Pu-242).

TECHNICAL AND OTHER SERVICES

Tests of computer generated and compactified magnetic tapes on the prototype Gorton Lathe Control under development at the Manufacturing Services Laboratory were completely satisfactory. The technique for preparing these tapes appears to be both feasible and practical, and the tapes were capable of activating the pulse motors within the specified pulse-density restriction.

Results from further analysis of more extensive Quality Certification data indicate that somewhat simpler models than previously used will still provide a good relationship between fuel element distortions during irradiation and reactor variables.

Work continued on several reliability studies including an analysis of defects in NPR welded primary piping and an analysis of various reactor safety systems employing a "k out of n" type logic.

A review is being made of existing means of controlling the Pu-240 content of the final product.

Mathematical expressions have been developed and are being evaluated which express the self-interaction coefficients for annulus shaped containers used in storing fissile materials.

Preliminary discussions were held concerning experimental designs appropriate to the various physical properties measurements to be made as part of the forthcoming program to investigate irradiation effects on structural materials.

A study is being conducted of the characteristics of particle sieving from a probabilistic and statistical viewpoint.

Concentrations of fallout, as measured in Pacific Northwest air samples, decreased for the second consecutive month. Average values for the first three months of this year were 6.1, 5.3, 4.4 $\mu\text{c}/\text{cubic meter}$, respectively. The observed concentrations during March ranged from 1.4 to 8.5 $\mu\text{c}/\text{cubic meter}$.

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Two new cases of minor plutonium deposition (estimated at less than 0.5% mpbb) were confirmed by bioassay during the month. The total number plutonium deposition cases that have occurred at Hanford is now 285, of which 205 are currently employed. The two new deposition cases resulted when HLO Plutonium Metallurgy employees were exposed to air-borne plutonium during a plutonium carbide melting operation at 231-Z Building.

A survey of the Columbia River beaches and islands between Richland and Ringold revealed a maximum dose rate of 1.1 mr/hour and ground contamination up to 7,500 c/m. These values are about the same as measured last summer and fall.

There are 11 currently active projects having authorized funds amounting to \$2,672,000. The total estimated costs of these projects is \$7,581,000. Expenditures on them through February 28, 1962 were \$952,000. Two new project proposals, covering a Small Particle Technology Laboratory and a Graphite Machining Shop, were submitted to the Commission.

SUPPORTING FUNCTIONS

The PRTR output was 605 MWD for a plant efficiency of 26.8% and a total experimental time efficiency of 33%. Accumulated exposure to date is 6217 MWD. Additional exposure-to-date information is as follows:

Maximum UO ₂ exposure/element	1930 MWD/TU
Average UO ₂ exposure/element	1288 MWD/TU
Maximum Pu-Al exposure/element	79.0 MWD
Average Pu-Al exposure/element	39.4 MWD

The eighth refueling was accomplished on March 6, 1962. Four new LX Pu-Al elements were charged around the outside of the Pu-Al zone. The loading configuration of 39 Pu-Al and 46 UO₂ elements remained as before.

Heavy water losses chargeable to PRTR operating cost for the month of March amounted to \$7,650 -- a significant reduction from last month. Total scrap accumulated this fiscal year amounts to \$239,000 of which an estimated 12 per cent will be reclaimed here.

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Work was completed on the Plutonium Recycle Critical Facility, with exceptions, and the Operating Procedures and a Training Guide were issued. Over-all construction of the fuel element rupture test facility is 88% complete, and work on the water plant contract is 75% complete. Construction work on the gas-cooled loop has been suspended pending receipt of the new heater scheduled during May.

Recruitment - In the advanced degree area, eight Ph.D. applicants visited HAPO for employment interviews, six offers were extended, two acceptances and six rejections were received.

In the BS/MS area, forty-two offers for the Technical Graduate Program were made; ten acceptances and 14 rejections were received.

Five offers were made for direct placement, with one acceptance and one rejection.

Technical Graduate Program - Eight technical graduates were placed on permanent assignment and two members terminated from the rolls leaving a total of 44 members currently on the program.

W.K. Woods

for Manager
Hanford Laboratories

HM Parker:WKW:st

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REACTOR AND FUELS RESEARCH AND DEVELOPMENT OPERATIONTECHNICAL ACTIVITIESA. FISSIONABLE MATERIALS - 2000 PROGRAM1. METALLURGY PROGRAMCorrosion Studies

Samples of Zircaloy-2 and Zircaloy-4 were removed from KE Reactor after an exposure of about 5×10^{20} nvt. During irradiation the pre-autoclaved samples had been exposed to a simulated NPR gas atmosphere at 290, 350 and 400 C. The irradiated samples are being prepared for hydrogen analyses. Samples exposed out of reactor to the same environment did not pick up any hydrogen from the atmosphere.

Erosion-Corrosion of Aluminum Alloys. Samples of aluminum alloy 1245 were exposed to 300 Area tap water containing various amounts of sodium dichromate at a flow velocity of 76 feet a second and a temperature of 102 C. Dichromate concentrations were 1 ppm and 0.7 ppm. Exposure times were 24, 48, and 72 hours. No corrosion was seen after 24 hours of exposure at either dichromate concentration. Samples of alloy 1245 exposed for 48 hours and 72 hours showed a slight pitting attack in the area covering the water intake hole compared to severe pitting attack that occurred on samples of X-8001. At dichromate concentration of 0.7 ppm, alloy X-8001 had shown pitting along the exposed groove area of highest flow, but alloy 1245 did not.

Effect of Heat Treatment on the Corrosion and Hydriding of Various Zirconium Alloys. The effect of heat treatment on the corrosion and hydriding of Zircaloy-2, low nickel Zircaloy-2, a heat of ultra fine grained low nickel Zircaloy-2, Zircaloy-4, and crystal bar zirconium is being investigated. Each alloy is being tested in the beta quenched, alpha plus beta quenched, alpha annealed, and as-received conditions. The various heat treatments produced virtually no effect on the short exposure corrosion rates or hydriding characteristics of the various alloys with the exception of the crystal bar zirconium. Alpha annealing lowered the corrosion rate of crystal bar zirconium from 700 mg/dm² for 3 days of exposure to 400 C steam to 55 mg/dm² after 73 days of exposure.

Following the corrosion transition, a very noticeable hydriding effect from the various heat treatments can be seen. The various alloys

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A-2

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which were beta quenched or alpha plus beta quenched have shown no increase in the percent of hydrogen pickup, whereas the alpha annealed and as-received material have shown an increase of a factor of two.

Metallurgy Studies

Zirconium Components for Coextrusion. The program to determine if improved N-fuel clad uniformity results when Zircaloy components are made via a primary extrusion of ingots rather than forging is on schedule. The machined ingot was received at Hanford, immersion plated with copper and successfully extruded at 1300 F using a trade lubricant, Fiske 604. The extruded surface was exceptionally smooth with a surface roughness of about 125 microinches. Five billets were machined from the extrusion and shipped off-site for conversion into outer tube inner components. Fifteen components should be available in early May.

Metallic Fuel Development

Fuel Irradiations. The NPR tube-in-tube fuel element which is being irradiated at the ETR has been measured after a total average exposure of 700 MWD/T. The outer tube has shown an average increase in diameter of 0.004 inch. The inner tube, which was irradiated during only the last cycle and has 150 MWD/T average exposure, showed an average diameter increase of 0.001 inch. The warp of the outer tube reversed again during the last cycle of irradiation, as was expected, since the fuel element was rotated 180 degrees from the previous cycle. One of the "buggy spring" supports on the inner tube failed, presumably from fatigue. For the next cycle of irradiation the inner tube will be replaced by an inner tube having "suitcase handle" supports in order to evaluate the performance of this support design.

Autoclave data, dimensional measurement data, residual stress, end profile, braze zone microstructure of the cladding, density data, and metallography data have been collected for the prototype N-fuel elements to be irradiated in KER Loops 3 and 4. Twenty-eight irradiation quality inner elements have been obtained from the CX-2 test lots. Since 44 are needed for the KER Loop charges, production elements will be used to complete the makeup of the charges. It is planned to use six CX-2 inners and five production inners in the 1000 MWD/T charge in Loop 3 and a larger number of CS-2 inners in the 2000 and 3000 MWD/T charges. The 1000 MWD/T test will be charged at the next reactor outage.

An experimental single tube, dual-enriched metallic uranium element has completed one cycle of irradiation in the M-3 pressurized water loop in the ETR. Peak power generation in the element during this period was 180 kw/ft with a corresponding maximum metal temperature of 565 C. A maximum exposure of 300 MWD/T was accumulated. The element was removed from the reactor and will be examined before recharging for additional exposure.

A thermocoupled fuel rod (crud probe) has now successfully accumulated 770 MWD/T of exposure in KER Loop 1. Water temperature has been 252 C, core temperatures have been 605 C to 640 C, and surface heat fluxes have been in excess of 600,000 Btu/hr-ft². During the next outage the loop is to be decontaminated, and the probe will be used to monitor possible crud depositions during subsequent loop operation.

Radiometallurgical examination has begun on the production brazed irradiation test from the MTR, GEH-4-63 and 64. These samples underwent six cycles of irradiation for a total burnup of approximately 1400 MWD/T at a specific power of 62 kw/ft. No indication of imminent failure was observed on the surface of the elements, but an area was noticed around the two end caps in the high flux region (one end cap on each element) which appeared to be a stressed area in the cladding. These areas are being sectioned to determine the integrity of the brazed closure and the cause of the stressed-appearing area.

The variable braze thickness test, GEH-4-68, 69, and 70, has completed five cycles of irradiation at the MTR for a total burnup of approximately 1140 MWD/T. A sixth cycle has been requested.

Dimensional Measurement of Tubular Fuel Elements. A method of analysis of measurement data has been programmed to analyze the data in terms of mean diameters, displacements of centroids of inner and outer surface, ovalities, warps, and changes in the circumferential curvatures. Experimentally obtained data contain random errors (noise) which are associated with the measuring system. The method of analysis was so formulated that a noise analysis could be run after sufficient data had been collected. A noise analysis, using the data collected to date, indicated the data had a random error below one mil. Also, approximately eight Fourier coefficients are independent of the noise.

Underwater Measuring Equipment. The equipment for measuring irradiated N-fuel elements is being constructed according to schedule. Present planning indicates that the equipment will be ready for installation in the KE basin the latter part of June, 1962. This equipment will measure and record ID, OD, warp and wall thickness of the irradiated tubes.

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Coextrusion Studies. Nine, 0.6-inch diameter coextrusions at 13 to 1 reduction have been made to study the effect of billet component properties on the clad-core interface. The cladding components were made from extruded unalloyed zirconium and from extruded zirconium - 3% Nb - 2% Sn alloy. The brinell hardnesses were 117-128 for the pure zirconium and 220-230 for the alloy compared with 170-190 for Zircaloy-2. The uranium cores were made from 14:1 alpha extruded normal ingot and from gamma heat treated rolled dingot. The latter was used to simulate the coarse grained structure of a cast billet which produces extremely rough interfaces in coextrusion. Five of the billets were canned in copper, two in 1010 steel, and two were extruded bare. The zirconium on all the billets was sealed by vacuum electron beam welding. The following table shows the billet assembly combinations and the results of visual observation of extrusion cross sections. In all cases the harder clad has a smoother interface than the softer. In all cases the bare and steel cans afforded smoother interfaces than the copper cans. At 500 C, where the pure zirconium and uranium have the same extrusion constants (equal stiffness), the interface is rougher than that of the same billet extruded at 620 C. Longitudinal sections have not yet been prepared but may show different relative roughnesses than do the cross sections.

<u>Billet Assembly</u>	<u>Interface Relative Roughness</u>
P-D-Cu	10
A-D-Cu	8
*P-E-Cu	7
P-E-Cu	5
P-E-S	4
P-E-B	4
A-E-Cu	3
A-E-S	2
A-E-B	1

P - Pure zirconium clad
A - Alloy zirconium clad
D - Gamma heat treated dingot core
E - Alpha extruded ingot core
Cu - Copper can
S - Steel can
B - No can

*Extruded at 500 C preheat -- all others at 620 C.

Relative roughness

10 = Very rough
1 = Smooth

Clad Deformation Studies. In studying the effects of surface irregularities on the performance of Zr-2 cladding on fuel elements, various forms of notched tensile specimens and tube burst specimens are being employed. It is intended to determine the effect of thickness variations on fuel element cladding behavior. Burst test data were obtained on extruded Zr-2 tubing containing longitudinal notches on the inner surface. These specimens, which were in the annealed condition, were tested at room temperature with hydraulic pressure. Under this stress condition, the Zr-2 displayed none of the notch strengthening that has been observed in tensile specimens. Both large radius and sharp notches effectively reduced the circumferential elongation of the specimens. Notches which were deeper than 20 percent of the wall thickness allowed a uniform elongation of less than five percent, whereas unnotched specimens strained over 20 percent before failure. Additional testing of notched tubes at room and elevated temperature is scheduled.

A method was devised for testing the irradiated 0.593-inch diameter cladding specimens which are not suitable for burst testing. An expanding mandrel was built which performed adequately on unirradiated specimens. This mandrel will be used for room temperature testing of the irradiated specimens.

Fuel Component Development. Room temperature fatigue tests were made on autoclaved inner fuel tube supports for N Reactor fuel elements. When the supports held a load equivalent to the weight of an inner fuel tube and were vibrated through an amplitude of 0.020 inch, failure occurred after 10^5 to 10^6 cycles. Reducing the vibration amplitude to 0.006 inch increased the fatigue life to the extent that tests terminated after 10^7 cycles did not produce failure. Support designs with improved ability to sustain dynamic loads are being investigated.

A range of soft elasticity was built into the inner supports so the inner fuel tube could be assembled into the outer with an interference support fit and with a minimum of scratching of the autoclaved surface. A method of sizing the supports upward after the fuel tubes are assembled is being tested. If this method works satisfactorily, it will obviate the need for soft elasticity in the supports and the need for careful control of support circle diameter before they are assembled.

Component Fabrication. Fifteen, NOE uranium coextrusion billets were beta heat treated as a final fabrication step before coextrusion. The treatment consists of a single beta phase heating and quench.

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Two, 11-inch diameter, 19-inch long NOE castings of Fe-Si additive material were received for heat treatment, primary fabrication, and coextrusion. One ingot was given a 15-hour alpha soak at 540 C (1000 F) followed by a single beta heat treatment, consisting of one hour, 15-minute immersion in 730 C chloride salt, 50-second transfer to 20 C circulated water quench. Primary extrusion, #Pu-65, was completed by Coextruded Products Engineering, sectioned for macro examination, and two coextrusion billet lengths machined. Examination of the macrostructures and of non-destructive billet tester results indicate that the structure is at least equivalent, or superior, to those billets processed by the standard production process. Coextrusion evaluation will be made in April, 1962.

Extrusion Development. Fifteen, 5/8-inch diameter rod coextrusions containing 1.47 percent enriched uranium were made at a reduction of 13 to 1 for use by FEDO in cladding failure mechanism studies. Longitudinal grooves of various sizes and shapes were machined into the Zircaloy-2 sleeve ID to produce the desired striations at the Zircaloy-2 uranium interface. Longitudinal interface striations of the desired size and shape were previously produced by this method using a slightly altered billet design and extrusion technique.

The extrusion billets were lubricated with Fisks 604 (an aluminum particle suspension) and graphite which produced a very smooth external surface on the extrusions. Lubricants previously used on extrusions of this type tended to "ball up" on the extrusion surface, producing rough areas in the extrusion copper which reflected onto the Zircaloy-2 surface of the coextrusion.

Fuel Closure Development. Techniques are being developed for the vapor deposition of pure beryllium on zirconium end caps with two possible applications in mind. One application is the preplacement of beryllium on the end caps to form the Zr-5 w/o Be brazing alloy by diffusion in a high temperature operation. This would require Be films 0.001 inch or more in thickness. The second application would be the use of thin films of Be as an initial bonding agent in the projection welding of end caps to hot-headed fuel elements and would probably require films much less than one mil thick. The vapor deposition technique has now been developed to the point where high quality tightly adherent Be films can be consistently deposited with a fair degree of control up to film thicknesses of 0.005 inch. Thicker films are possible and should be obtained shortly.

Supporting Fabrication. Increased penetration is an advantage of welding copper using nitrogen rather than argon or helium as a shielding gas. Porosity appears to be a problem when using

nitrogen. Davis and Terry⁽¹⁾ claim that this can be overcome. The increased penetration has been verified at Hanford. Some porosity was evident in the welds. It remains to be determined whether this porosity can be reduced sufficiently to produce acceptable extrusion billet closures.

Closure and Joining. Several methods of applying a thin, electrically insulating, hard, ceramic lining to steel tubes designed to restrain the expansion of NPR fuel elements during closure bonding by use of the Heavy Duty Sciaky spot welder have been experimentally tried. Since it appears that such a lined tube may be used only once or twice for this purpose before the lining must be replaced, a simple and quick method of applying the lining is essential. The best way found to date for lining these tubes is to make a paste by adding alundum flour to Sauereisen cement and spreading it on the inside wall of the tube. After air drying, the tube must be carefully baked in an oven with temperature gradually increased from 50 C to 200 C over a period of several hours. Too rapid heating results in a blistered coat which renders the tube useless for the intended purpose. The preparation of such a tube is currently in progress. A few fuel element specimens with closures bonded by the spot welder process have been found soundly bonded in the ultrasonic test, and this approach appears to be the most advantageous of several now being exploited.

One alternate process that appears promising, if a suitable restraining die can be developed, is that of applying longitudinal restraint by means of a low expansion draw bolt (416 S/S) while induction-heating the closure end of the element inside a ceramic die of proper size. This approach has been tried with some success, but the ceramic die was broken by the thermal shock. The procedure was tried without the die, resulting in some sound bonds, but with undesirable bulging of the element for an inch or so below the cap. As a variation, the element with the draw bolt was heated to ~850 C and hot swaged. This yielded a satisfactory diameter, but a poor cap-core bond. With better control of the processing conditions, there is a good probability that this method would yield a satisfactory product. Experimental work is being continued.

To reduce the number of variables in the projection welded-brazed closure the center wedge portion of the cap was removed. With the wedge removed, Zr-to-Zr fusion at the projections can be studied without interference from uranium contact. This type of cap has

(1) British Welding Journal, February, 1954, E. Davis and C. Terry, "Nitrogen Arc Welding of Copper".

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shown that the projections were being overheated by plain resistance heating and interface heating between the projections was insufficient to initiate fusion. Also, the heating rate of the ID projection is greater than the OD projection which results in inadequate follow-up for the ID. Resistive heating is being eliminated by altering the weld schedule and uniform heating rates will be attained by decreasing the OD projection width.

Fuel Straightening. A group of eight NIN (Group B) elements used for previous straightening experiments were beta heat treated to determine effects of beta heat treatment after straightening. Straightening history is:

- a. Extrusion (Fuels Fabrication Development)
- b. Beta heat treatment (N-fuel standard)
- c. Hot roll straighten from 575 C
- d. Vacuum anneal at 600 C, 4 hours
- e. Autoclave 72 hours at 400 C
- f. Autoclave additional 72 hours at 400 C.

The beta heat treat induced warp vectors lie in nearly the same plane in two cases. In the remainder of the elements the warp vectors produced by a second beta heat treatment scatter randomly from the ones produced in "b" above. The data were plotted from steps "a" through "f" above, and little or no correlation was found between the warp plane at any point in the process, and the plane of extrusion shift lead or lag, or the total heavy or light wall plane. The elements are now being autoradiographed and ultrasonically examined to determine variations in clad thickness. Results will be plotted to determine if bi-metallism can be considered a warp contributing factor in this case.

PRTR Flux Wire Fabrication. Two 24-pound turned and cropped zirconium alloy ingots have been double arc melted in vacuum for the purpose of producing approximately 2500 feet of 0.073" diameter wire. Nominal composition of the ingots is 0.15 percent Fe, 0.10 percent Cr, 1.50 percent Sn, 0.5 percent Co, balance zirconium.

The low oxygen value of 500 PPM in the starting sponge produced alloys of 156 and 159 BHN. This relatively low value should improve its workability over commercial Zircaloy-2 which runs in the 180-190 BHN range.

This material has been swaged from 0.493-inch rod down to 0.125-inch rod. The material swages well and is sufficiently ductile to take as much as a 60 percent reduction in cross section area before annealing is required.

The rod will now be drawn down to 75-mil wire on the draw bench. The Exeter lubricant will be used in this final operation.

Die Lubrication. Further tests have been made on the new lubricant using the Exeter Oil Ltd., base. So far, reductions of 10 and 20 percent per pass have been made without rupture of the lubricating film. These large reductions have been made on material containing as much as 40 percent cold work. The fastest drawing speeds used to date vary between 10 and 13 feet per minute. A 0.250-inch diameter titanium rod was swaged down to 0.129 inch in diameter and then drawn down to 0.050-inch diameter wire. This was also done using the Exeter asphalt as a lubricant and again no galling or film failure was noted.

Shaped Fuel Elements. Cylindrical fuel elements with corrugated exterior surfaces have been proposed as a fuel element design capable of undergoing volume expansions without localized thinning in the exterior cladding. Elastic analyses of the cladding, assuming the fuel behaves as a fluid, have been made to estimate stress conditions in the cladding. Since shaped fuel elements would be used to extend the irradiation life by allowing for sizeable volume increases, the cladding would deform plastically. If the corrugations are formed by tangential circular arcs, a criteria of equal bending strains in the arcs upon reaching the limiting circular geometry can be used to establish a ratio of arc radii. Calculations based on this criteria have been completed for six, eight, and ten corrugations for various depths of corrugations. Both the results of the elastic calculations and the calculations based on equal limiting bending strain have indicated that the radius of curvature of the valley should be several times that for the peak.

Design was started on an experimental N Reactor element in the form of a single tube with an outer surface shaped to accommodate uranium swelling without stressing the cladding in tension. The preliminary design is based on an element with eight flutes in the outer surface, with an expansion capability of between five and ten percent. The central bore will be cylindrical and a flow splitter will be employed to balance coolant flow. The element will be made from a Zircaloy-2 clad coextrusion and will have the same weight of uranium per foot as the present tube-in-tube element. Present indications are that this element will operate at a maximum power of 150 kw/ft in the ETR M-3 Loop.

ETR Shroud Tube Failure. The in-reactor section of the 3x3 P7 Loop has an aluminum shroud tube surrounding the pressure tube. This shroud, made of aluminum, failed below the reactor core during Cycle 44. The cause or exact location of the shroud tube failure

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has not yet been determined. The tube has been in continuous operation since August, 1958. A new tube will be installed in May, 1962.

Service Work. A test element for evaluating rupture detection equipment was developed for Instrument and Electrical Development, and a prototype was made for ex-reactor testing. The natural uranium fuel is in the form of unclad plates assembled inside of a 7-1/2 inch long aluminum dummy. The fuel element consists of a parallel array of 18 uranium plates, 0.010-inch thick, with a total of 200 square inches of uranium surface.

2. REACTOR PROGRAM

Corrosion and Coolant Systems Development

Hydriding of Zircaloy-2 at Low Temperatures. Continuation of the low temperature hydriding experiment reported last month has shown some pre-autoclaved Zircaloy-2 coupons to undergo accelerated hydriding in 20 days at 150 C in helium-1% hydrogen that had been carefully purified of oxidants (O_2 , H_2O) by passing the gas over hot calcium metal at 650 C. There is no other known case of hydriding at 150 C observed in a controlled experiment without prior vacuum annealing to remove the inhibiting ZrO_2 film from the metal. Hydriding in this experiment was due to the absolute purity of gases used. This confirms that the oxidant level of the gas is the primary factor controlling zirconium hydriding. Temperature, hydrogen pressure, metal surface roughness and prior ZrO_2 films all affect hydriding rates, but are not determining factors as to whether a sample will or will not hydride.

Electrical Resistivity of ZrO_2 Films. Experiments continued to elucidate the rates of resistance of ZrO_2 films in oxidizing and non-oxidizing atmospheres. After allowing the film resistance to drop below 100 ohms in vacuum at 450 C, the sample was cooled to room temperature and air admitted. The resistance of the film began to rise and reached a value of 160 K ohms in 8 hours. The system was evacuated without heating and the resistance dropped again to 300 ohms overnight. These results show the ZrO_2 film resistance to be sensitive to adsorbed oxygen. The sample was then oxidized at 450 C until the resistance was 600 K ohm. The resistance remained at 600 K ohm after the system was evacuated indicating heating the sample produced "permanent" changes in the properties of the film, in contrast to the superficial changes caused by adsorbed gases at room temperature.

Two capsules for measuring the electrical resistance of a zirconium oxide film in-reactor have been fabricated and assembled. Laboratory tests are being made prior to charging one capsule into KE Reactor. Tentative charging date is April 2, 1962.

Temperature Drop Through Corrosion Films. The thermocouple slug in KER-1, which was charged February 16, continues to function well. Analysis of temperature data indicates that this slug, in which only the central core temperature is measured, should be sensitive to film build-up resulting in temperature drops of 50 C or more. Smaller variations in core temperature cannot be distinguished from those due to power fluctuations. It is planned to decontaminate the loop with Turco 4512 at the next outage and study post-decontamination film deposition.

Analysis of Dissolved Hydrogen in KER Coolant. Five samples of KER-1 coolant were analyzed for dissolved hydrogen in the period 2/20 to 3/13. The hydrogen content decreased steadily from 8.2 cc/l at STP on 2/20 to 1.5 cc/l on 3/13. Since these data were obtained during the period just after the loop was charged, it is reasonable to assume that the initial high hydrogen concentrations were caused by the rapid initial corrosion rate of freshly charged carbon steel spacers. A detailed sampling schedule will be followed to obtain hydrogen concentration data during the period following startup of the KER-1 loop after it is decontaminated.

Corrosion Probe. A corrosion probe has been installed in TF-5 and hooked to a corrosometer. This probe will give a continuous measurement of the instantaneous corrosion rate of carbon steel in this loop. Similar probes will be placed in the KER Loops and in NPR.

Corrosion of Reactor Materials. Last month it was reported that as-received surfaces of a new group of 406 stainless steel samples exhibited intergranular attack after eight weeks of exposure to 300 C, pH 10 water. Metallographic examination of the surfaces indicated that the attack proceeded one or two grain lengths downward from the surface, then continued parallel to the surface. This indicates that a non-corrosion resistant surface layer exists on the as-received surfaces. The as-received surfaces did not have an adherent, glossy heat temper film as was found on previous samples. The glossy heat temper film was found to be corrosion resistant.

A corrosion test to determine the corrosion rates of A212 carbon steel, 304 stainless steel, and Zircaloy-2 in 290 C, neutral pH, deionized water is continuing. At 1200 hours exposure the rates are 0.26 mil/year for carbon steel and 0.094 mil/year for stainless steel. These rates are about twice as high as those obtained in pH 10 water under similar conditions. Zircaloy-2 corrosion is less than 0.02 mil.

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A test at pH 10 (NH_4OH) in 280 C, deionized water to compare the inhibition effectiveness of ammonium hydroxide with that obtained in other tests using lithium hydroxide is continuing. After 500 hours of exposure, A212 carbon steel and 304 stainless steel penetrations are 0.06 and 0.02 mil, respectively. This is comparable with penetrations obtained in LiOH inhibited systems.

High Temperature Corrosion Studies. The Breakaway Corrosion Loop (TF-15) has operated for about two weeks with coupons during the month. Operating conditions were 900 F, 4800 psi, 30 fps and neutral pH. Coupons were discharged after one week of operation. The 316 stainless steel, Hastelloy-X and D-979 had about 0.007 mil penetration, the 304 stainless steel about 0.021 mil penetration, and the Zircaloy-2 had a penetration of 0.36 mil. Some 406 stainless steel coupons had an apparent penetration of 0.04 mil, but later tests showed the 400 series stainless steel corrode excessively in the film stripping procedure. Some work is being done to determine better cleaning solutions for these coupons.

Decontaminants for Present Reactors. Both in- and ex-reactor tests have shown that decontaminants based on bisulfate (Wyandotte 1112 and 5061) or sulfamic acid (Turco 4306-C) are satisfactory from a corrosion standpoint if chlorides and fluorides are absent. The corrosion rates of aluminum and carbon steel are satisfactorily low; the corrosion rates of Zircaloy and stainless steel are almost too low to be measured. Galvanic corrosion of coupled samples is no problem. During the past month two new variations were tried: Turco 25B, a bisulfate, and Turco 27A, a sulfamic acid concoction. In the preliminary tests the uniform corrosion rates were low, but when aluminum was galvanically coupled to stainless steel, the aluminum pitted excessively. Additional tests are scheduled to obtain more quantitative data.

Two new compounds, Wyandotte 5150-0 and Turco HAPO-ADD, have been submitted for decontamination tests. These compounds contain high percentages of chlorides and fluorides and therefore should be very corrosive to aluminum and Zircaloy. In one in-reactor test it was found that three decontaminations in three months with a similar compound (Turco 4306 B) gave 12 mils corrosion of Zircaloy and approximately the same amount of corrosion of aluminum. These compounds should be useful for in-reactor decontamination in cases of emergency only.

Film Studies on Process Tubes. An aluminum tube was pushed from KE Reactor on 3/6/62. Removal was accomplished in the standard fashion: tube splitting followed by shearing into segments of unequal length. A section was removed from each recovered segment

with an underwater hand shearing device. Different portions of the tube showed different types of film. Sections near the reactor rear face were coated with heavy loose deposits of a reddish-brown material. The front face sections and the central zone pieces were practically free of this reddish-brown material.

K Reactor Nozzle Snap Rings. Samples of nozzle snap rings fabricated from Inconel are being exposed in 57-60 C filtered Columbia River water. Samples of 304 stainless steel and carbon steel are also being exposed as controls. After three weeks of exposure there is no visible corrosion on the Inconel or stainless steel. Small tubercles are forming on the carbon steel, indicating a pitting attack. Samples of Inconel and 304 stainless steel are also being tested to boiling 42% magnesium chloride. All samples are stressed to yield. After a one-week exposure the 304 stainless steel failed. No failure has been obtained on the Inconel after three weeks of exposure.

Control Rod Pitting. Recommendations for control of scattered, deep pitting of aluminum cooling channels in reactor control rods included five items: (1) maintain continuous cooling water flow at a rate 2 to 4 times the present rate; (2) clean flow channels, and treat with sodium dichromate; (3) add dichromate later, during unavoidable low flow periods, (4) examine other cooling channels wherever removed; and (5) consider Alclad aluminum tubing for fabrication of replacement cooling channels.

Structural Materials Development

Stress Rupture Tests on NPR Tubes. An extensometer is under development for remotely measuring the deformation of tube specimens undergoing creep tests. The ends of a section of roller chain wrapped around the tube separate as the tube expands. This motion may be sensed and transmitted out of the furnace by mechanical, optical, pneumatic, or electrical means or combinations of these methods. The method presently being explored uses a high temperature linear transducer. This device sends out an electrical signal that varies as its core is moved by the separating ends of the roller chain. In its initial trial it was applied to a tube specimen that had been under test for about 700 hours. A creep curve had been prepared up to that point from measurements made during periodic shutdown of the test. After about 70 hours in place on the specimen the extensometer indicated an increase of 0.001 inch in diameter. This point fell exactly on the predicted extension of the existing creep curve.

The three stress rupture tests of samples of 18 percent cold worked NPR process tubes at 300 C have now reached the stage summarized below:

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<u>Percent of Ultimate Pressure</u>	<u>Stress psi</u>	<u>Hours at Temperature</u>	<u>Total Creep Percent</u>
90	53 000	1000	0.28
93	55 000	970	0.71*
97	57 000	700	0.50

*The 93 percent test has terminated. At about 1150 hours a failure of the temperature control instrumentation allowed a temperature rise to 465 C over the space of about 30 minutes. At that point leakage developed at one of the seals allowing the pressure to drop and shutting down the unit. A bulge developed in the tube to a total deformation of about 9.5 percent. These results indicate very little drop in ultimate strength between 300 C and 465 C since the tube did not rupture.

Prototype Containment Vessel for Burst Tests of Irradiated Tubes. The prototype test unit has functioned well during five burst tests on sections of three-inch schedule 80 carbon steel pipe which burst at pressures of 13,000 to 14,000 psi at temperatures of 300-360 C. No damage to the furnace or vessel has occurred during these tests. The maximum pressure build-up in the vessel for the last two tests was measured by a pressure transducer to be 63 psi and 65 psi. The strain gage measurements on the outer surface of the vessel have been rather inconsistent on the last three tests. Instantaneous stresses equal to internal pressures of 70, 97, and 95 psi were observed.

The furnace required only 13 percent of the available power during a 16-hour test without any external cooling. With external water cooling, a situation simulating basin service, 15 percent of the available power was required.

Nonmetallic Materials

NPR Graphite Irradiations. The series of long-term irradiations of NPR graphite being conducted in the General Electric Test Reactor continues to progress satisfactorily. The first capsule in the series, H-4-1, was removed from the reactor on March 12, after three cycles of irradiation. This exposure, although subject to revision when flux monitors are measured, is estimated to be equivalent to approximately 14,300 MWD/AT in C Reactor or 5200 MWD/AT in the NPR. The external surfaces of the can were slightly darker than has been observed in the past. The capsule was disassembled in the hot cell and found to be in excellent internal condition. Some deposits of carbon were observed on the internal can surfaces adjacent to the two lower sample

positions. The quantity of carbon deposited was approximately equal to that seen in previous capsules. All 24 graphite samples and 65 flux monitors were successfully recovered and are being returned to HAPO for post-irradiation measurements. Three thermocouples were recovered for future calibration at Vallecitos.

The second capsule in the series, H-5-1, is now operating in the third reactor cycle. All nine thermocouples continue to function correctly. The capsule was removed from the reactor during the last shutdown for visual inspection. All wear surfaces were closely inspected, but no additional wear was observed. No evidence of fatigue failure was found. A small mark on the top plug which had appeared to be a hairline crack was found to be a very small aluminum chip. This was easily removed from the capsule. The prototype capsule was also inspected during the last shutdown. No evidence of additional wear or fatigue was observed.

The third capsule, H-6-1, was installed in the D7 reactor position on March 13 with a minimum of difficulty. All thermocouples are functioning properly. This capsule is of the same design as H-5-1 and is installed in the second beryllium filler piece. Difficulties encountered during capsule fabrication necessitated sealing of the outer thermocouple leadout tube. Lead was melted into the annular space between the two tubes to decrease the radiation beam reaching the plastic materials in the seal pot.

Graphite-Water Vapor Reaction. An apparatus has been constructed which will allow an investigation of the effect of the graphite sample size on the rate of reaction with water vapor or other controlled atmosphere. The quartz reaction chamber has an internal diameter of about four inches and a usable heated length of about 15 inches, thereby allowing the use of relatively large samples. Controlled temperatures up to 1000 C are readily attainable. The entire system can be evacuated, thus permitting the outgassing of the sample before admitting the oxidizing atmosphere.

Initial measurements will be performed on a series of cylindrical TSX graphite samples of various diameters. The water vapor partial pressure will be approximately 4.6 mm, so that measurable weight losses can be obtained in a reasonable length of time at temperatures of 700 to 1000 C. Reaction rates at each temperature will be obtained from the measured graphite weight loss and the time of reaction.

Radiation Damage to Plastics. Mass spectrometer analysis of gases evolved from irradiated Delrin (polyformaldehyde) gave approximately 44 percent CO₂, 9 percent CO, 23 percent CH₄, 11 percent H₂, 1.5 percent O₂, and the remainder various other fractions of the polymer

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molecule (masses of 60, 46, 32 and 30). The structure of Delrin is

Vacuum irradiation of oxygen-bearing polymers is being conducted to help determine the mechanism of radiation degradation.

Thermal Hydraulic Studies

Fuel Temperatures Resulting From a Flow Loss to a Process Tube on C Reactor. Experiments were completed in the laboratory to investigate heat transfer conditions and fuel temperatures which would result from the coolant flow loss following a failure of a hydraulic component at the inlet of a process tube at C Reactor. The purpose of these experiments was to define the power levels and other operating variables at which damage to the fuel elements would be expected following such an event.

The test section used was a full scale, electrically heated mockup of 32 C-II-N elements in a standard C Reactor process tube. A 23.6-foot long metal tube, 1.464 inches OD and 0.375 inch ID, represented the fuel elements and was made of materials with different electrical resistivities along the length to simulate a chopped cosine heat generation. During a typical experimental run, the test section was operated at normal conditions of power and flow and then subjected to a sudden loss of coolant by valving the coolant supply to a by-pass line and opening the inlet piping of the process tube to drain. Three seconds later the heat generation was decreased at a rate corresponding to a 1400 in-hour reactor shutdown. Recordings were made of pressures and temperatures throughout the test section, and each run was discontinued when the surface temperature of the test section either exceeded 1000 F or returned to normal temperatures when the flow from the rear header was sufficient to cool the test section. Rear header pressure was maintained constant during each run by means of a small centrifugal pump.

Results were obtained for sudden flow losses at various rear header pressures for initial tube powers of 1000, 1300, and 1500 KW with a 0.385" venturi on the front nozzle, and at 1000 and 1300 KW with a 0.302" orifice in place of the venturi. A preliminary analysis of the data indicated that a rear header pressure of approximately 58 psig would be necessary to assure sufficient cooling water from the

rear header to prevent melting of aluminum jackets on tubes equipped with venturis and originally at 1000 KW. Rear header pressures approximately 13 psi higher would be required for those tubes equipped with orifices. The results at higher power levels were still being analyzed at the time of this report.

Visual Studies of Boiling on Hanford Type Fuel Elements. Studies were continued to study the heat transfer conditions when fuel elements are not situated in a coaxial position within a process tube. The test section in this study consisted of a 13.6-inch long electrically heated rod, 1.304 inches in diameter, placed within a 1.504-inch ID glass tube. In selected runs metal strips were fastened to the rod to simulate the support pieces used on "self-supported" fuel elements.

The first run was made without any metal support pieces but with the heated rod arranged eccentrically within the glass tube with a minimum annular clearance of 0.030 inch. With a flow rate of 5 gpm and inlet water 90 F subcooled, boiling was initiated on the heated surface along the narrowest part of the annulus. At a heat flux of 285,000 B/hr-sq ft film boiling conditions were encountered. This compares to a heat flux of 232,000 B/hr-sq ft at similar conditions but where the surface had metal support pieces. When the flow was increased to 20 gpm and the inlet subcooling to 150 F, no boiling was detected at 700,000 B/hr-sq ft with no support pieces on the heated surface. However, vigorous boiling took place under similar flow conditions with support pieces and film boiling conditions were encountered at a heat flux of 555,000 B/hr-sq ft. It appeared that bubbles were initiated in the stagnant water under the support pieces and failed to condense when they moved out into the flow stream. Plans were made to continue the experiments since it appears that high fuel surface temperatures could result if the support pieces on fuel elements were crushed and the fuel pieces allowed to become eccentric within the process tubes.

Vibration of NPR Fuel Pieces. A visual study was made pertaining to the vibrations of the inner piece of an NPR fuel element within the outer piece. This was done in a special hydraulic test section consisting of an inner fuel piece placed in a plastic tube which had an inside diameter approximately the same as the outer piece of the NPR fuel element. Additional inner fuel pieces were placed upstream and downstream of the test piece to provide uninterrupted flow past the test piece. Part of the tests were carried out in the presence of strong mechanical vibration from a misaligned pump, part with a minimum of mechanical vibration from the pump, and part where

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the test section was isolated from the flow piping with rubber gaskets. Separate tests were made using fuel pieces supported within the plastic tube with "buggy spring" and "suitcase handle" supports.

It was found that when the "buggy springs" were used for support, there was about a 0.010 inch relative movement between the fuel piece and the plastic tube at a flow of 68 gpm and with strong pipe vibrations originating from the pump. This movement was all absorbed within the buggy springs. There was no vibration of the fuel piece with "suitcase handles" under these conditions. When the pump was aligned so that the mechanical vibration was at a minimum, the fuel piece did not vibrate with either type of support. Also, no vibration of the fuel piece was noted when the test section was isolated with rubber gaskets and the flow was 68 gpm. However, when the flow rate was increased to 100 gpm, the fuel piece supported by the buggy springs vibrated slightly at the same frequency as the motor speed. High speed motion pictures were taken of all tests and further analysis of the results were planned following developing of the film.

Flow Mixing Studies. A special test section was built to determine the mixing efficiency of the devices used in the present production reactor to mix the annular and hole coolant streams created by I&E fuel elements. The test section allows water at different temperatures to be introduced into the two flow channels and provides for temperature measurements to be made throughout the flow channels downstream of the mixer.

Data obtained on the mixers presently in use indicated that the efficiency of interchange between the hole and annulus was about 80%. The change in temperature from upstream to downstream between the ribs was about what would be calculated theoretically assuming the hole flow were distributed evenly to the annulus. However, very large temperature variations were observed around the annulus.

Data were also obtained on a new helical mixer under development. These data indicated a much more even temperature distribution around the annulus, less efficiency in exchanging the annular and hole flows, and an indication that considerable circumferential mixing takes place between the annulus outside of the ribs and the annulus inside of the ribs. This latter mixing would be desirable for some cases in the present tubes where the water temperatures are high in the annulus between the ribs.

The next step in the study will be to repeat the tests using a process tube in which the ribs have been ground down 0.030 inch to represent a tube in a worn condition.

Hydraulic Studies. During the past year seven reactor shutdowns have been caused by Panellit oscillations on tubes with K Reactor Zone 12 orifices. When tested in the hydraulics laboratory, Panellit pressure oscillations of ± 15 psi were observed for flow through this orifice. It was found, however, that the oscillations could be reduced about 50% by extending the length of the orifice beyond the Panellit pressure tap hole.

Miscellaneous. Calculations were performed to determine the hazards involved in shipping a lot of 508 I&E slugs eight hours after shutdown from DR Reactor to C Area. These elements were irradiated 1200 MWD/T and were in the reactor 130 operating days. The criterion for permissible shipment procedure was that the coolant in the well cars should not boil before delivery. It was found that this could be satisfied on a 50 F day if the shipment was divided equally between three wells (approximately 169 elements per well) and the shipment was completed within five hours.

B. WEAPONS - 3000 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.

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C. REACTOR DEVELOPMENT - 4000 PROGRAM

1. PLUTONIUM RECYCLE PROGRAM

Thermal Hydraulic Studies

Consequences of Failure of PRTR Rupture Loop Pressure Control Valve. The PRTR Rupture Test Facility operates with a once through coolant flow during certain fuel element rupture tests. The flow is from a storage tank through a positive displacement pump, the test section, a pressure control valve where the pressure is reduced to near atmospheric and then either directly into a storage tank, or through ion exchange units into the storage tank. The loop is operated at about 2000 psi and 600 F. A study was made to assess the possibility of rupturing the ion exchange units or the storage tank if the pressure reducing valve failed to the full open position and exposed the outlet piping to the full loop pressure of about 2000 psi. The calculation of transient pressures throughout the loop following failure of the valve is complex, and a satisfactory calculational procedure has not yet been developed. However, a preliminary semi-qualitative calculation makes it appear likely that bursting pressures would be reached in both the storage tank and ion exchange unit if the PRV were to fail open. Plans were made to continue on a more exact calculation of the situation.

Heating of Exposed PRTR Shim Rods. Currently, the PRTR shim rods stick occasionally and also suffer from failure of the instruments indicating rod position. This raises concern about the cooling of a shim rod which may be unknowingly stuck in its uppermost position when the reactor is operated at reduced moderator levels. A fully extended shim rod is at 98 inches height whereas the reactor may be brought critical at 72 inches moderator level and may operate at 89 inches moderator level up to 7 MW. In these cases the shim rod above the moderator level will not be water cooled. Calculations were made which show that up to eight inches of the shim rod may be exposed at full power of 70 MW without causing melting of the aluminum components of the shim rod hardware. The calculations also show that no melting should occur no matter how much of the shim rod is exposed if the power level were no greater than about 25 MW.

Component Testing and Equipment Development.

PRTR Components:

1. Liquid Poison Shim Rods

As reported last month, out-of-reactor laboratory tests have shown cadmium sulfate with additives to be a suitable poison

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material for liquid poison shim rods. However, in-reactor testing of the cadmium sulfate in contact with aluminum is required to fully evaluate its applicability. To duplicate the exact environment of future shim rod use, irradiation testing in the PRTR has been proposed to Test Reactor and Auxiliaries Operation.

2. Mechanical Shim Rods

Design work continued on the second generation mechanical shim rod. Present PRTR shim rods have experienced problems in rod limit and position indicators. Since it has not been possible to determine whether these problems are of in-reactor or out-of-reactor origin, the design is proceeding on the basis that faults exist in the components internal to the reactor. The second generation mechanical shim rod design is being modified to use rotary switches in place of the present limit-type micro switches and servo-transmitters in place of the position transmitting potentiometers.

3. Environment Control Test Facility

The adequacy of Lucite for use in the shim rod environment control test facility was determined. A 6" OD x 1/4" wall tube with a nominal 2" branch cemented to the outside wall was hydrostatically tested at 175 F. Failure occurred in the 2" side outlet pipe at 52 psig. The failure is considered peculiar to the particular specimen tested because it had been fabricated from laminated sheets, and the failure occurred at the sheet bondings, presumably from the cement being leached by the hot water.

The hydrostatic test results suggested further examination of the bonding characteristics of the cemented joint between the 6" and the 2" tubing. The specimen was immersed in 175 F water for 162 hours and in 135 F water for 16 hours. No undesirable characteristics were observed from these tests. As a result, the use of Lucite tubing in the test facility is considered acceptable, and design is proceeding on this basis.

4. Reactor Auxiliaries Test Facility

A scope flow diagram for the reactor auxiliaries mockup was completed. Preliminary design from which detail design and construction cost estimates can be made has been undertaken by CE&UO.

The reactor auxiliaries mockup is a facility designed for testing gas system components. To provide flexibility, the facility will be designed with two basic piping configurations. In one, equipment such as compressors may be tested up to 2500 psig. In the other, blowers and components may be tested dynamically up to 1300 F at about 500 psig.

5. Shroud Tube Replacement Mockup

A mockup of a section of the PRTR top and bottom primary shields and calandria is needed to test equipment and procedures for replacing calandria shroud tubes. A detailed design was prepared of the mockup assembly based on erection in the 309 M&M Building mockup pit; however, greater convenience and economy can result from installing it in the 314 Building. CE&UO has undertaken preliminary design of the mockup for installation in a pit in the 314 Building.

Critical Facility Components:

1. Safety Rods

Assembly and testing of the remaining two safety rods is complete. The cadmium plating on the control elements for these two rods is badly blistered. New aluminum tubing sufficient for six new control elements is being machined prior to plating. The blistered control elements will be used for design testing only and will be replaced prior to loading fuel in the Critical Facility.

Rupture Loop:

1. In-Reactor Test Section

Since various portions of Rupture Loop Design Test No. 4 will be performed in the EDEL-1 flow loop, it has been necessary to alter the loop for operation at rupture loop operating conditions. Checking and rewiring of electrical and instrument safety circuits disclosed that all pneumatic instruments were laden with oil and many of the thermocouple leads were shorted or grounded. These conditions were corrected by installing an oil filter and by replacing damaged thermocouple leads. Replacement of the pump seal heat exchanger piping is required because it does not meet the code requirements for rupture loop operating conditions.

Although EDEL-I was designed, fabricated, and tested for operating conditions of 2400 psig and 650 F (sufficient for rupture loop conditions), its operating life to date has been at PRTR conditions (1250 psi and 530 F). Consequently, another hydrostatic test at 3750 psig will be performed upon completion of the pump seal heat exchanger piping.

Preliminary operation of the facility, at 500 psig, to check instrumentation, pump seals, safety circuits, etc., has been completed satisfactorily.

2. Discharge Equipment

End Closures and Cooling Components. Design of the special process tube end closures and hose arrangements is essentially complete. Fabrication was started during the week of March 19. Requisitions for off-plant supplied items have been issued.

Fuel Handling Modifications. The necessary modifications to the fuel handler to accommodate the tube assembly with end fittings and cooling lines attached have been defined. These include (1) raising the cask portion of the unit by approximately one inch to provide clearance for hoses under the cask (this will be accomplished by shimming under the cross-tracks), (2) slotting the shielding skirt for the hoses, and (3) adjusting the limit switches controlling the lifting hook assembly to accommodate the tube-nozzle assembly.

Fuel Transfer System. A mockup of the rupture loop tube assembly, with end fittings, was handled through the PRTR fuel element transfer system in a test conducted on March 5. Minor modifications of the transfer system will be required to accommodate the assembly.

Special Tools. The present criteria require that all work on the outlet end of the process tube be accomplished semi-remotely (no contact work). A special shielding cart has been designed through which tooling operations and component manipulation will be conducted. Several hand and/or power tools will be required for the tube discharge procedure. Over-all tool design is approximately 15 percent complete.

Hazards Analysis

Process Specifications. Two revisions of PRTR process specifications

were approved during the month. Preparation of Critical Facility process specifications was continued.

PRTR Fuel Element Rupture Loop. A comment draft of the Fuel Element Rupture Loop Final Safeguards Analysis was issued. The analyses conducted lead to the conclusion that the presence of the rupture loop in the PRTR will not materially increase the hazard of operating the reactor.

Reactor Safeguards Reviews. Operational safety reviews of the PRTR and the Plutonium Recycle Critical Facility were conducted by the General Electric Technological Hazards Council on the tentative experimental program for the PRTR, the technical aspects of the PRTR freon incident, and the methods which will be used for limiting the excess reactivity in the Critical Facility.

Reactivity of High Exposure Pu-Al Fuel Elements. Irradiation of plutonium-aluminum fuel which exceeds the reactivity value of that provided by 330 grams of Pu-239 in the cold-clean condition has been interpreted as outside the safeguards limitations on the PRTR. The exposure variation of the reactivity value of Pu-Al fuel containing various amounts of Pu-240 was investigated for plutonium containing 5.3, 16.5, and 29.6 percent Pu-240. No increase in reactivity with increasing exposure was observed for these cases under the assumed PRTR neutron spectra. Details of this investigation are presented in HW-73107.

PRTR Startup History

The first drafts of individual contributions for the PRTR startup history document were completed.

Plutonium Fuels Development

UO₂-PuO₂ Fuel Fabrication for PRTR. The fabrication, by cold swaging, of UO₂-PuO₂ fuel rods for PRTR use was begun during this period. An initial group of one hundred and twenty fuel rods were loaded and will be through the fabrication process this week. The fuel rods were incrementally loaded with arc-fused UO₂ (-20 mesh - 325 mesh) and a blend of UO₂-PuO₂ (-325 mesh); the loaded fuel core containing 0.48 w/o PuO₂ (15.8% Pu-240). After loading, the rods were decontaminated, and one-fourth inch of UO₂ (-60 mesh + 100 mesh) was placed on top of the fuel to cover the fine PuO₂ containing material. No welds have become contaminated during the welding of the second end cap since the adoption of this technique. Following welding, the rods were swaged and grouped by length prior to final process steps and assembling.

Rods for four clusters have been selected. Three of the clusters will have a length variation of $\pm 1/16$ inch; the remaining cluster will vary $\pm 1/8$ inch. Over-all length difference between all the rods for the four clusters is less than 0.75 inch.

From the information obtained during the fabrication of this initial group of rods, several process modifications are now being made which will permit an increased fabrication rate and reduce over-all costs.

A prototypical incremental feeder for metering $\text{UO}_2\text{-PuO}_2$ into zirconium tubing was installed in the vibrational compactor glove box. The powder and shot measure feeder was found satisfactory for the three UO_2 size fractions required for vibrational compaction after it was determined that the feeder used for swage compaction fuel elements would not handle $-6 + 10$ mesh UO_2 . Fabrication of PuO_2 -bearing fuel elements with existing equipment is starting and a gravimetric feeder is being procured which will be used for either uniform or cosine-enrichment loading.

Extended Surface Plutonium Fuels. Evaluation of the zirconium clad Pu - 15 w/o Zr fuel plates continued. One plate was autoclaved 370 C, 68 hours; no unusual corrosion was detected and no preferential attack in the bond area was visible. A section through the core from another plate will be tested in the autoclave to determine the corrosion resistance and undercutting of the core. Metallographic examination showed the core material to vary between 0.003 inch and 0.004 inch with areas of thickening to 0.012 inch at intermetallic sites. The cladding was correspondingly thinner at the intermetallic sites. The core material before roll-cladding was very soft and ductile at room temperature and had a melting point above 1000 C. The delta phase of plutonium was apparently retained at room temperature; no large second phase particles were detected metallographically. As noted last month, the intermetallics found in the core after rolling are thought to form from a reaction between the aluminum foil wrapping (used for contamination control) and the Pu-Zr alloy. Elimination of the aluminum foil wrapping will be attempted. A suitable fuel design for testing in the PRTR is being developed.

Phoenix Experiment. The irradiation and reactivity measurements on the high-exposure aluminum-plutonium samples are continuing in the MTR and the ARMF. At the present time, the sample with plutonium which initially contained 6.25 w/o Pu-240 (GEH-21-1) is being irradiated for its fourth cycle; the sample containing plutonium with 16.33 w/o Pu-240 (GEH-21-3) has been irradiated for its third cycle and is being cooled prior to the reactivity measurements, and the reactivity measurements have been made on sample containing plutonium with 27.17 w/o Pu-240 (GEH-21-19)

which will now be irradiated for its second cycle in the MTR.

Al-Pu-B poison standard samples have been made for calibration of the ARMF. These samples have been clad in Zircaloy and reactivity measurements will be made in the PRTR at Hanford to determine the boron uniformity and concentration. The compositions of the PCTR calibration samples are pure aluminum, Al - 3.17w/o Pu (16.33 w/o Pu-240), Al - 3.17 w/o Pu - 0.03 w/o B, and Al - 3.17 w/o Pu - 0.06 w/o B.

Irradiation Testing. The 42-inch long UO₂-PuO₂ cosine enriched seven-rod cluster element (GEH-11-8) has successfully completed its 27.3-day operating period in the ETR 3x3 high temperature loop. It was fabricated by swage compacting UO₂-PuO₂ fuel to 89 percent of theoretical (arc-fused UO₂ and -325 mesh PuO₂) using swageable type end caps. The element operated for 16 days of this period with a coolant outlet temperature of 510 F and due to loop operating difficulties, 12 days with a 150 F coolant temperature. A maximum total exposure of about 1100 MWD/T and an average of 850 MWD/T was accumulated during this irradiation period. Vertical peak/average power flattening was achieved by varying the PuO₂ concentration along the length of the element to compensate for the normal longitudinal neutron flux variations across the ETR core. As a result, this element generated 30 percent more power than a uniformly enriched element at the same maximum heat flux (345,000 Btu/hr-sq ft) and same maximum central core temperature (1850 C). The element will now be returned to Hanford for examination.

Radiometallurgical examination of the 42-inch long swage compacted UO₂-PuO₂ seven-rod cluster (GEH-11-7) has commenced. This element was fabricated by swage compacting arc-fused UO₂ enriched with -325 mesh PuO₂ using swageable end caps. The use of swageable type end caps does not allow any end clearance between the end cap and the fuel material. The element operated for 23 full power days (800 MWD/T maximum) at a maximum surface heat flux of 345,000 Btu/hr-sq ft and a maximum core temperature of 1850 C. PuO₂ enrichment was uniformly distributed along the length of the fuel rods by using the incremental loading technique. The external appearance of the element was unchanged as a result of irradiation and there were no signs of bowing, warping, or loose spacing wires. Autoradiographs of the individual rods were made by exposing them to glass and the center sections of the rods were generally darker than the ends which is expected as a result of the normal flux pattern in the ETR. Areas of increased exposure on the glass were observed indicating increased localized fission product concentration resulting from plutonium segregation. The number of overexposed areas corresponds with the number of UO₂-PuO₂ increments loaded into each fuel rod. This indicates there is appreciable PuO₂

segregation within each small fuel increment even though the plutonium is "uniformly" distributed over longer fuel lengths. This observation demonstrates that the very small PuO_2 particles are not redistributed significantly as a result of the swaging operation. It also demonstrates that the PuO_2 does not diffuse or relocate longitudinally when operating under these conditions for this period of time. Efforts will be made to improve PuO_2 distribution since gross segregation could cause localized hotspots and consequent fuel burnout. The transverse sections of the fuel rods will be examined for evidence of radial plutonium relocation.

The eight MgO-PuO_2 and the eight $\text{ZrO}_2\text{-PuO}_2$ fuel capsules have successfully completed their first cycle of irradiation in the ETR. Four of the MgO-PuO_2 capsules have received the desired exposure and have been discharged from the reactor.

The irradiated Zircaloy-clad capsule (GEH-14-27) containing an Al - 2.1 w/o Pu - 2.0 w/o Ni alloy core and fabricated by injection casting was shipped from the MTR to HAP0 this month. The specimen received an exposure of 82.5×10^{18} fissions/cu cm and a burnup of plutonium atoms of 60 percent.

The examination of the eleven discharged high and low density $\text{UO}_2\text{-PuO}_2$ capsules is continuing. Two of the capsules (GEH-14-85 and 86) are being returned to the MTR for irradiation to higher exposure. The test proposal, HW-72892-RD, was completed and issued. The present average exposure of the capsules is about 160×10^{18} fissions/cu cm or 5500 MWD/T of $\text{UO}_2\text{-PuO}_2$. Hardness measurements were made on the UO_2 - 1.02 a/o PuO_2 core of GEH-14-66, the capsule with the low density oxide which formed a central void and a metallic sphere (0.1 inch diameter). The hardness data are shown below:

<u>Region</u>	<u>Average Hardness, KHN</u> <u>(500 gm load)</u>
Zr-2 cap ($\sim 3/32$ inch in from edge)	225
Zr-2 cap (core-cap interaction zone)	198
$\text{UO}_2\text{-PuO}_2$ (central region)	723
Metallic sphere	543

Conclusive results have not been obtained on the sphere to date by means of x-ray diffraction. Present plans are to drill out samples from the sphere for analysis and to etch the sphere in an attempt to view the microstructure.

On the remaining specimens of the group, the maximum calculated exposure is 7700 MWD/T of $\text{UO}_2\text{-PuO}_2$; however, burnup analyses indicate up to 10,800 MWD/T of $\text{UO}_2\text{-PuO}_2$. Examination of the capsules is continuing.

Uranium Fuels Development

PRTR Fuel Elements. The vibrationally compacted, the hot swaged, and the maximum exposure (1700 MWD/T_U) cold swaged UO_2 , 19-rod cluster PRTR fuel elements were visually examined in the PRTR storage basin during a reactor outage. None of the elements exhibited signs of corrosion or mechanical damage and, following the examination, were recharged into the reactor.

Evaluation of AISI-406 Stainless Steel Cladding. An irradiation test of AISI-406 stainless steel clad fuel rods at the MTR (GEH-4-71) was terminated after ~ 20 hours of full power operation because of high activity in the coolant. The fuel rods operated with an average surface heat flux of 400,000 Btu/hr-ft² (determined calorimetrically) and a maximum of 520,000 Btu/hr-ft². The four-rod cluster comprised two rods clad in 0.009 inch wall x 0.563 inch OD 406 stainless steel and two rods clad in 0.015 inch wall x 0.563 inch OD 304-L stainless steel. The fuel was sintered and crushed UO_2 , vibrationally compacted to a bulk density of ~86 percent TD (compaction efficiency of ~90 percent).

Vibrational Compaction Studies. Fuel rods were vibrationally compacted for six Mark-I elements. These elements will be wrapped with cobalt-containing zirconium wire for a physics test to be conducted in the PRTR. The UO_2 fuel densities obtained varied from 87 to 88.5 percent of theoretical; these correspond to compaction efficiencies of approximately 89 to 90.6 percent. The rods were compacted by vibrational energy transmitted through a steel beam. The efficiency of compaction along the beam varied, both with respect to position and to the total number of rods being processed. An unusually high number of the vibrationally compacted fuel rods have returned from the autoclave with a white or "milky" surface. These rods were buffed and re-autoclaved, after which most of them passed inspection. The cause of the non-standard oxide film has not been established; it may be related to the zirconium ingot quality.

Materials Development

Automatic Measurement of Chloride and Phosphate in Coolant Streams. Laboratory evaluation of the procedures for chloride and phosphate analyses were completed. Analysis of the results demonstrate that

these procedures are extremely accurate and sensitive. The chloride procedure is accurate to ± 5 ppb in the 0-100 ppb range. The procedure used involves the reaction between slightly dissociated mercuric thiocyanate and chloride ion in the presence of acidified ferric nitrate solution to form mercuric chloride and ferric thiocyanate. The chloride concentration is determined indirectly by measuring the ferric thiocyanate complex concentration.

The phosphate procedure is accurate to ± 5 ppb in the 0-200 ppb range. The procedure used involves the reaction between acidified ammonium molybdate solution and phosphate ion to form phosphomolybdic acid, followed by reduction with stannous chloride to molybdenum blue. The phosphate concentration is determined indirectly by measuring the concentration of the molybdenum blue complex formed.

Hydriding of Zircaloy-2. A study of the amount of corrosion product hydrogen picked up by Zircaloy-2 during the initial oxidation stages in static 400 C steam has been completed. Five-mil thick Zircaloy-2 samples were exposed in an induction heated autoclave for periods of 15 minutes to one hour. Analysis of the samples has shown that the percentage of corrosion product hydrogen pickup during these short exposure periods is constant with time and independent of oxide thickness. The amount of hydrogen pickup was 24 percent which compares closely with the 28 percent previously found for Zircaloy-2 exposed to static 400 C steam for periods up to 28 days.

Heat Treatment of Oxide Films on Zircaloy-2. Autoclaved Zircaloy-2 samples have been vacuum heat treated to investigate effects on subsequent weight gains and breakaway times. Samples autoclaved in steam at 400 C for 28 and 38 days were vacuum heat treated at 510 and 600 C. Heat treating times were 5, 20, 60, and 120 minutes at 510 C and 5, 10, 20, and 60 minutes at 600 C. The amount of oxide film dissolved during heat treating was 0.3 to 2.1% of the total film at 510 C and 2.2 to 7.7% at 600 C, depending upon the time at temperature. Heat treatments at 510 C did not alter the normal breakaway time for Zircaloy-2 of approximately 42 to 45 days. Heat treatments at 600 C for 60 minutes after autoclaving 28 days produced an immediate increase in oxidation rate; weight gains increased from 35 mg/dm² to 60 mg/dm² in five days additional autoclaving. Furthermore, a second breakaway occurred at 50 days (total autoclaving time) at a total weight gain of about 70 mg/dm². The data of Kass on heat treating of Zircaloy-2 at 590 C and subsequent autoclaving appear to show accelerated multiple transitions. Heat treatments for 10 and 20 minutes at 600 C after 38 days prior autoclaving resulted in slight increases in oxidation rate and appeared also to have extended the normal breakaway time slightly.

In-Reactor Monitoring. Seven tubes were inspected during the month. No general changes in inside diameter, within instrument accuracies (± 0.002 inch) were found. The gas gaps in channels 1348 and 1857 were found to be about 35 mils compared with a nominal 270 mils. In cooperation with PRTR Operations and Maintenance, the effects from piping jumpers, nozzle hold-downs, and radial forces on the nozzle were measured on channel 1348. Although application of radial forces on the nozzle gave increased gaps to 100 mils, the gap returned to about 35 to 40 mils when the forces were released. Means to increase the gap in channel 1348 are being investigated by PRTR Operations. The process tube in channel 1857 was rotated 180 degrees which increased the gap to nearly 100 mils.

Wear corrosion marks at the contact points with fuel element end bracket measured in five of the tubes inspected varied from 0 to 6 mils deep. The tube in channel 1857 which was installed new in October 1961, had grooves up to 2 mils deep. Contact with outer spiral wire wrap has produced grooves up to 4 mils deep.

The mock-up facility in 314 Building for testing modified monitoring probes is complete. Preliminary traverses using a shortened instrument tube and the TV-Omniscopes and ID gage have been made with no major problems found. Preliminary check-outs of the nozzle extension insert have indicated satisfactory operation. Lack of suppliers for the material to make the body of the Mark III gas gap instrument has caused a month's delay in instrument fabrication. A supplier has now been found and shipment is scheduled by the end of this month. A granular quartz-filled silicone resin molding has been selected for the instrument body based on machining characteristics and no damage in irradiation tests to 10^{10} roentgens accumulated gamma dosage.

2. PLUTONIUM UTILIZATION STUDIES

Plutonium Dioxide

Prior to obtaining the thermal expansion characteristics of PuO_2 at high temperatures (hopefully, 2000 C), runs were made at 200, 450, 600, and 825 C on the high temperature diffractometer attachment. The specimen was mounted as a film on a Mo ribbon which was heated by resistance. The lattice expansion data obtained agreed very well with dilatometer data and runs to higher temperatures are planned.

Relative intensities of the strong $\alpha\text{-Pu}_2\text{O}_3$ and PuO_2 lines have been correlated with O/Pu ratio and metallographic observations have shown the microstructural changes as a function of composition at room temperature. A tentative phase diagram has been constructed

for the Pu-O system based on the dilatometric, resistivity, and quench experiments to date. Several confirming experiments are currently in progress.

Annealing of $\text{PuO}_{1.62}$ in dry hydrogen at 1450 C was performed in effort to transform the high temperature fluorite phase into the C type rare earth form. The phases resulting from the anneal were PuO_2 with the normal $a_0 = 5.396\text{\AA}$ and about 20 percent PuO .

An attempt to form beta- Pu_2O_3 by reacting Pu and PuO_2 at 1500 C in vacuum gave equal quantities of alpha- Pu_2O_3 and PuO_2 . This experiment will continue at higher temperatures.

PuO_2 - UO_2 Sintering

In the production of UO_2 -1.5 w/o PuO_2 pellets for the Euratom loading some spots of high activity were found by both gamma spectrometry and alpha autoradiography. These spots were observed on two pellets after successive sinterings of 4, 24, and 90 hours at 1550 C. Surprisingly, the spots showed little change in the course of the sintering treatment. They appeared slightly more diffuse after the longer sintering times on alpha autoradiography but were otherwise unchanged.

A mechanism to explain this unusual behavior was postulated. The green pellets are known to have small areas rich in PuO_2 , as would be expected in the case of any mixture of two powders. From other sintering studies it has been determined that the sinterability of UO_2 is decreased with small additions of PuO_2 while it is enhanced by large additions. If one assumes that a small agglomerate contains 80 percent PuO_2 - 20 percent UO_2 while the surrounding area is 1.5 percent PuO_2 - 98.5 percent UO_2 , it can be calculated that the isolated area will have a five percent greater volume shrinkage than the matrix in the first four-hour sintering run. This might logically cause severe cracking in the region of the spots. These cracks would act as diffusion barriers which would inhibit diffusion in this region.

In an effort to test this hypothesis the two pellets which had accumulated 90 hours of sintering time were mounted in plastic. The longitudinal flat which had been previously ground was given a light metallographic polish and etch. A few small areas (10-20 mils diameter) were visible which were obviously not attacked by the etchant. When these areas were examined under the microscope, they invariably revealed a network of microcracks. The structure showed fewer voids than the surrounding matrix indicative of a higher density. Under the microscope it was also apparent that these areas were truly more resistant to attack by the chemical etchant. To obtain further

corroboration of the hypothesis the polished surface will be alpha autoradiographed again in an effort to match the microscopic results with the signs of high alpha activity.

Plutonium Carbide

The Pu-C alloys reported last month, which showed only a single PuC phase by x-ray diffraction below 40 a/o C, were examined metallographically. A small amount of an intergranular phase was seen which is believed to be alpha plutonium. Above 40 a/o C, the alloys appeared single phase; however, the quite rapid oxidation of these samples makes detection of small quantities of a second phase difficult. No conclusive evidence of zeta was found. Above 48 a/o C, Pu₂C₃ was observed.

Another anneal has been performed on a set of Pu-C alloys extending from 30-52 a/o C. These samples were held for 65 hours at 610 C in vacuum and cooled to room temperature over an eight-hour period. The first few diffraction patterns have been run on this material, and again only a PuC phase was found in the range 30-48 a/o C. The lattice constant below 40 a/o C appears to be on the order 4.972A, while above this composition lattice constant values as high as 4.980A were again seen.

Hardness as a function of carbon content was obtained on arc melted and annealed alloys. Below 40 a/o C, hardness values increased slowly with carbon and reached 360 DPHN at 39.7 a/o C. Above 40 a/o C, there was a large jump to about 860 DPHN. This jump in hardness at 40 a/o C, the low melting point in this region, the large change in lattice parameters, and the occasional detection of zeta all point toward some sort of crystallographic change at this composition. An interesting possibility is one in which a change in structure type occurs at about 40 a/o C due to repositioning of the nonmetal atoms. This could account for low hardness by weaker metal-nonmetal bonding and could explain the lattice parameter change and might account for zeta, since this phase could be a transition between structures.

A possibility for the structure change is one in which the metal atoms retain the face centering positions, but the carbon moves from the octrahedral holes in the NaCl type lattice (PuC) to the tetrahedral positions. This gives a zinc blende type structure deficient in carbon. Though the bond number would decrease with this type of shift, the change in bond direction and the subtraction type defect lattice could still explain the low hardness and melting point below 40 a/o C. This would also explain why we haven't seen zeta after long anneal times, i.e., the transition would be completed.

To test this hypothesis, intensities of reflections from carbides containing 31.9 and 51.5 a/o C were compared. Differences were noted between the relative intensities of the (111)/(200) and the (422)/(511) planes on the two samples. Structure factor calculations are planned to determine whether these intensity changes are related to a transition in structure type.

Additional PuC rich PuC-UC alloys were arc melted and then quenched from 1750 °C in vacuum. Visual observation on the as-quenched samples and x-ray data confirms that 25 a/o UC stabilizes the PuC peritectic decomposition. Metallographic examination will be used to check this stabilization limit.

The reaction of PuC with hydrogen has been studied in more detail. In several earlier tests carbide buttons had shown attack when exposed to hydrogen both at elevated (1400 °C) and low (40 °C) temperatures. The hydrogen used in these experiments contained 100 ppm water vapor. It was uncertain whether the attack had been caused by the hydrogen or by the associated water vapor.

Four arc melted samples of varying carbon content were exposed to extra high purity hydrogen (10 ppm total impurities) which had been passed through a liquid nitrogen cold trap. This reduced the water vapor to 10^{-18} ppm or less. The results of these tests show rather conclusively the nature and cause of the attack. The two low carbon samples, 40.5 a/o C and 14.8 a/o C, were reduced to powder overnight. The higher carbon samples, 46.0 a/o C and 47.8 a/o C, both showed some resistance to attack. The 47.8 a/o C sample started to crumble after 72 hours while the 46.0 a/o piece took one week before it showed serious attack.

The appearance of the pieces after attack suggested that samples of PuC had been attacked at the grain boundaries leaving bright, large grains of PuC unattacked. Photomicrographs of the samples showed a second phase, presumed to be Pu, at the grain boundaries. In the case of the low carbon samples a continuous network had been formed. The higher carbon samples showed the same phase but to a much lesser extent. From thermodynamic considerations it was much more logical that a reaction was taking place between the hydrogen and the metal than between the hydrogen and the carbide. This combined with the 96 percent volume increase going from the metal to the hydride offers a very reasonable explanation for the rather unexpected attack.

Plutonium Nitride

The plutonium nitride presently being studied was synthesized by arc melting plutonium under one atmosphere of nitrogen. This single-phase, face-centered cubic material has a unit-cell dimension of $4.905\text{\AA} \pm 0.002$.

Melting studies were made in gettered helium. The tungsten ribbon upon which the plutonium nitride was placed was resistance heated to 2650 C. Melting was not evident, although a rapid diminution of the specimen occurred by volatilization. The plutonium nitride did not appear to react with the tungsten ribbon. The vapor was vented out of the furnace and condensed on a glass plate in a nitrogen atmosphere. X-ray diffraction results show that the vapor consisted of plutonium nitride plus a trace of plutonium dioxide. It may be hypothesized that the plutonium nitride decomposed on melting. The plutonium vapor then recombined with the nitrogen in the hood atmosphere. The trace amount of plutonium dioxide present is reasonable since the vapor consisted of very fine particles and the nitrogen atmosphere did contain a small percentage of oxygen, less than one percent. Analyses of vapor condensed inside the tungsten ribbon furnace will be performed during the next reporting period in order to test the decomposition hypothesis.

3. UO₂ FUELS RESEARCH

High Energy Impaction

Uranium dioxide powder packaged in stainless steel foil, rather than in specially fabricated containers, was successfully heated under conditions required for subsequent compaction in the Dynapak machine. The new Model 1220-B Dynapak was installed and checked out successfully.

Hot Isostatic Pressing

In a cooperative HAPO-BMI program, approximately fifty UO₂ fuel rods, vibrationally compacted at HAPO, were hot isostatically pressed at BMI. Results indicate that for high (> 98 percent TD) densities in mixtures of fused and micronized UO₂, approximately thirty w/o micronized UO₂ powder is required. To prevent gross distortion of the cladding during pressure bonding (at 10,000 psi, 1000 C) initial rod densities of ~ 75 percent TD are required, and the maximum particle size of the fused UO₂ component must be less than 20 mesh in rods having thin wall (0.015 inch or less) cladding.

Fuel materials having high surface area and small particle size are more amenable to precompaction by swaging than by vibrational compaction. Swaged rods comprising both FWR-grade and micronized UO₂ powders were hot isostatically pressed to high densities. Diametral variation and surface condition were satisfactory.

Fuel Evaluation

Melting Studies. Sintered UO_2 with irradiation exposures between 40 and 700 MWD/T has consistently revealed melting points higher than those of non-irradiated oxide. For example, the melting point of sintered UO_2 was observed to increase approximately 100 C during irradiation to 300 MWD/T . However, the melting temperature of fused UO_2 having a 300 MWD/T exposure did not change significantly. Fifty melting point measurements of specimens selected from known locations in each microstructural region of an irradiated 1.44 OD cross section all fell within 50 C of the melting point of unirradiated UO_2 . Presumably, the fused UO_2 does not become more refractory during irradiation because it contains no excess oxygen which seems to be required to stabilize higher melting compositions.

Fission Fragment Migration. Principal features of the gross fission fragment relocation curve were confirmed by complete analyses of a second irradiated UO_2 fuel section. The element (9/16-inch diameter, irradiated to 1000 MWD/T at 2.5 kw/inch) was sectioned and samples taken at 0.050-inch intervals along the diameter. Radiochemical analyses of each sample for Ce-144, Zr-95, Cs-137, Ru-106, and Pu yielded distributions with minima corresponding to high density columnar grains. The distribution of each isotope was qualitatively the same except for cesium, which showed a maximum depletion at a greater distance from the thermal center. In general, the more volatile the isotope, the greater the relative concentration variation along the diameter. Plutonium concentration varied by nearly a factor of two from the outer edge of the fuel to the high density columnar grain region (a distance of approximately 0.050 inch).

Electron Microscopy

The first examination of an irradiated (5000 MWD/T) UO_2 crystal in the electron microscope was successfully completed using reflection microscopy techniques. The fracture surface studied appeared similar to that of non-irradiated material previously examined, and no unusual features were revealed during the initial observations. Electron microscopy of replicas of the fracture surface of the irradiated single crystal showed a wide variation in surface characteristics over a small area. Well-formed cleavage steps, irregular rough areas, very smooth areas, and structure due to inclusions, localized crystal imperfections or cleavage artifacts were observed. Microscopy of an adjacent non-irradiated fracture surface is in progress.

High Pressure Melting of UO_2

Uranium dioxide was fused and held in a molten puddle for several minutes under 1500 psi of argon. Much lower vaporization losses

occurred than at atmospheric pressure. Crystal growth by vapor deposition was entirely suppressed. Decreasing the pressure while holding the temperature constant at the melting point resulted in increased vaporization rates and a sudden acceleration of crystal growth below 100 psi. The use of high pressures should permit controlled study of molten UO_2 and other volatile and dissociating materials.

Single Crystal UO_2 Standards

Spherical and rectangular specimens of single crystal UO_2 were purified and characterized. Some of them were sent to other sites for use in fundamental UO_2 studies. The specimens were made from large, single crystals of UO_2 which were heat treated in hydrogen containing ~ 1000 ppm H_2O at 1750 C to (1) eliminate very small inclusions, (2) assure stoichiometry, and (3) relieve stresses. Characterization studies upon the bulk materials included: O/U atomic ratio, density, porosity, carbon content, gas evolution, microstructure, and impurity levels (determined spectrographically). Radiographic examinations were completed upon the final specimens to confirm the absence of porosity and inclusions. X-ray diffraction measurements were made to confirm crystallinity.

4. BASIC SWELLING PROGRAM

Irradiation Program

Irradiation of four general swelling capsules is continuing. Two of the capsules are being operated at constant temperatures of 575 C, and one is being operated at a constant temperature of 625 C through all reactor operating conditions. The fourth capsule is operating at a temperature of approximately 400 C during reactor operation and at coolant temperature when the reactor is "down". The above capsules are installed in tandem, two to each test tube. One additional general swelling capsule is ready for charging, including all laboratory tests, and another is approximately thirty percent complete. This latter capsule will contain slightly enriched uranium samples requiring a minor change in fin configuration. Two capsules previously irradiated are being sent to Radiometallurgy for post-irradiation disassembly and examination of the specimens.

Tests have been conducted at the reactor capsule instrument facility to determine what improvements may be made to the capsule control instrumentation response. A prototype capsule is being used for these tests. Additional tests are being conducted in the laboratory to check out one set of previously used instruments that have been

reworked by the vendor. Upon completion of these tests, seven additional units will be returned to the vendor for rework.

An investigation is being conducted to determine the feasibility of installing a proposed data logging system at the reactor. Specifications for the new data logging system are complete and are now being reviewed.

Post-Irradiation Examination

An irradiated U-U Diffusion couple, GEH-14-281, which consists of a central core of uranium partially depleted in U-235 and a bonded shell of uranium enriched in U-235 has been processed again for metallography. This specimen, irradiated at < 200 C, showed no porosity in the as-irradiated state and none after vacuum annealing at 500 C for one hour. Annealing at 650 C for 100 hours, however, has produced porosity in the enriched uranium shell (irradiated to a nominal burnup of 0.3 a/o). This porosity, ranging from 0.05 to 0.5 μ , seems to be primarily located at the grain boundaries of newly formed, recrystallized grains. All of the enriched layer has not recrystallized and hence the porosity is widely scattered. The depleted uranium shows no porosity. Because of the scattered nature of the porosity in the enriched shell it is impossible to determine whether or not the number and/or size of pores change as the U-U interface is approached. There is no indication on this sample that diffusion of fission products into the depleted uranium core has occurred. No additional annealing of this specimen is anticipated at this time, but the sample will be stored for possible future examination.

Four other capsules, GEH-14-334, 335, 336, and 337, each containing six U-U diffusion couples similar to GEH-14-281, have been received in Radiometallurgy. Two of these capsules will be opened immediately and the specimens recovered. The other two will be stored until the samples are needed. The irradiation history of the capsules is indicated below:

Capsule No.	Cycles*	Flux (nv)	Exposure (nvt)	a/o Burnup	
				Shell	Core
334	2	5×10^{13}	1.31×10^{20}	0.234	0.011
335	2	5×10^{13}	1.31×10^{20}	0.234	0.011
336	4	5×10^{13}	2.47×10^{20}	0.442	0.022
337	4	5×10^{13}	2.42×10^{20}	0.431	0.021

*Irradiated in the MTR starting with Cycle No. 165.

The indicated burnup values are estimates based on reactor operations data and represent averages for the entire capsule. Flux gradients will cause some variation in exposure between specimens. Sampling techniques are currently being developed so that the enriched shell of these samples can be chemically analyzed for burnup.

Pore Size Distributions

Pore volume and pore density-size distributions, total pore volumes and pore densities, and total number of gas atoms/c.c. of specimen have been determined using a new computer program. The data received refers to all specimens which have been studied by quantitative metallography. Comparisons and analysis of results are currently under way.

Second Phase Distribution in Dilute Alloys of Uranium

Metallographic examination has continued of the thirty uranium specimens that represent three alloys in ten different conditions of heat treatment. Most of the specimens are now polished, etched, and replicated for electron microscope examination. The presence and morphology of the second phase precipitates will be intercompared when photomicrographs are available. The distribution of second phase will be determined by quantitative metallographic techniques.

5. IRRADIATION DAMAGE TO REACTOR METALS

Alloy Selection

A number of alloys for study under the new program for determination of irradiation effects on reactor structural materials have been selected. These alloys include the low alloy ferritic steels A302B and HY80, the austenitic stainless steels 304 and 348, a modified ferral aluminum alloy 406, a semi-austenitic steel AM 350, and the nickel base alloys Inconel alloy X-750, Inconel alloy 600, Hastelloy alloy X-280, and D979. Efforts to start procurement of two to four tons of each alloy will be initiated with the writing of specifications.

Personnel in Operations Research and Synthesis Operation have been contacted to develop a statistical approach to the planning of a testing and data processing program for each alloy to be studied. This program will help in utilizing effectively the materials, personnel, and testing facilities to be used in this program.

In-Reactor Measurement of Mechanical Properties

A number of in-reactor activation energy measurements were made in a creep capsule at the conclusion of a creep test. Specimen fracture precluded the completion of the activation energy study. The mode of specimen fracture was significantly different than experienced ex-reactor. Elastic modulus measurements indicate little or no necking occurred in the specimen before fracture. In addition, creep behavior usually associated with tertiary or third stage creep was not observed. The plastic creep strain at fracture was near five percent; about one percent of this was produced during the 310 C - 30,000 psi creep test. The remaining four percent occurred at the various temperatures in the 300 to 440 C range at stresses of 30,000 psi to 42,000 psi used in activation energy studies.

The capsule is still performing well after several months of continuous use. All thermocouples are still intact showing specimen fracture does not break the thermocouples as one might expect. The heaters are still operating and both the transducer and micropositioner are still operable. Again, as in previous capsules, no increase in helium consumption has been observed indicating the integrity of the Durock sealed helium tight header is not decreased with fast neutron exposure.

Modifications to provide water cooling in the next capsule containing a 20 percent cold worked Zircaloy-2 specimen have been completed and the capsule is ready for reactor charging. Laboratory evaluations of the capsule show that approximate operating temperature, with water cooling, will be 230 C. During evaluation, increases in helium pressure had no effect on specimen temperature at 200 C, while water was circulated in the cooling coils. Helium pressure, therefore, contributes nothing to temperature control under these conditions. The micropositioner and sensor performance met their design requirements.

Provision was also made in the capsule for a flux monitor in the immediate vicinity of the specimen. The monitor will be charged into the bellows reference tube. A dry lubricant on the flux monitor is used to facilitate discharging and charging of monitors during reactor shutdown. Flux monitors consist of small Al-Co, Fe, Ni, Ti, and Mn wires enclosed in the end of a high purity aluminum wire carrier.

Irradiation Effects in Structural Materials

The purpose of this program is to investigate the combined effects of radiation and reactor environment on the mechanical properties of structural materials. Special attention will be given to the

determination of mechanical property changes produced in metals by irradiation at elevated temperatures in contact with water.

A new hanger rod and quick disconnect assembly was fabricated for use in the ETR, G-7, hot water loop. The new assembly accommodates a central flux wire capsule that can be easily replaced and disassembled, permitting cycle-to-cycle flux determinations for each of the seven quadrant levels. Plates of type 347 stainless steel were rolled to final tolerance, finished in the 0, 10, 20, and 40 percent cold worked conditions, and sheared into blanks for machining notch tensile and bend test specimens. A shipment of 244 irradiated tensile specimens was received from the ETR during the month. These include 172 Zircaloy-2, 48 type 347 stainless steel, and 24 type 304 stainless steel specimens representing various neutron exposures to 5×10^{20} nvt. Preparations under way for testing these specimens include the installation of a new gamma radiation detection monitor; a macro-photography chamber, camera, and transfer assembly; and a disposal cask.

A series of tensile tests were conducted at 300 C and at various strain rates for both annealed and cold worked Zircaloy-2. In general, the tensile stress, uniform strain, and reduction in area varies linearly with the log of the strain rate.

Signs of strain aging were noted at 300 C. This was manifested by a lower strain rate sensitivity in tensile strength at 300 C compared to room temperature. Furthermore, a drop-in-load yield point was consistently observed for the annealed specimens at rapid loading rates (above 0.5 in/min). This type of yielding has been observed only rarely (three occasions) in the over 300 tests of Zircaloy-2 conducted previously on this program.

Control notch tensile tests were conducted at room temperature on annealed and cold worked Zircaloy-2. Among the parameters measured are the breaking stress, notch strength, and crack toughness (K_C). Both the breaking stress and notch strength increase linearly with log strain rate over the range 0.0025 to 2.5 min^{-1} .

Efforts to procure the gas loop facility in the ATR have been initiated. This facility will provide high temperature oxidizing gaseous environments necessary for irradiating the high strength nickel base alloys. The loop facility will provide a CO_2 or CO_2 and steam environment with an outlet temperature of 1400 F. During the past month basic heat transfer calculations were prepared for the scoping of this facility.

Damage Mechanisms

Impurity elements are known to have a pronounced effect on the physical and mechanical properties of metals. Recent investigations have shown that impurities also are an important factor in determining the response of a metal to neutron irradiation. The purpose of this program is to establish the nature of interactions between impurity atoms and defects introduced by irradiation, and to investigate the possibility of neutralizing the deleterious effects of the impurities by means of chemical stabilization.

Ten pounds of zone refined iron, 99.995 percent pure, are on order. This material represents the highest purity available in lots greater than a few grams. One hundred pounds of 99.95 percent pure iron are also on order. With both materials, attempts will be made to reduce the interstitial impurity content below 10 ppm. In the case of the zone-refined material, the goal will be one ppm.

A glass vacuum system has been constructed for the purpose of decarburization and for making controlled additions of carbon and nitrogen. The system has been evacuated to a pressure of 5×10^{-6} mm Hg at the furnace chamber in preliminary trials, with the furnace operating at 600 C.

High purity iron foils, 0.003 inch thick, have been received. These foils will be further purified and examined by electron transmission microscopy.

6. GAS GRAPHITE STUDIES

Low Density Graphites

A reactor concept requiring low density graphite was investigated several years ago at Hanford. To demonstrate the capability of producing low density graphite with the raw materials and facilities available at the time, three vendors supplied small quantities of low density graphite manufactured in full scale equipment. Interest in the original application has decreased; however, irradiations of the graphites have continued in view of their unique behavior and the possibility that a new application may develop.

To date, irradiations have been conducted at 500 to 600 C to exposures up to 9100 MWD/AT. Relatively low contraction rates were observed for each graphite even though the following parameters usually associated with higher contraction rates were included in their production.

1. Texas Lockport coke filler was used. The structure of this coke is not as readily graphitizable, nor does it produce the degree of preferred orientation attained with the cokes currently used in nuclear graphites to reduce contraction rates in the direction transverse to extrusion.
2. The highest temperatures reached during graphitization were between 2300 C and 2450 C. Irradiations of graphite processed at temperatures ranging from 1370 to 2700 C have shown that a significant reduction of contraction can be obtained by processing at temperatures higher than 2450.

Each of the vendors supplying low density graphite employed different methods to reduce density. The methods used and contractions observed are listed below in the order of increasing stability under irradiation.

- Graphite A. A volatile filler was added to the base mix, then extruded through a 4 x 4 inch die. Micrographs reveal the filler was completely volatilized during heat treatment leaving pores which are rectangular in cross section. Average bulk density was 1.22 g/cm³. The contraction observed at 8000 MWD/AT was 0.06 percent in the direction transverse to the extrusion and 0.13 percent in the parallel direction.
- Graphite B. Particle size and the particle-to-flour ratio were increased. Extrusion through a 4 x 4 inch die was not successful. By using a 16 x 16 inch die and processing through graphitization in that size, sound pieces were obtained. These were cut to 4 x 4 inch bars for purification. The average bulk density was 1.37 g/cm³. Contraction after 9100 MWD/AT was 0.065 percent in the transverse direction and 0.085 percent in the parallel direction.
- Graphite C. About 20 percent gum rosin was added to the initial mix. Four percent of the mix was coal tar pitch normally used as a binder in nuclear graphites. A 4 x 4 inch die was used for extrusion. Average bulk density was 1.30 g/cm³. Contraction after 9100 MWD/AT was 0.05 percent in the transverse direction and 0.067 percent in the parallel direction.

In-Reactor Creep

The results of the first irradiations at 150 and 300 psi (1.0×10^7 to 2.1×10^7 dynes/cm²) indicate creep rates in the reactor in the range 9×10^{-12} to 3×10^{-11} per dyne/cm² per 1000 MWD/AT at total exposures near 1500 MWD/AT at approximately 650 C. Approximately half of the rate is attributable to radiation acceleration, the remainder to the thermal creep rate.

The in-reactor rate constants are of the same order of magnitude as those estimated in England from irradiations of graphite springs at constant stress and from restrained growth experiments. However, the thermal rate constants appear to be about an order-of-magnitude larger than those currently being measured at Hanford.

Gas-Graphite Reaction Studies

Four sealed quartz tubes originally containing pure carbon monoxide in the pressure range from 30 to 95 psi at 25 C and a small graphite sample were opened after a reactor exposure of 258 operating days in the temperature range 550 to 650 C. The following observations were made: (1) there was a decrease in carbon monoxide accompanied by the formation of carbon dioxide. The ratio of number of moles of carbon monoxide lost to number of moles of carbon dioxide gained varied from 2.2 to 2.7; the average ratio for the four samples was 2.5; (2) a thin, mirror-like film was formed evenly and uniformly on the walls of the tube and on the graphite sample itself; electron microscopy revealed the characteristic spherical structure of carbon-black; and (3) helium had diffused into the quartz tubes from the reactor atmosphere.

A carbon, hydrogen, and oxygen analysis is being performed on the carbon deposit. Other tubes prepared with CO will be discharged soon. The results of these experiments will provide information on mechanism of the radiation-induced decomposition of carbon monoxide.

Graphite Combustion Studies

Tests of chlorine suppression of graphite combustion were made in the EGCR burning rig. Air-chlorine mixtures were used only in the moderator-coolant channels because severe chlorine leakage from the fuel element channel was noted in preliminary tests. Chlorine was introduced when the maximum graphite temperature was 775 C. The rate of temperature rise at this time was 30 C/min. Under these conditions, 6 w/o chlorine was necessary to stop the runaway and cause temperatures to fall.

Gamma Irradiation Facility

Twelve pieces of cobalt-60 (8700 curies total) were encapsulated in stainless steel and transferred to the Gamma Irradiation Facility. The 12 pieces were alternated in the underwater source holder with 12 pieces containing 10,600 curies. The cobalt-60 rods enclose a volume 5 inches in diameter by 18 inches high. The source intensity as measured by ceric sulfate dosimetry and a Victoreen rate meter was 1.56×10^6 r/hr. The variation in dose rate over an eight-inch vertical distance was ± 2 percent. Additional measurements were taken at several horizontal and vertical distances from the source so that the gamma intensity may be estimated for any position within the tank. Reinsertion of the glass loops for graphite oxidation experiments is under way.

Electrical Resistivity of Hastelloy-X

A gas heater for the PRTR gas loop is being designed which will have only Hastelloy-X in contact with the heated carbon dioxide gas. The gas will pass over electrically heated Hastelloy-X strips contained in a Hastelloy-X pressure vessel. Since no data were available on the electrical resistivity of Hastelloy-X at high temperatures, this property was measured from 25 to 975 C. The sample, 0.120 inch in diameter, was heated by an electric furnace during the measurements. The electrical resistivity at 25 C was 120.5 microhm-cm and increased linearly to 129.5 at 350 C. From 350 to 700 C the electrical resistivity value of 129.5 microhm-cm remained unchanged and from 700 to 975 C the electrical resistivity increased to 133.5 microhm-cm. The resistivity change of the loop heating elements is therefore within the limits of the saturable reactor control system.

7. GRAPHITE RADIATION DAMAGE STUDIES

Irradiation of Lampblack Graphite

Lampblack-based graphite samples that had been heat-treated to 3000 C were reirradiated for an incremental exposure of 1314 MWD/AT to a maximum total exposure of 2570 MWD/AT. After the first irradiation period some of the samples were heated to 3000 C causing the samples to grow and recover 20 percent of the radiation-induced contraction. During the second irradiation period the unannealed samples continued to contract but at a rate of 0.074 percent per 1000 MWD/AT, which was considerably less than the contraction rate of 0.18 percent per 1000 MWD/AT measured during the first irradiation period. The samples which had been heated to 3000 C after the first irradiation, however, contracted at the same rate as during the first irradiation period.

There appears to be two plausible mechanisms by which these changes have occurred. The first is that contraction is mainly due to the radiation annealing of stresses that are generated upon cooling from graphitizing temperatures (about 2800 C). Reheating to 3000 C would regenerate the stresses, thereby causing the annealed samples to continue to contract at the initial rate. Because some of the stresses would be relieved by radiation, the rate of contraction of the unannealed samples during the second period might be expected to decrease.

A second plausible mechanism is that the annealable component of contraction limits the rate of unannealable contraction. Therefore, the absence of the annealable component may have allowed the annealed samples to contract more rapidly than the unannealed samples during the second irradiation.

Further annealing and radiation experiments are in progress to test these two theories.

8. ALUMINUM CORROSION AND ALLOY DEVELOPMENT

Corrosion of Aluminum with Heat Transfer

One test to measure corrosion under heat transfer conditions was concluded during the month. Aluminum (X-8001) cladding was exposed in TF-3 in neutral deionized water at 600 F and 228,000 Btu/(hr)(ft²). The cladding was severely damaged by impingement attack at the upper end where the flow enters the test chamber; weight loss data would have little meaning. The rod will be sectioned and examined metallographically to determine depth of attack.

Some coupons were included to determine weight loss without heat transfer in this system. The corrosion rate after 110 days for X-8001 was 2.6 mils/mo; for A-288, the rate was 1.7 mils/mo.

In-Reactor Corrosion Loop

The second attempt to weld disks of silver under the surface of aluminum coupons (in order to increase the beta flux at the surface when exposed in-reactor) again resulted in swelling when autoclaved, presumably from water leaking in. It was decided to delete these pieces from the first charge rather than risk contamination of the coolant with Ag-110. A few Zircaloy-2 and stainless steel coupons containing silver disks welded-in will, however, be included in the charge; these should show whether crud deposition can be increased by internal beta radiation. The coupon holders for the first charge have been loaded and are ready. The production test is still waiting approval.

9. AEC-AECL COOPERATIVE PROGRAM

A series of pit-shaped defects were electro-jet machined on the inside surface of a 0.680-inch ID x 0.035-inch wall tube to study the sensitivity of the ultrasonic test to this type of defect. Three types of crystals were used to measure the ultrasonic response to these pits -- a line-focused crystal, a rectangular unfocused crystal, and a spot-focused crystal. All of these crystals are ten megacycle, lithium sulfate. The ultrasonic response to these pits was measured at the optimum entry angle for each crystal, using both the longitudinal and the transverse test alignments. The longitudinal test data were rather erratic and difficult to reproduce, but with a relatively high gain setting the pits could be detected. The transverse test data are presented in the table below.

Ultrasonic Response as a Function of Artificial Pit Size on the Transverse Test at a 20 degree Entry Angle

Nominal Pit Size (mils)		Ultrasonic Response of Each Crystal		
<u>Dia.</u>	<u>Depth</u>	<u>Line Focused</u>	<u>Spot Focused</u>	<u>Unfocused Rectang.</u>
		<u>Gain 60</u>	<u>Gain 60</u>	<u>Gain 51</u>
4.0	2	10	10	20
6.7	4	35	72	55
13.0	8	62	89	80
16.2	16	70	102	95

Despite the small diameter of these pits, their response compares favorably with the response from notches. For example, using the unfocused rectangular crystal at a gain setting of 51, a transverse notch 3 mils deep by 30 mils long gives a full scale response, while a pit 6.7 mils in diameter by 4 mils deep gives a 55 percent response.

The ultrasonic response as a function of entry angle was measured on the transverse test using an unfocused rectangular 10 megacycle lithium sulfate crystal. The measurements were made on notches 3 mils deep by 30 mils long on both surfaces of a 0.680-inch ID by 0.035-inch wall tubing. The three distinct peaks, which were previously observed when using the line focused crystal, reappeared at 17, 20-21, and 26 degree entry angles. The amplitudes of the signals from both inside and outside surface notches were nearly the same at these angles. This situation makes it possible to test for defects of the same size whether they occur on the inside or on the outside surface.

The data for curves of response versus change in metal path length were obtained at each of the peak angles. Again, as when using the

line focused crystal, the slope of this curve was smallest at the 20-21 degree entry angle. The rectangular crystal could be moved 25 mils off the center of the tube before a ten percent reduction in response occurred as compared to only 7.5 mils movement for the line focused crystal.

In summary, the unfocused rectangular crystal is less sensitive to changes in entry angle but is slightly more sensitive to changes in metal path length than the line focused crystal. In addition, the rectangular crystal has a better signal-to-noise ratio and is less sensitive to transverse movement of the crystal than the line focused crystal by a factor of three. The data indicate that the unfocused rectangular crystal would give better results in the test for transverse defects than the line focused crystal.

10. REACTOR STUDIES PROGRAM

Advanced Reactor Concept Studies

A program for conceptual design of plutonium-fueled reactors for space applications was prepared at the request of the AEC as part of a program encompassing fuel and physics research. This program would include comparisons of plutonium- and uranium-fueled reactors and the conceptual design of a plutonium oxide or plutonium carbide fueled reactor during FY 1963. Subsequent concept studies would be based on other plutonium compounds.

Conceptual design activity during the month was focused on a liquid metal-cooled solid fuel reactor for generation of electric power on space vehicles. A turbogenerator power conversion system was selected for this study because of the meager useful information available on thermionic systems and fuel materials. An advanced reactor for this application would employ a direct cycle coolant system operating at 1000-1200 C in the reactor. Rubidium, potassium, and sodium are potentially suitable as boiling coolants in this temperature range. Plutonium oxide fuel clad in tantalum appears to be the most promising fuel element materials. A reactor power rating for the study has not been firmly established but will probably be within the 1-10 Mw(e) range in order to meet potential power requirements in post-1970 decades.

An attempt was made to evaluate inherent control concepts on fast and semi-fast reactor concepts utilizing hand calculation methods. It became apparent that the accuracy attainable was not acceptable for even rough approximation. Therefore, multigroup transport computer

calculations are being prepared to carry out this work and to evaluate reactor core concepts.

A research and development program on supercritical pressure water coolant technology was prepared for submission to the AEC. This program includes fuel, materials, and reactor engineering studies for a five-year period aimed at developing the technology required for design and construction of a prototype reactor.

D. RADIATION EFFECTS ON METALS - 5000 PROGRAM

Preparation, encapsulation, and irradiation of a variety of molybdenum specimens have continued. Both single crystal and polycrystalline specimens containing several levels of carbon as an interstitial impurity are being studied to establish the combined effect of neutron irradiation and carbon impurity level on the properties of the metal.

The orientations of 62 single crystal tensile specimens have been determined. It is hoped that five unique orientations can be investigated in crystals from each of the three classes containing different concentrations of carbon (10-30, 100-200, and 400-500 ppm). Since four exposure levels are to be investigated, 60 specimens are required. The orientations of additional specimens are being determined to provide characterized specimens which meet the requirement of matched orientations.

A capsule containing four single crystal bend test specimens has been submitted for irradiation to 10^{19} nvt (fast). One crystal containing 10-30 ppm C has two polished perpendicular faces of the form (110) and (100); the second, also containing 10-30 ppm C has faces of the form (110); and the third and fourth, containing 400-500 ppm C, each have polished perpendicular faces of the form (110).

Five single crystal x-ray specimens with polished (100) and (110) surfaces have been prepared. Additional specimens will be prepared and irradiated in the same capsules containing the single crystal tensile specimens to assure equivalent exposure conditions.

Nine stored energy specimens from polycrystalline molybdenum have been sealed in three capsules and 24 polycrystalline tensile specimens have been sealed in six capsules for irradiation. Polycrystalline foil specimens, representing four carbon impurity levels, were sealed in 24 capsules and submitted for irradiation.

Recovery of cold work damage in molybdenum has been characterized by means of microhardness measurements. An isochronal anneal was conducted

for one hour periods and temperature increments of 25 C over the range from 25 C to 1125 C. Recovery occurred in a single stage, centered at 1025 C. An isothermal anneal was conducted at 1000 C. The two annealing curves were used to calculate the recovery kinetics. The recovery stage was determined to consist of two separate processes, one occurring with an activation energy of 2.2 ev and the other about 4 ev. The latter process is most certainly recrystallization. Reaction orders will be determined on the electronic digital computer. The computer program has been written to calculate the kinetics of future recovery studies.

E. CUSTOMER WORK

Radiometallurgy Laboratory

Radiometallurgy Examination. Twelve irradiated Co-60 capsules were removed from their irradiation containers, re-encapsulated, and transferred to the gamma irradiation facility (RM 358). Metallographic examination, x-ray diffraction studies, and density determinations from the top, midpoint, and bottom of the second core sample from Purex Waste Tank No. 103 indicated that a much greater quantity of crystalline material was present in this sample than was present in the first sample (RM 361). Five uranium metal samples from fuel element failures were x-rayed and results indicate the samples have a preferred orientation of the uranium grains (RM 437). Metallographic examination of the dingot uranium failure from 2088-B disclosed the element had been partially cracked in the transverse direction. The end closures were found to be sound (RM 442).

A sample of cold uranium dioxide from ORNL was run to correlate the pattern produced on our equipment with the x-ray diffraction equipment at ORNL and GE-VAL. The data were forwarded to W. V. Cummings of GE-VAL (101.4C).

Equipment Development. All cell castings for the high level utility cell have arrived and have been erected. Due to careful work by the vendor, the castings fit together very well. Installation of the milling machine and cell service facilities are progressing on schedule. Major pieces of equipment which have not been received are the lathe, extended reach manipulators, and the flat sample former. The lathe is scheduled to be shipped 3/30/62, and the manipulators are scheduled for shipment on about 5/1/62.

Preliminary measurements of the gamma flux from selected UO₂ samples were so low that counting periods up to 10 minutes were required using the larger diameter collimator. The poor signal-to-background ratios were

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caused by the presence of Co-60 in the stainless steel cladding and a two-year fission product decay period of the fuel material.

Micro sampling of irradiated UO₂ continued in the mock-up behind temporary shielding to evaluate a number of candidate sampling drills. Seven samples were removed successfully from a one-half inch diameter UO₂ wafer using a 0.020-inch diameter, diamond impregnated drill. The samples were collected with a vacuum pickup and submitted for fission product analysis.

Equipment to burst test 5/8-inch OD Zircaloy tube fuel cladding having 0.020-0.030-inch wall thickness has been designed and successfully tested on unirradiated cold samples.

An improved method for handling NaK from irradiated capsules has been investigated and will be used in forthcoming NaK opening operations in E Cell. Basically, the procedure consists of opening the NaK capsules under kerosene, allowing the NaK to settle to the bottom of a kerosene container and then reacting the NaK with Cellusolve. Reaction products are soluble and leave a clear liquid which is easy to handle and discard to waste. Although the hazard of a fire in the cell is reduced by this process, liquid nitrogen, which has proven to be an excellent fire extinguisher, will be kept handy if such an emergency should develop.

Metallography Laboratory

Procedures which are acceptable to both the Radiation Monitoring and the Industrial Hygiene Operations have been worked out for processing thorium and pure beryllium metallographic samples in the 306 Building laboratory. The established procedures, without the use of a hood or glove box, use the Buehler Automet Polisher for grinding and preliminary polishing. Subsequent final polishing is done with the Syntro vibratory polishers.

N-Reactor Charging Machine

Modifications. Modification of 15 fuel element magazines and 15 magazine end plugs was completed. As originally fabricated, polyethylene liners in the magazines extended beyond the bayonet connections on the ends of the magazines. The liner tended to separate from the magazine tube at this point. The liner was removed to a point ahead of the bayonet connection. An aluminum sleeve was added to the end plug to replace the polyethylene removed.

The original pressure roller supports deformed under operating conditions. Stronger supports were designed and installed.

The plug unlatching mechanisms on the front of both plug conveyors were disassembled, modified, and reassembled. This was required because of difficulties encountered in removing and replacing plugs in the barrier wall.

All of the high pressure water piping was replaced with piping which complies with code requirements, and the system satisfactorily tested.

Installation of a pressure gauge and manifold arrangement on the consoles for monitoring hydraulic system and charging water pressures was completed.

Testing. Additional tests were conducted on the vertical lift drive system to determine the hydraulic pressure versus speed relationship of the modified system. As mentioned in last month's report, modification of the system consisted of replacing gears in the center miter box to decrease the speed and increase the torque available by a factor of two.

Two columns of fuel elements were charged from one magazine to another. During charging of the first column, several carbon steel shoes were stripped from the fuel elements. This was caused by the plastic nozzle adapter liner, previously mentioned, being extruded into the process channel by the 6000-pound magazine end thrust and the sharp end of the shoes catching on the plastic. After the plastic liner was replaced with an aluminum liner, a second column was charged without incident.

The life test of the magazine drive roller system was completed. The drive roller did not successfully complete this test. After 65 cycles, the bond between the rubber and the steel core of the drive roller separated. Corrective action has been initiated.

Fission Product Transient Samples for Phillips

The fabrication of 72 fission product transient samples for Phillips Petroleum Company is continuing. The coextruded tubular elements are to be five inches long with a nominal 0.8-inch OD. The inner and outer cladding is high purity aluminum 0.020-inch thick and the aluminum-uranium and aluminum-plutonium cores are 0.020-inch thick. Two of the problems encountered in developing the coextrusion process, a lack of core-to-clad bonding and a "shifting" of the core, appear to have been solved. Redesign of the billet in order that a larger outgas tube could be located on the lead end of the billet and closer control of extrusion conditions appear to have improved the quality of the tubes.

Analysis and sampling of the plutonium-aluminum alloy core material has been completed and final machining of the cores is in progress. The first two plutonium-bearing cores extruded showed a greater "dogbone"

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or core thickening at the trailing end than earlier extrusions containing aluminum-uranium with 1100 aluminum cladding. A more severe end configuration is being machined into the remaining plutonium-aluminum cores to minimize this effect.

Fuel Samples for Aluminum Corrosion Studies

The Coolant Systems Development Operation, HLO, requires 150 fuel elements with three different aluminum alloy claddings to test the corrosion properties of the cladding under irradiation conditions. The fuel is an Al - 20 w/o Pu alloy clad by coextrusion. The fuel will be one-inch diameter and 7-1/2 inches long with two cladding thicknesses, -0.050 inch and 0.130 inch. Seven billets were cast for development to determine the size and configuration required for the coextrusion billet assemblies and to determine the extrusion properties of the Al - 20 w/o Pu alloy.

J. J. Cadwell
For Manager, Reactor and Fuels Research
and Development

PHYSICS AND INSTRUMENT RESEARCH AND DEVELOPMENT OPERATIONMONTHLY REPORTMARCH 1962FISSIONABLE MATERIALS - O2 PROGRAMREACTORN-Reactor Exponential Experiments

Flux traverses taken in the N-Reactor exponential mockup are being analyzed with fast source theory harmonic corrections and examined in detail to determine an internally consistent, reasonably precise set of final buckling values. Part of this work has been the selection of a set of extrapolation distances which best fits all experiments. The results are 2.0, 1.5, and 2.0 inches for side to side, front to rear, and vertical traverses, respectively. The vertical extrapolation distance, λ_c , is uncertain by about 0.5 inches and an improvement in precision is being sought. The uncertainty in the buckling from 0.5 inches in λ_c is 7.3% if the data points as close as 12 inches to the top of the pile are included in the fit. The error is reduced if more points near the top are dropped, but these points are needed to confirm the existence of an equilibrium region of exponential flux decay, especially for the cases with control rods where data points near the bottom are lost. The total standard deviation in the buckling (including three extrapolation distances and the fit to the exponential) should be six μB if $\sigma(\lambda_c)$ can be reduced to 0.25 inches.

The control rod spacing in the mockup exponential was found to be incorrect. The four corner rods were eight inches too close to the center of the pile. The pile is being rebuilt to move the rear rods eight inches backward and the front rods eight inches forward. The rebuilding affords the opportunity to add several modifications:

1. Addition of a/6, b/6 source positions for traverses with reduced harmonics and for traverses with eight sources to reduce harmonics with control rods inserted.
2. Addition of control rod extensions into the base to reduce the transient in the vertical flux caused by inserting rods.
3. Replacement of 1/4-inch graphite shims by 1/8-inch aluminum shims in the shutter gap to reduce the leakage through the shutter.

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4. Addition of a horizontal traverse bar below the shutter to provide a more sensitive measurement of harmonic correction effects.

The change will alter the control rod strength results but will not affect the previously completed buckling and flooding effect measurements.

Optimization of Retubed Lattices

All of the natural and enriched overbore fuel has been received for the C-Reactor exponential pile. Measurements will be resumed when the Pu-Be sources are returned from Mound Laboratories.

PCTR Measurements for Large Diameter Fuels

A draft of an article entitled "Lattice Parameter Measurements for Solid and Concentric Tube Uranium Fuel in Graphite Lattices" has been written for submission to Nuclear Science and Engineering. The manuscript contains the lattice parameters for the concentric tube fuel (1.66 x 1.12-inch tube inside a 2.5 x 2.0-inch tube) in 8.375- and 10.5-inch lattices and the 2.5-inch solid fuel in a 10.5-inch lattice, neutron temperatures for the concentric tube fuel in the 10.5-inch lattice, and the dual fission chamber calibration for fast fission effect measurements.

Data Correlation and Analysis

An earlier derivation⁽¹⁾ of the effective neutron temperature in the moderator for graphite-uranium systems has been extended to include the 1.66-inch fuel in a 6.5-inch lattice. The derivation was revised to include the effects of the new copper and gold resonance integrals. The neutron temperature used in one or six group P_3 calculations of the cell flux distribution is adjusted until the experimental ratio of the flux in the graphite to the flux in the fuel is matched. The derived neutron temperature divided by the moderator temperature is then plotted versus the scattering parameter, $\Delta = 4\Sigma_g/5\Sigma_s$. The equations of the fitted lines are $T_n/T_m = 1 + .64\Delta$ for the one group method and $T_n/T_m = 1 + .25\Delta$ for the six group method.

The six group method predicts an increase in moderator temperature over the physical temperature of $\Delta T = 44^\circ\text{C}$ for the concentric tube fuel in a 10.5-inch lattice. The values of ΔT from lutetium measurements are 38°C at the

(1) D. E. Wood, The Effects of Neutron Temperature and Epithermal Flux on Lattice Cell Flux Traverses, pp. 67-74 of NPRO Quarterly Report for July-September 1960, HW-67219.

cell edge, 48°C halfway through the graphite, and 62°C at the fuel-moderator boundary.

The Modified Heavy Gas Model and Neutron Spectra

An iteration scheme similar to that of Hurwitz was used to produce a spectrum from the modified gas model for a graphite-uranium system. This spectrum agrees with the spectrum computed directly from the code SPECTRUM everywhere to within a few percent. Agreement of the modified gas model with Coates' and Gayther's experimental results is also within a few percent, while the unmodified gas model disagrees by as much as twenty percent. Thus, the modified gas model shows promise for graphite systems. These results were obtained in time for inclusion in a paper to be presented at the Brookhaven Conference on Rethermalization, April 30-May 2, 1962. The water-uranium system also under investigation presents difficulties, since preserving the average energy loss per collision does not give a smooth transition from the modified heavy gas model to slowing down theory at high energies. Currently the effects of preserving the logarithmic decrement rather than the average energy loss per collision in modifying the gas model are being investigated. Such a modification insures that the gas model gives the correct high energy slowing down theory.

Computational Programming Services

The reactor kinetics code, TRIP004, underwent a number of rigorous tests during the month and several errors were located and corrected. A TRIP descriptive document is in rough draft form. Some experiments were made with the debug assistance possibilities of the new Fortran system and a description of the main points of interest observed during these tests was given to EDPO for distribution to on-site Fortran users. The KERNEL code, which calculates water scattering kernels by the Nelkin model, was revised in accordance with information recently received from KAPL, to select ranges on the basis of energy transfer rather than initial energy. An addendum was issued to HW-68806, describing formulation changes and increased capabilities of VTACL, the exponential pile data processing code.

Resonance Escape and Nonleakage Probabilities in Heterogeneous Lattices

It was pointed out in last month's report (HW-72902 B, p. B-3) that, for a heterogeneous lattice, the material buckling B_m^2 is a function of the shape and size of the system, and that appreciable error may result in buckling values derived from exponential experiments if no correction is made for the geometrical effect. In order to determine the magnitude of the correction, expressions are needed for the resonance escape and nonleakage probabilities in a heterogeneous lattice cell within a finite assembly.

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Such expressions have been found for a square cell in both critical and sub-critical assemblies, under the assumptions that age theory is applicable, that absorption occurs only at a single sharp resonance, and that the extrapolated dimensions of the assembly coincide with cell boundaries. In this case the quantities are independent of position of the cell within the assembly. Evaluation of these expressions for a square cell, as well as extensions of the formulae to other geometries, is underway.

Instrumentation

All graphite nuclear purity tests and diffusion length measurements were completed for N reactor in a cooperative project with personnel of Operational Physics and Instrument & Electrical Development of IPD. General performance of the BF_3 proportional counters and all electronic instrumentation was excellent. Examination of the counting data obtained showed the positioning accuracy of the detectors was satisfactory for all tests.

Tests were started on one gamma energy analysis chassis of the NPR Fuel Failure Monitoring System. The transistorized chassis, the GE-APED prototype of 24 such units to be fabricated to HAP0 specifications, performed satisfactorily in several tests completed to date. A number of precision techniques had to be developed to permit testing to the accuracy limits required by the HAP0 specifications. The amplification linearity was measured to an accuracy of $\pm 0.05\%$, and the emitter follower preamplifier output impedance was measured to an accuracy of ± 1 ohm. Other tests completed included amplifier linearity, gain and discriminator stability versus temperature, and line voltage variations. Tests were formulated for the count-rate meters, both high and low voltage supplies, and the differential alarm module.

Advice and assistance were rendered to Coolant Testing, IPD, for instrumentation improvement of the KER Fuel Failure Monitor. The problem, basically, is one of improving the general system reliability. Methods of placement of the neutron detectors were presented, along with methods of obtaining circuit redundancy for alarm tripping.

Advice and assistance were given on the KER Test Hole Fuel Failure Monitor of Irradiation Testing, IPD. A brief study resulted in the outlining of a proposed monitor system to include both a delayed neutron channel and a gross gamma channel.

A study was completed and a report prepared concerning the determination of heat generation rates in the reflector and in both the thermal and biological shields of existing Hanford reactors as a function of the process material utilized in the fringe process tubes.

Systems Studies

Preparations were made to simultaneously measure neutron flux and tube outlet temperature transients resulting from rod movements made during equilibrium operating conditions at 105-KW. An unscheduled outage delayed the tests and they are now planned for the next equilibrium operating period.

The Bailey Meter Company plants in Cleveland, Ohio, were visited, with representatives of NPR Project Operation, to review and discuss NPR instrumentation problems. The analog simulation work being done by Bailey was also reviewed. They are presently testing a simulation of the secondary loop which is more detailed than previous HAPO simulations in some respects, but considers all steam generators, dump condensers and surge tanks as lumped together into one of each type. They plan to gradually simplify the simulation by determining which approximations can be made without causing appreciable error. If sufficient simplification can be made in this way, they will then attempt to include additional process units.

Arrangements were made to obtain the dynamic characteristics (frequency response data) of many of the instruments and control valves being supplied by Bailey for the NPR. This information will be helpful in setting up an accurate simulation of the NPR.

A side trip to the Enrico Fermi Reactor at Monroe, Michigan, was made to obtain more detailed information on their reactor simulator. It was interesting to note that, in addition to its intended use as a training aid, the simulator had been extremely useful as a debugging and analysis tool. An example was given of a secondary loop pressure control problem which appeared to be unsolvable until it was turned over to the systems analysis group. This group used dynamic analysis techniques to isolate the cause of the problem and then used the simulator to test proposed solutions. The system is now operating satisfactorily.

A meeting was held on March 16 in Los Angeles to inform Information Systems, Inc., about HAPO requirements for isolating the NPR Flow Data Logger input from the Flow Monitor transducers. Since the discussions provided no solution to the problem acceptable to ISI, the difficulty has become mainly administrative.

Meetings held here on March 6 and 12 were aimed toward appraisal of proposed methods for NPR Flow Monitor (Avien, Inc.) automatic component testing, and NPR Zone Temperature Monitor (Panellit) en masse limit setting.

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The NPR secondary loop stability analysis has been completed. The EASE and GEDA analog computers were used to simulate the system. The results obtained contributed to a better understanding of the heat dissipation system. Different types of controllers and their settings were evaluated as well as an alternate method for controlling the dump condensers. The response to manual control was investigated.

An eleven-node kinetics simulator for the reactor instrumentation study is being fabricated. All necessary parts have been delivered and fabrication is 75% complete. This simulator will use two amplifiers per reactor section and allow the reactor parameters to be varied easily. This problem will use 130, out of a total of 132, amplifiers on both computers. It is anticipated that use of such a large percentage of the total available equipment will cause some difficulties in simulating the system.

Further analog studies of reactor speed-of-control characteristics were requested by IPD. The studies have been completed and the results forwarded to IPD. The reactor model used was the same six-node model as that used in the November 1961 speed-of-control simulation. This model affords investigation of diffusion in the vertical direction and simulation of VSR (vertical safety rods) insertion fully or part way into the pile at various speeds. Reactor power and temperature excursions due to various positive reactivity inputs were recorded. A number of reactor parameters were varied in an attempt to obtain information relating to any likely set of parameter values. Among the parameters varied were VSR strength, coolant water reactivity worth, metal temperature reactivity coefficient, operating power level, and VSR instrumentation delay. It is anticipated that as changes in values of measured reactor parameters occur, interpolation of the data collected in this study will provide a basis for prediction of reactor behavior due to introduction of positive reactivity disturbances under various operating conditions.

Information was collected to determine probable required information channel capacities and storage capacities for a digital approach to an NPR Simulator for comparison with the analog approach. The Electronic Data Processing Operation has agreed to assist in the determination. All variables with their estimated required accuracy and highest frequency, and all equations relating the variables were collected. The required information channel capacity of a variable is determined on the basis of six samples per cycle times the highest expected frequency in cycles per second times the number of bits (binary digits) required to express the variable to the desired accuracy. Storage capacity and additional information channel capacity is required to solve the equations.

SEPARATIONSExperiments with Plutonium Solutions

Criticality studies were continued with $\text{Pu}(\text{NO}_3)_4$ solutions in the 14-inch diameter stainless steel sphere. Measurements were made with the sphere bare, reflected with $\frac{1}{2}$ -inch and one inch of paraffin and with four inches of concrete. Two of the experiments were carried out to measure the effect of a 0.030-inch cadmium shell over the vessel backed by four inches of concrete. The experiments with the paraffin reflected sphere provided additional data from which to obtain more consistent values for the reflector savings of paraffin. Calculations pertaining to the reflector savings of paraffin are presented in Table I.

TABLE IREFLECTOR SAVINGS

<u>Reflector</u>	<u>Pu Concentration (g/l)</u>	<u>Critical Volume (l)</u>	<u>Radius (cm)</u>	<u>Reflector* Savings (cm)</u>
0.044" SS	97	26	18.38	
		(Corr. for SS, 26.6)	(18.52)	
	53.6	30	19.28	
		(Corr. for SS, 30.6)	(19.40)	
0.044" SS } + $\frac{1}{2}$ " Paraff. }	97	22.0	17.38	1.14
	53.6	25.2	18.19	1.21
0.044" SS } + 1" Paraff. }	97	19.7	16.75	1.77
	53.6	22.2	17.45	1.95
0.044" SS } + 4" Concrete }	97	18.0	16.26	2.26
	51.6	20.4	16.95	2.45

* Preliminary values, subject to change on further analysis.

The previous value of 1.56 cm, quoted for the reflector savings of the $\frac{1}{2}$ -inch paraffin and 0.044-inch stainless steel combination reflector, is now given as ~ 1.2 cm, which is a more reasonable value. The reflector savings for one inch of paraffin and 0.044-inch of stainless steel is about the same as quoted before. The effect of covering the vessel with a 0.030-inch thick layer of cadmium to reduce the effectiveness of the

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concrete reflector is shown in Table II.

TABLE II
EFFECT OF 0.030-INCH Cd SHELL ON CRITICAL VOLUME

<u>Reflector</u>	<u>Pu Concentration (g/g)</u>	<u>Critical Volume Increase %</u>
4" Concrete	231	10.2
4" Concrete	51	10.8
*4" Concrete	33	12.7
*1" Paraffin	127	11.4

* Reported previously

As the Pu solution becomes more dilute, the cadmium becomes slightly more effective in reducing the reflectivity of the concrete.

The results of the criticality measurements completed during March are summarized in Table III.

Two new critical assembly vessels were received from Tavco, Inc., after considerable difficulty and delay in their fabrication. These were a 13½ and a 15-inch diameter stainless steel sphere. These units will be used in subsequent criticality studies with Pu(NO₃)₄ solutions.

Miscellaneous Experiments and Calculations for Nuclear Safety Specifications

Measurement of k_{∞} in the PCTR for Aqueous U²³⁵ Solution

The final result of the experiments in the PCTR to determine the limiting critical concentration of the 93% enriched UO₂F₂ solution is 12.97 ± 0.05 g U/g (12.09 g U²³⁵/g). The value determined at ORNL from large scale critical experiments is 12.1 g U²³⁵/g. The method of determination, utilizing the PCTR, is thus in excellent agreement with the value obtained from critical experiments.

Limiting Critical Concentration of U²³⁵ in Aqueous Solution by Monte Carlo Techniques

Monte Carlo calculations are being made of the limiting critical concentration of U²³⁵ in an aqueous solution. The limiting concentration has been

TABLE III

**CRITICALITY STUDIES WITH PLUTONIUM SOLUTIONS
IN 14-INCH DIAMETER STAINLESS STEEL SPHERE**

(Measured Sphere Volume 23.22 Liter; Wall Thickness, 0.044-inch)

Date	Exp. No.	Reflector Condition	Pu (g/s)	Acid Molarity	Sp.Gr.	H ₂ O (g/s)	Total NO ₃ (g/s)	H/Pu (Atomic Ratio)	Critical Vol. (s)	Critical Mass (Kg Pu)
2-27-62	1141065	Bare	30.2	1.10	1.090	934	118	829.7	~37.0	~1.12
2-28-62	1143066	{ .030" Cd. + 4" Concrete	50.9	3.88	1.201	849	294	461.0	22.7-.16	1.16
3- 1-62	1143067	{ .030" Cd. + 4" Concrete	50.9	3.88	1.201	849	294	461.0	22.6-.02	1.15
3- 2-62	1143068	4" Concrete	51.0	3.61	1.204	841	307	454.7	20.4-.05	1.04
3- 5-62	1143069	4" Concrete	120.0	5.57	1.366	842	496	197.4	19.7-.08	2.36
3- 6-62	1143070	4" Concrete	135.0	5.22	1.380	752	481	157.1	18.9-.09	2.55
3- 7-62	1143071	4" Concrete	231.0	4.83	1.511	722	559	88.0	19.6-.03	4.53
3- 8-62	1143072	{ .030" Cd. + 4" Concrete	232.0	4.86	1.508	738	546	89.5	21.6-.04	5.01
3-12-62	1143073	4" Concrete	96.4	3.67	1.281	842	348	241.0	18.0-.04	1.74
3-13-62	1141074	Bare	96.8	3.69	1.280	840	345	239.5	26.0	2.52
3-14-62	1141075	1/2" Paraffin	97.4	3.70	1.282	845	342	239.4	22.0-.01	2.14
3-15-62	1141076	1" Paraffin	97.6	3.70	1.280	840	344	237.5	19.7-.03	1.92
3-16-62	1141077	1" Paraffin	53.6	3.68	1.211	845	305	434.9	22.2-.01	1.19
3-26-62	1141078*	1/2" Paraffin	53.6	3.68	1.211	845	313	434.9	~25.2	~1.35
3-28-62	1141079*	Bare	53.6	3.68	1.211	845	313	434.9	~30	~1.61

* Further chemical analyses are being made of the plutonium solutions in these experiments.

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determined both from k_{∞} measurements in the PCTR, and from critical experiments at ORNL⁽²⁾. The calculations for the U^{235} solutions will be used to check the HISMIC (Monte Carlo) code against a well determined experimental value. A Monte Carlo calculation will then be made to evaluate the limiting critical concentration for aqueous Pu solutions in an effort to resolve existing differences which exist between computed and measured values.

In the case of Pu solutions, computed values for the just critical concentration in an infinite system are one to two g/l less than the measured value in the PCTR.

From the PCTR experiments the limiting critical concentration of 93.15% enriched UO_2F_2 solution is 12.9 ± 0.05 g U/l. Three runs of 1,000 neutron histories each have been completed on the HISMIC code for this particular concentration; the average value of k_{∞} for the three runs is 0.970. Additional calculations are being carried out to determine the rate of change of k_{∞} with uranium concentration.

Computational Program in Support of Plutonium Critical Mass Experiments

Shape Conversion of Partially Filled Spheres

A method was derived for making "constant buckling" conversions of measured critical masses of partially filled spheres to equivalent full spheres of reduced radii. Criticality is frequently reached when the sphere is partially full. Sometimes the unit is subcritical when full. In order to extend the usefulness of the data obtained from the experiments, it is proposed to provide an expression for the buckling of partially filled spheres which will, a) give a proper extrapolation for criticality in spherical geometry when the sphere is subcritical and full, and b) provide a correction to the critical, partially filled spheres to obtain the equivalent critical volume in spherical geometry. When successfully checked experimentally, the method will extend the usefulness of the experiments with considerable saving of money and time. In the derivation, note is first taken of the difference in buckling expressions for the hemisphere and the sphere:

$$B^2 (\text{sphere}) = \frac{4/3 \pi R^4}{V} = \frac{129.9}{V_s} \text{ and}$$

$$B^2 (\text{hemi}) = \frac{(4.49)^2 (2/3 \pi R^4)}{V_h} = \frac{189.5}{V_h}$$

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- (2) Gwin, R. and D. W. Magnuson, "The Measurement of Eta and Other Nuclear Properties of U^{233} and U^{235} in Critical Aqueous Solutions", Nuclear Science and Engineering, 12, 364-380 (1962).

which exhibit the obvious comparison that V (equivalent full sphere) = V (hemi) (.686). It is then postulated that in general for a partially filled sphere (of diameter $2a$ and solution height of h_a):

$$B_h^3 = \frac{X(h)}{V(h)} .$$

It is assumed that $X(h)$ is linear in the surface-to-volume ratio of the partially filled sphere, i.e.,

$$X = A\left(\frac{aS}{V}\right) + B, \text{ where } a \text{ is the radius of the vessel, and } S, V$$

are the surface area and volume of the partially filled sphere, respectively. Evaluation of the constant yields:

$$X = 10.7 + 39.75\left(\frac{aS}{V}\right), \quad 3 \leq \frac{aS}{V} \leq 4.5.$$

The volume of the sphere whose buckling equals that of a partially filled sphere is given by:

$$V \text{ equ.} = \frac{129.9}{B^3(h)} \text{ and thus}$$

$$V \text{ equ.} = \frac{129.9}{X(h)} V(h).$$

When account is taken of extrapolation distance, the calculation must proceed in reverse, using a derived solution height, h' , given by:

$$h' = \frac{h_a + 2\lambda}{a + \lambda} \text{ to find the equivalent "extrapolated sphere".}$$

The equivalent extrapolated sphere is then converted to an equivalent real sphere by equating the radius of the equivalent sphere to the radius of the real sphere plus the extrapolation distance. The method of the above interpolation is being extended to the case of the partially filled, partially reflected spheres. Some experimental data will be needed to verify the methods.

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Control Rod Worth in Partially Filled Spheres

It was noted in the critical approach procedure of the experiments at the Plutonium Critical Mass Laboratory that upon withdrawal of the stainless steel control rod, the power first rose (as expected) and then went through a maximum and leveled off. The proposed explanation is that the worth of the rod as a volume displacement is initially (upon insertion directly into the solution) greater than the worth of the rod as a poison. To show this the rod worth is taken as the sum of two terms, one a volume displacement term and the other a flux-squared weighted poison term:

$$\frac{dk}{k} = (C_1 + C_2 \phi^2) dl.$$

To evaluate the constants, it is assumed that the one group theory is valid, then:

$$\frac{dk}{k} = -(\tau + L^2)(2/3)(4/3 \pi^4) \frac{dv}{v^2 B} = C(1 \text{ or } 2) dv$$

where for a Pu concentration of 33.8 gm/l:

$$(L^2 + \tau) \approx 31.8 \text{ cm}^2,$$

$$v^2 = (37 \text{ liters})^2, \text{ and}$$

$$B = (0.139) \text{ cm}^{-1}.$$

Finally

$$\frac{dk}{k} = -1.325 \times 10^{-5} (\text{per cc}) \phi^2 dv = -.00577 \phi^2 dl.$$

To evaluate the rod displacement, use is made of the same expression as before, except that the actual volume of the sphere is only 23 liters. The expression for rod worth becomes:

$$\frac{dk}{k} = +4.52 \times 10^{-5} (\text{per cm}) dl.$$

Adding the two expressions we have:

$$\frac{dk}{k} = (4.52 \times 10^{-5} - .00577 \phi^2) dl.$$

Integrating, this becomes (assuming $k = 1$ at $z = 0$):

$$k = \exp(4.52 \times 10^{-5} z - .00577 \int_0^z \phi^2 dz),$$

or since both quantities in the bracket are small:

$$k = 1 + (4.52 \times 10^{-5} z - .00577 \int_0^z \phi^2 dz).$$

The integral appearing in the above equation was carried out numerically. The results show a hump in the k versus z curve, whose peak value is 8.4 cents (.0022 = \$1.00) reactivity, and a net total rod worth of \$2.05. These values compare well with the measured values of a 10 ± 2 cent peak (obtained from the critical approach procedure), and \$2.29 net rod worth obtained from a rod drop experiment.

Subcritical Interactions

The numerical integration of the interaction probability function was completed for slab geometry. This code gives a theoretical albedo at each of the interacting surfaces. These albedos were used as input to a two-group HFN calculation to predict the eigenvalue for measured^(3,4) critical systems. Cross sections were taken from WAPD⁽⁵⁾ calculations, and fitted with ln-ln curves as a function of H/U-235 atomic ratio. A summary of the results follows:

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- (3) ORNL-2389, Annual Progress Report, September 1, 1957.
 - (4) ORNL-2609, Annual Progress Report, September 1, 1958.
 - (5) WAPD-TM-244, Nuclear Analysis of Thermal Reflected Cylindrical Homogeneous Critical Assemblies.

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H/U-235	Experimental Critical Conditions for Two Interacting Slabs			Calculated Parameters	
	Slab Thickness	Height Range	Separation Range	Albedo Range	k_{eff} Range
50.4	3"	14 - 28"	.5 - 3"	.99 - .93	.993 - 1.022
337	6"	15 - 31"	10 - 60"	.5 - .1	1.036 - 1.001
293(4)	6"	11.7-14.5"	6 - 12-1/6"	.6 - .4	1.007 - 1.029
254 PR*	6"	9 - 11"	5 - 30"	.6 - .12	1.034 - 1.015
325 PR*	6"	10 - 12"	5 - 25"	.6 - .17	1.05 - 1.028

* Partially reflected system. Noninteracting side reflected with 6 in. water.

The observed trend in k_{eff} , either consistently increasing or decreasing with increasing separation at a given H/U ratio, has not been explained.

The corresponding interaction function for cylindrical geometry was also programmed. In this case the neutrons entering one component from another must be reapportioned uniformly over the entire circumference since no code has a radially dependent albedo. The integration must be over four coordinates: one angle and one vertical distance for each cylinder. The vertical dependence for the absorbing slab was analytically integrated and three summations programmed. The eigenvalues for a set of experimental data using the average albedos from this calculation appear to be less than one percent over critical. However, normalization is still in doubt, hence, this value could reach 2 - 3%.

After the three summations were programmed, it was discovered that the emitting vertical coordinate could also be analytically integrated, if the flux were averaged. It is advantageous to use this revised method as it cuts machine time to $1/N$ where N is the number of vertical integration points. A program with this new feature has just recently been debugged and previous calculations are being redone. Comparison with a set of cylindrical interaction data from Russia will also be made.

Interaction of more than two systems has been studied briefly, but not enough experimental data are available to determine whether the present method will work sufficiently well.

Transport Theory Analysis of Plutonium Metal-Solution Systems

A transport theory analysis is being made of systems consisting of a large sphere of plutonium surrounded by a spherical annulus containing a Pu-H₂O solution. These calculations are needed to provide safe operating limits

for plutonium metal dissolution. The 9-ZOOM 19-group diffusion theory calculational procedure customarily used does not work for such systems because it does not adequately treat the plutonium low energy resonance absorptions.

The present calculations are being done by Program S-X. Because of the large flux gradient in the fuel and the great variation in the spectrum, from a fast spectrum in the interior to a thermal spectrum in the reflector, both a large number of energy groups and a large number of radial points are required. This combination so taxes computer capacity that it was necessary to remove some portions of Program S-X and to make a judicious choice of radial points to get the program into computer memory.

As a preliminary to the calculations actually desired, analysis is being made of a measured critical system of 5.6 kg of plutonium in 15 cm of water. Since this system should represent the maximum difficulties involved, its satisfactory solution should lend strong support to the validity of the Pu solution calculations. At present, the trial problem has not been run successfully and the sources of the difficulties are being traced. The uniqueness of these systems will provide future researchers in plutonium systems with some information on pitfalls of Program S-X.

Consulting Services on Nuclear Safety - Criticality Hazards

Nuclear Safety in HLO

The nuclear safety of a stainless steel pressure chamber was reviewed for Plutonium Fabrication Development.⁽⁶⁾ The chamber was approved for 4.5 Kg of plutonium metal.

The nuclear safety of handling and storing metallographic samples in 231-Z Building was reviewed for Plutonium Metallurgy Research.⁽⁷⁾ These samples consist of 1-50 g pieces of plutonium metal in 20 cc plastic blocks. The processing hood contains about 100 samples, and there may be several hundred samples in storage. Safe handling and storage limits were recommended, based on optimum conditions of moderation and reflection.

(6) Letter from C. L. Brown to R. R. King, Comments on the Nuclear Safety of Isostatic Pressure Chamber, March 8, 1962.

(7) Letter from C. L. Brown to D. D. Hayes, Comments on the Nuclear Safety of Handling Plutonium Metallographic Samples, March 2, 1962.

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Nuclear Safety in Transportation

Shipments of fissile material which contain less than 10% of a minimum critical mass do not require nuclear safety review and approval prior to shipping from Hanford. The frequency of such shipments, however, must be controlled to help assure that an unsafe quantity of material will not be accumulated in a warehouse or transfer terminal. As a guide in making shipments, limits that represent 10% of a minimum critical mass were submitted to Nuclear Materials Operation.⁽⁸⁾ Limits of the frequency of shipments were also given.

Nuclear Safety - Training and Education

The criticality display was shown at the Benson Polytechnical School Science Fair in Portland, Oregon, on March 3-4. For the remainder of the month, the display was shown at the Redox and Purex Plants in CPD.

Off-Site Consulting

a) Nuclear Safety Review of APED Facilities at San Jose and Vallecitos

A nuclear safety review was made of APED facilities at San Jose and Vallecitos during March 12-22, 1962. The administrative procedures used to establish critical mass controls were reviewed, and the facilities at San Jose and Vallecitos were inspected. The nuclear safety aspects of licensing were also reviewed. A contract was arranged between APED and the AEC to pay for these services.

b) Nuclear Safety Review of Fuel Element Reprocessing Plant for CNEN

A criticality review was completed of the preliminary designs for an enriched uranium reprocessing plant for CNEN (Italian National Commission for Nuclear Energy).

The plant as designed by the Vitro Engineering Company, New York, is "limited safe" from criticality. Some of the process vessels are safe by geometry, for all possible concentrations, while others are safe on the basis of the U^{235} concentration in the solution and the mass of U^{235} being processed.

In those areas where administrative controls are required, nuclear safety has been conservatively based on a "two contingency" criterion in which at least two simultaneous and independent operational errors or process malfunctions must occur before criticality is theoretically

(8) Letter from P. F. Gast to F. J. Zelle, Memorandum on Transportation Nuclear Safety - Approval Requirements and Shipping Frequency, March 8, 1962.

possible--such as double batching together with simultaneous excessive over-concentration or precipitation of U^{235} from solution.

The preliminary designs for the plant as reviewed on March 16, and the methods of criticality control being used appear to be in conformity with good nuclear safety practice. (The nuclear safety of the centrifuge could not be evaluated since the preliminary design for this unit had not been fully established as of March 16).

With suitable written procedures for administrative control, it should be entirely feasible to operate the proposed plant in a nuclearly safe manner at all times with the MIR-type U-Al alloy fuel elements being proposed.

Mass Spectrometry Studies in Support of Chemical Processing

The mass spectrometer for this program provided analyses for the Isotopic Analysis Program during the month. In addition, a series of uranium standards were isotopically analyzed to provide a calibration of the spectrometer detection system for uranium analyses. The results of these standards analyses verified the existence of systematic effects and instabilities in the performance of this spectrometer system. A statistical analysis of the standards data infers the following about the present performance of this spectrometer: 1) A ratio dependent bias exists which varies from -1.60% to 2.00% for U^{235}/U^{238} ratios from 0.01 to 10 respectively with a 95% confidence interval of about $\pm 0.4\%$ bias, 2) the mass discrimination in the U^{235}/U^{238} measurement is $0.29\% \pm 0.33\%$ at the 95% confidence interval, and 3) the variability of results of repeated analyses which is greater than the precision of a single analysis limits the 95% confidence interval for a single analysis to 1.70% after the bias correction is made.

Instrumentation and Systems Studies

The "C" Column Simulation consists of an attempt to evaluate several tentative mathematical models of a pulse column used in chemical separations. The models were developed by Chemical Research and Development. A new analog circuit was designed to provide for a greater range of variation of the two unknown constants and to introduce the capability of raising one of the problem variables to a non-integral power. Some twenty experimental runs are to be evaluated. The portion of this study requiring the use of the analog computer will be performed in April because no computer time was available in March.

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Work on the first model of the scintillation type log n-period meter for the Critical Mass Laboratory is nearing completion. The present instrument utilizes tube type operational amplifiers with chopper stabilization where necessary. The final instrument may be transistorized. Some degree of reliability may be gained at the expense of accuracy by use of a passive RC integrating circuit. Lack of a simple calibration procedure for the present instrument may be a problem. The only method used to date consists of varying the current to a lamp used to excite the scintillation phototube. Other problems involve the lack of a simple method of electrically varying the phototube high voltage.

An effort has been made to determine the form and amplitude of the line voltage fluctuations at the Critical Mass Lab. An instrument developed by Equipment and Instrument Development, CPD, was used in an attempt to detect fast transients. No fast transients were found.

The control rod drive assembly for use at the Critical Mass Laboratory was tested. The system, because of inertia, will require a minor redesign and modification to provide the rod-drop velocity and appropriate selsyn readout required.

NEUTRON CROSS SECTION PROGRAM

Quasi-Elastic Neutron Scattering from Water

Analysis of previously obtained data on the quasi-elastic component of the differential scattering of monoenergetic neutrons from room temperature water continued during the month. Data taken at neutron energies of 0.1 ev and 0.25 ev were processed through the INELASCAT data reduction program. The least-squares fitting of the reduced data has now been completed for nearly all of the measurements.

Inelastic Scattering of Neutrons from Water

Measurements were made to determine the effects of sample container window thickness for high temperature, high pressure water samples.

Analysis was completed of measurements to determine the correction due to half-order reflected neutrons from the Be $\overline{0002}$ reflections. The results of this analysis showed that the corrections to the scattered neutron intensity are less than 10 percent for initial neutron energies of 0.15 to 0.4 ev.

Analysis of the scattering measurements from a vanadium sample at liquid nitrogen temperatures is in progress. A transport theory code, Program S,

is being used to attempt to correct the measurements for finite sample thickness. The results of these measurements are used to determine the over-all efficiency of the analyzing spectrometer as a function of energy and angle.

Fast Neutron Cross Sections

The analysis of the previously obtained total cross section data has been continued. Methods of correcting the data for the shift of the zero point on the time scale have been investigated. Calculations have been made to determine the effects of in-scattering of neutrons which show these effects to be negligible for the geometry used in the measurements. Reports on the results of the first measurements and of the experimental techniques are in progress.

REACTOR DEVELOPMENT - O4 PROGRAM

PLUTONIUM RECYCLE

Fast-Thermal Reactor Complexes: "Fuel Reuse"

The "fuel reuse" concept is part of a more general program of investigations regarding the utility of fast-thermal reactor complexes to achieve optimum fuel utilization, and, hence, minimum fuel costs.

In fuel reuse, the breeding blanket of a fast reactor is used as a high quality (low in Pu-240) enrichment facility. Fuel elements fabricated for the thermal reactor are enriched in the fast reactor blanket before insertion into the thermal reactor. If desired, irradiated fuel elements from the thermal reactor are re-enriched in the fast blanket and then re-inserted into the thermal reactor.

The work on the fuel reuse concepts is being carried out in a joint effort with HLO's Programming Operation. In the present analysis a typical fast reactor oxide blanket is used as the enrichment facility and a D₂O machine is considered to be the companion thermal facility. Using the approximate relation for plutonium concentration: $N_{40}/N_{49} \sim N_{49}/N_{28}$

(which holds over a wide range of exposures in the fast blanket) a series of fuel compositions for insertion into the D₂O machine was constructed. Provision for residual U-235 and the presence of fission products (accumulated during the fast blanket irradiation) in the fuel was made. These fuel compositions were then burnt in the D₂O machine, employing a quasi-graded cycle. The burnup calculations were performed by means of HLO's "MELEAGER" burnup code. The MELEAGER output information was then processed to obtain comparative fuel costs. For the economics analysis HLO's "QUICK"

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code was employed. A range of jacketing and separations cost from 0 \$/# to 40 \$/# was assumed. The results indicate that for single pass (i.e., no re-enrichment) fuel reuse, fuel costs might be reduced by .3 mills/kwhe to possibly as much as 1 mill/kwhe depending on the circumstances involved. Greater fuel cost savings might be achievable by means of multiple fuel reuse.

At the present time, work on multiple fuel reuse is under way for the same D₂O Reactor - Fast Blanket complex. In future work, this analysis will be extended to other thermal reactor types and to different modes of reactor operation.

Ceramic Reactors for Space Application

A nuclear parameter survey of ceramic reactors for space application is being carried out. Propulsion units - rather than stationary power units - have the greatest potential for possible weight reduction, since the engine weight in these devices represents a substantial fraction of the entire machine weight. The present survey has, therefore, been directed toward reactor configurations for propulsion units.

Calculations were performed on highly idealized reactor models. The reactor cores were assumed to be spherical and to consist only of fuel carbide-ZrC elements and of void. (The core void represents the gaseous coolant.) Fuel element materials examined were U-235 C - ZrC, U-233 C - ZrC, and PuC - ZrC. All cores have a 10 cm graphite reflector.

Total reactor weights (core and reflector) have been obtained as a function of flow area. Flow area has been selected as the significant parameter because it is indicative of the power capability of the unit.

The results indicate a number of interesting facts:

- (1) Curves of reactor weight vs. enrichment (for a fixed flow area) show rather pronounced "saturation" characteristics. Up to ~40% enrichment (i.e., 40 v/o fuel carbide, 60 v/o ZrC) the total reactor weight drops rapidly. Beyond that point, the weight drops more gradually with increasing enrichment.
- (2) Pu-239 C and U-233 C fuel materials are very similar as far as their effect on reactor weight is concerned. For very low enrichments, the Pu-239 C reactors are somewhat heavier, while for very high enrichments, the U-233 C reactors are heavier by a very small margin.

- (3) Over the range of parameters examined the Pu-239 C reactors show weight savings of ~20 to 35% as compared to the U-235 C reactors.

All calculations were performed by means of the HFN multi-group diffusion code. Cross sections were taken from the Yiftah, Okrent, Moldauer (YOM) fast cross section compilation. To test the validity of diffusion theory, it is planned to carry out some transport calculations for a selected number of reactors. Cross section sets other than the "YOM" set will also be employed to obtain a more realistic treatment of reactors with a more degraded spectrum. In future calculations, the effect of structural materials, different types and thicknesses of reflectors and Pu-240 contamination of Pu-239 will also be considered.

Parametric Survey of Fast Oxide Breeders

Critical loadings have been found for the fast oxide breeder reactors mentioned last month. The twenty reactors surveyed were fueled with both U-235 and plutonium with isotopic abundance 68.17% U-239, 19.2% Pu-240, 10.4% Pu-241, and 2.3% Pu-242. The HFN code was used to find the critical loads, and the CLERK JAYNE code was used to find conversion ratios and power splits between core and blanket. Analysis of the data has not yet begun.

Status of FTR Fuel Irradiation Experiments

a. Low Exposure Plutonium-Aluminum Elements

The desired exposures on the six L_x Pu-Al physics test elements in ring 5 are equal increments of 14 MWD. The first three elements, installed on December 3, 1961, now have somewhat over 30 MWD each. Of the last three, installed on January 17, 1962, element 5051 exposed in channel 1247 received 13.1 MWD, and was removed on February 2, 1962. At the next shutdown, element 5052 will have received about 28 MWD and will be removed. The last element should reach 84 MWD about November, 1962.

b. Mark I UO₂ Elements

Six Mark I UO₂ elements containing 1/2 w/o Co in the Zircaloy wrapping wire for flux monitoring purposes should be ready to install in ring 1 at the next shutdown. Goal exposures are equal increments of 41 MWD, and the last element will require about 16 months in the reactor.

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c. High Exposure Plutonium-Aluminum Elements

Since it appears that there will be no H_x Pu-Al zone to provide an environment for the six physics test elements, the feasibility of matching an H_x spike to a moxtyl (mixed oxide crystal) environment in their absorption and fission rates is being considered. A loading of roughly 600 g Pu per element with Pu-240 content between 25 and 30 percent would be required to match the clean elements. Since the test elements receive batch exposure while the environment is a graded (approximately constant) environment, a match would occur at only one point in the exposure.

d. Flux Monitors

To obtain spectral data (r and T) for a Westcott flux description, the activity of bare and cadmium covered flux monitors is needed. From an analysis standpoint, the instantaneous rather than the integrated spectral data are the more useful. To obtain brief exposures of flux monitors on partially burned fuel elements is one goal, requiring the ability to attach and detach foils under water in the PRTR basin. As a first step in approaching this problem, during the next shutdown it is planned to attach bare and cadmium covered lutetium foils in aluminum holders to two clean L_x Pu-Al elements for a 30-minute irradiation at minimum power in a ring 5 location.

e. Gamma Scanning Facility

A scope drawing was forwarded to PRTR Technical Planning Operation showing the Gamma Scanning Facility to be installed in the Fuel Examination Facility. Gamma scanning an assembled element, individual rods, and monitor wires is planned, using the existing fuel element handling equipment and manipulators together with special holders for rods and wires.

f. Radiochemical and Mass Spectrometer Work

Element 5075, which had received 49.5 MWD exposure before the end bracket came off, precluding further irradiation, has been cut up by Radiometallurgy and 13 one-inch samples have been dissolved. To date five of these have been analyzed by Analytical Laboratories for plutonium content by coulometric titration, and for Cs-137 beta activity. Further analyses will include mass spectrometry for plutonium isotopic composition, plutonium alpha counting, and possibly Sr-90 gamma activity. In the absence of flux monitor data the internal consistency of the analyses will be used to evaluate the accuracy of the methods.

PRTR Experiment-Theory Correlation

A study is under way to determine the precision of measurement of experimental conditions which is required in order to deduce basic physical constants of plutonium fuels within a useful limit of error. Using the MELEAGER computer code, the effect of parametric variations on sensible reactor conditions such as isotopic burnup rates and system multiplication is found. Comparison of conditions during burnup cycles for a given fuel in the PRTR was made between cases with different initial conditions of spectral hardness, neutron temperature, flux magnitude, and changed restrictions on spectral hardness during burnup. The fuels considered were L_x Pu-Al, natural UO_2 , H_x Pu-Al, and PuO_2-UO_2 . Further variations will include changes in the neutron cross sections of the plutonium isotopes.

Deduced from the results (in a sense which is reverse to the machinations of MELEAGER) will be errors to be associated with cross sections calculated from measured data on integrated exposure of fuel, in-reactor spectra, isotopic analysis of burned fuel, and multiplication perturbing effect of fuel during exposure, etc.

PRTR "Phoenix" Fuel Experiment

Spectral hardening is being studied by adding poison and enriched uranium in the central cell of the PRTR. A report is presently being prepared to explain the methods and procedures followed in this investigation, and to present the major results of the study. Results have been obtained of "power ratio" of the average power in the test rod to the average total reactor power vs. "U-235 enrichment" in the eighteen surrounding rods of the 19-rod cluster in the central cell of the PRTR and of "Westcott r" and " Δk_{eff} " vs. "U-235 enrichment." The absorption cross section of the process tube is used as the parameter in all of the calculations for additional hardening of the spectrum, and for reducing the power in the U-235 enriched rods. The required absorption could be added in the fuel rod cladding rather than the process tube. The test rod which is the central rod in the central cell cluster is 1/8 w/o Pu-Al in the initial studies.

It is now proposed to run a burnout calculation with the MELEAGER code; the output of MELEAGER will then be used in the SWAP code to determine the changes in k_{eff} and Westcott r with burnout.

Code Development

a. MELEAGER

The MELEAGER code has been modified to make use of self-shielded

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burnable poisons. Any isotope may be self-shielded such that the ratio of the average thermal flux within the poison lump to the average flux in the fuel is given by $f(\phi t)$, a flux time dependent quantity

$$f(\phi t) = \frac{1}{1 + \gamma_1 P + \gamma_2 P_2}$$

where γ = constant dependent on the geometry of the poison lump.

$$P = N(\phi t)/N(o)$$

$N(\phi t)$ = isotopic concentration of poison at time ϕt

$N(o)$ = isotopic concentration of poison at time zero

The self-shielding may be applied up to four isotopes simultaneously. If two γ 's are to be applied to the same poison, that isotope must appear twice in the isotope list.

A special MELEAGER deck was made for Programming Operation to provide external control on the calculation of the spectral index r . The term to be modified is β where

$$\beta = \gamma/(1 - br)$$

$$b = \text{constant}$$

β' will now be calculated from the old β then in turn be used to calculate a new γ .

$$\beta' = \beta - K \beta^2$$

$$K = \text{input constant}$$

This form of β' appears to bring MELEAGER into agreement with SPECTRUM results for hard spectra. This MELEAGER deck is for exploratory work only and does not replace the decks now in use.

b. Modified MELEAGER

The first version of ANNE, the ex-MELEAGER code for calculation of average multi-group fluxes in a cylindrically, symmetric, two-region, lattice cell was coded and debugged. A one-group calculation, which took 2 ± 1 milliseconds, yields average fluxes in excellent (four to

five figure) agreement with HFN. The HFN run could not be accurately timed. The theoretical expressions for the average fluxes in a slab cell have also been derived and are being coded.

The changed portions of MELEAGER, which are designed essentially to compute the neutron spectrum as a function of fuel burnup will then run as follows:

At the beginning of a burnup analysis on a given system, a slightly modified version of GAM will compute eight-group epithermal average microscopic cross sections for all isotopes to be used based on a homogenized infinite medium spectrum of some average composition. Control will then pass to TEMPEST, which will compute average microscopic cross sections for a thermal group under similar conditions. As they are computed, these average cross sections will be saved on tape for intermediate analysis by a program which will use these average cross sections to prepare a data tape for MELEAGER. (Different versions of this code could prepare data for other programs, such as S-X or HFN.) Control will then go directly to MELEAGER, for the burnup analysis.

At each step in the burnup analysis, the macroscopic cross sections corresponding to the current composition of the fuel region will be computed, and a nine-group cell analysis performed. The cell code, which will be based on experience gained with ANNE, will compute nine-group average fluxes which will be used, with the nine-group microscopic cross sections to compute one group average microscopic cross sections for each isotope in the fuel. These one-group cross sections will replace the Westcott-type cross sections presently used for burnup calculations.

The nine-group energy structure will probably be:

<u>Group</u>	<u>Approximate Energy Range (ev)</u>	<u>Characteristics</u>
1	$10^6 - \infty$	most fast fission in this group
2	$10^5 - 10^6$	fast fission rises from zero
3	$10^4 - 10^5$	unresolved resonance region
4	$10^3 - 10^4$	sodium resonance; resonance capture
5	$10^2 - 10^3$	resonance capture in most elements
6	10 - 100	resonance capture in many elements
7	2 - 10	Pu-239, Pu-241 resonances
8	.65 - 2	Sm, Pu-240 resonances
9	0 - .65	Sm "thermal" group

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c. Program SHUSH

Program SHUSH, Spherical Harmonics Using the Standard HFN, was used to calculate the flux as a function of position near the boundary between a critical half-space and a void, or black, half-space (Milne problem). The SHUSH-HFN fluxes were used to compute the extrapolation distance, using each of the seven available approximations. Results were satisfactory in all except the P_1 (diffusion theory) and double- P_2 calculations. The P_1 calculation went awry because, in running the calculations, the void was mocked up by a non-scattering absorbing medium. This system, necessitated by limitations in HFN, does not behave the same as a void when analyzed using the diffusion equation. When the normal exterior boundary conditions are used (albedo equals zero), HFN gives the usual diffusion theory result that the extrapolation distance is $2/3$ the mean free path. The DP-2 equations in SHUSH are apparently wrong. Since they have been checked against the FLIP document, and no differences found, the document may contain error. The expected light usage of this particular approximation does not warrant further investigation of the error at this time.

The higher order approximations to the transport equation are computed by HFN, by using several "groups" to compute the flux at each energy, one-group being required for every two flux moments. The number of "groups" required when using each of the approximations to analyze one energy is tabulated below, as well as the corresponding extrapolation distances (in units of the mean free path). The latter values compare favorably with the exact value of .7104. The approximate HFN running times are also tabulated:

<u>Approximation</u>	<u>No. "Groups"</u>	<u>Extrap. Dist.</u> <u>(m.f.p.)</u>	<u>HFN</u> <u>Running Time</u> <u>(sec)</u>
P-1	1	.5778	20
P-3	2	.6946	40
P-5	3	.7043	80
P-7	4	.7073	130
DP-1	2	.7119	40
DP-2	3	.3580	200
DP-3	4	.7109	110

The equations used in SHUSH assume a slab geometry. A few trial calculations indicate that they might yield good results in cylindrically symmetric systems which do not contain voids. The trial calculations, all of which contained either voids or low density regions, demonstrated

that the HFN assumption of constant flux across a cylindrical void is inferior to the diffusion theory condition that the current is inversely proportional to the radius in cm annular void. Possibly because of the constant flux assumption, the trial calculations (using P-7 theory) did not compare well with results from Program S-X.

d. RBU

The flow charting of the Monte Carlo portion has been 85 percent completed. Process of verifying the logic structure using the flow charts has been started with no errors yet detected.

The formulation and a schematic method of accounting for the crystalline and molecular binding effect at low neutron energies have been completed and are ready to be inserted into the Monte Carlo collision routine.

Lattice Parameters for Low Exposure Pu-Al Fuel

The following reports have been revised for publication in Physics Research Quarterly Report for January, February, March, 1962: "The Fuel Temperature Coefficient of k_0 for a Pu-Al Fuel", "Experimental Studies in Low Exposure Pu-Al Lattices", and "An Investigation of the Effective Neutron Temperatures in Pu-Al Lattices".

Cell calculations for the Pu-Al fueled, graphite-moderated lattices have been resumed. Calculated absorption rates are to be compared with the experimental results in the above reports. A previous attempt to generate multigroup fluxes in a lattice cell using the computer code, $S_n^{(9)}$, was unsuccessful, resulting in negative fluxes in some groups. This was probably due to faulty input data. A new set of input parameters has been obtained using the computer code⁽¹⁰⁾, GAM-1. These new data are currently being used as input to S_n in another attempt to obtain realistic results.

(9) Duane, B. G., Neutron and Photon Transport - Plane Cylinder Sphere Program S Variational Optimum Formulation, XDC-59-9-118. Jan. 9, 1959.

(10) Joanou, G. D. and Dudek, J. S., GAM-1: A Consistent F1 Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants, GA-1850. June 28, 1961.

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Plutonium Recycle Critical Facility

Preliminary approvals have been obtained for the physics experiments which will be conducted at the startup of the PRCF. The experiments are described in HW-71214.

Input data for a kinetics code (RE-135) are being assembled to study the neutron kinetics of the PRCF for additions of anti-reactivity. The results of the calculations will be used in the analysis of rod-drop experiments which are to be conducted during the startup experiments.

Modifications of the PRCF which will be necessary to conduct H_2O experiments have been discussed with personnel of TR & AO and REDO. The light water experiments are scheduled to begin approximately eight months after the startup with D_2O .

The safety rods which have been fabricated are aluminum tubes which have been plated with cadmium. In order to measure the thickness of the cadmium plate as well as its uniformity as a function of position on the rod, a "thickness gauge" has been set up which utilizes a NaI counter and an Am^{241} gamma-ray source. Initial studies to perfect the measurement were conducted with a 256 channel analyzer. The counter which is in use as the gauge, however, is a single channel analyzer which is biased to count only the 60 kev gamma ray which occurs in the decay of Am^{241} . The thickness gauge was calibrated by successively adding 5 mil layers of cadmium to an aluminum tube. The transmission of the 60 kev gamma rays, from the source positioned inside the rod, were measured. The 1/4" thick wall of the aluminum tube reduced the intensity to 0.70 of that incident on the wall of the tube. The cadmium sheets reduced the intensity by 0.84 per mil of cadmium placed between the source and the counter. Such sensitivity is adequate for measuring the thickness of the cadmium to ± 0.001 ".

An evaluation of the reactivity effect due to placement of H_2O in a neutron-reflecting position has been completed. Of principal concern is an accidental flooding of the PRCF cell by H_2O from the PRTR load-out canal. Possible reactivity increases vary with the fuel configuration in the core, and can be nearly eliminated if the integral of the statistical weight for reflector additions outside the tank is reduced to a small value. Three-group diffusion theory calculations have shown that the presence of a 30 mil liner of cadmium outside the tank is sufficient to isolate the well-thermalized PRCF core from outside reflector reactions. The eigenvalue for a typical bare core was 1.0 mk smaller than the eigenvalue for the same system surrounded by the cadmium liner and 9.0 cm of H_2O .

Neutron Spectrum Studies

The November 1961 monthly report gave characteristic parameters for the relaxation length of neutron temperatures for a two thermal group model and a continuous slowing-down model. The two-group parameter was the relaxation length for rethermalization, which has been obtained experimentally. The continuous model parameter was the square root of the age for a specified change in equilibrium temperature. The age was calculated from the time rate of change of equilibrium temperature reported by Kothari [NSE, 7,240-244, (1960)]. The comparisons were within 30%. A similar comparison has been made using calculations by C. W. Lindermeier. The agreement was within 10% to 16%.

Although the fairly good agreement obtained in these comparisons is encouraging the significance is not understood. The basic problem is the validity of the comparison itself. Should $\sqrt{\tau}$ in the continuous model equal L in the two group model? At present it is believed that it would be better to make detailed comparisons of the spatial distributions of effective temperatures of neutrons using the two models.

Critical Mass Studies for 1.8 w/o Pu-Al Fuel

Experiments to determine nuclear parameters for 1.8 w/o Pu-Al fuel are being conducted in the tank in the TTR reactor room. The experiments will consist of approach to critical and exponential measurements with lattice spacing of 0.75, 0.80, 0.85, 0.90 and 0.95 inches. The unclad fuel rods are 44 inches long and .506 inches in diameter. They are clad with .030 inches of zirconium.

The approaches to critical using the 0.75, 0.85, 0.90 and 0.95 inch lattice have been completed. Table I gives the critical loadings which were indicated by extrapolation of the inverse multiplication curves. The highest multiplications occurred for loadings of 95% of the critical masses. These results may be modified to some extent when the data are analyzed by the least squares method.

TABLE I

<u>Lattice Spacing in Inches</u>	<u>Critical Mass</u>	
	<u>In Rods</u>	<u>In Grams of Pu</u>
0.75	563	4009
0.85	493	3510
0.90	512	3645
0.95	583	4151

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These measurements will help establish nuclear safety specifications for PRTR fuel reprocessing, and will supplement similar measurements to be made later in the PRCF.

Instrumentation and Systems Studies

Detailed drawing preparation was started for the collimator and detector shield of the Applied Physics fuel rod and activated wire scanning facility. Continued technical assistance was provided and shielding calculations were made.

Prototype fabrication was nearly completed on the transistorized electronics chassis for the PRTR Liquid Effluent Monitor. Tests were made on each module as it was fabricated and all have been satisfactory. Experiments were performed on various phototubes to be used with the terphenyl-in-polyvinyl-toluene detector. An EMI-9536-B phototube showed excellent stability for an extended test period with counting rates typical of those expected at the highest alarm levels. The EMI tube will be used in the probe. A reliable, transistorized fail-safe circuit was developed to provide the high-level trip with backup protection. It will prevent possible blocking of the AC coupled circuits for the remote possibility of extremely high counting rates. The circuit will also energize a suitable alarm if the power supply fails or if the 300-foot-long connecting cable breaks. The contacting meters, which are to be used for alarm activation, have shown a tendency to cause noise spikes on the chart recorder. The instrument was designed to monitor liquid effluent fission product concentrations from 10^{-5} $\mu\text{c/cc}$ to 10^{-1} $\mu\text{c/cc}$ with both low and high level selectable alarm trips provided.

Continued support was rendered on the design of the shielding for the Fuels Recycle Pilot Plant Facility.

Calculations were made and presented for the design of shielding rings for the top shield entry-way of the proposed PRTR rupture loop.

Tests performed in connection with PRTR Test #19 indicate that the control interactions between the degasser and pressurizer level systems are not strong enough to account for the observed level fluctuations. Further tests are planned as soon as suitable test equipment is received.

Further tests on the moderator level systems are being planned for the next PRTR startup and operating period. Simultaneous recordings of two different moderator level transducers and two different flux monitors will be obtained during low level operation. This data should yield improved estimates of the effective neutron lifetime and an insight into the nature of the moderator level fluctuations.

Physical Metallurgy reviewed the developmental microwave linear displacement transducer for possible use in in-reactor metallurgy creep experiments. It doesn't appear practical to use the microwave system with the present creep capsules because of the investment in variable permeance transducer instrumentation which satisfies present experimental needs. However, the microwave system may offer definite advantages for future capsules in higher flux environments which generate more gamma heating and higher capsule temperatures. No further development of the microwave transducer is planned until a specific design is needed for a particular application. It is estimated that the final development, design, and fabrication of a microwave system for an in-reactor creep measurement would cost about \$25,000. The in-reactor equipment for each additional creep measurement experiment would cost about \$2000.

The four-transducer, ultrasonic resonance probe assembly for measuring the wall thickness of the PRTR process tubes is in the final stage of operational assembly. Considerable difficulty was experienced this month in fabricating suitable leaf springs for the probe. The problem was finally solved with two 0.010-inch strips of Elgin clock spring which has been heat treated, copper flashed, and chrome plated, in each of the eight locations.

NEUTRON FLUX MONITORS

Review was nearly completed on the feasibility study formal report concerning various long-life plutonium and uranium in-core flux monitors of the Phoenix or regenerating types. All calculations utilized the well-known Westcott cross section approximations to account for effects of epithermal neutrons, thermal neutron temperature, flux level, and integrated exposure on the burnout and buildup of the detector fissile radionuclides. For certain assumed conditions, optimal detector compositions were determined, using computer codes, which yield the best performance for the greatest length of time.

Difficulties are still present in obtaining a suitable base material for the experimental plutonium-uranium foils which are to be irradiated to evaluate the effectiveness of the calculated compositions. At present, the best base material appears to be pyrolytic graphite; however, flaking remains a problem to be solved. A general experimental program has been outlined and will be started as soon as suitable foils are available.

Refinements were made to the preliminary calculations reported last month on the possible uses of Nuclear Magnetic Resonance (NMR), Electron Paramagnetic Resonance (EPR), and Microwave Spectroscopy (MS) for in-core neutron flux measurements at high temperature and high flux. Consideration was given to a system in which a gas (such as freon) is piped into the reactor,

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irradiated, and then piped out for analysis for neutron-induced changes. Both NMR and EPR now appear to be impractical methods for this analysis. In a sample calculation for a system with a gas recirculation period of ten seconds, the flux required for a signal-to-noise ratio of one would be at least 6×10^{18} nv for NMR and 2×10^{16} nv for EPR. However, under the same conditions, it appears that a flux as low as 5×10^{12} nv could be measured to ten percent accuracy with MS using the $Cl^{35}(n,\gamma)Cl^{36}$ reaction. This makes the MS method an attractive candidate for further consideration along with the microwave noise analysis technique. It is hoped that equipment deliveries will permit some experimental tests of the microwave methods to be started during the next month.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

A study of factors affecting the separation of parameters in the multiple parameter eddy current test equipment was made. Tentative conclusions based on this study are that the condition required to obtain good separation of three and four parameters are not met in the present experimental arrangement. Changes in the system based on this new study are expected to reduce undesirable inter-channel effects observed when attempting to separate three and four parameters.

A generator was developed for providing a composite test coil drive signal comprised of three rising exponentials for use with the orthogonal exponential analyzer multiple parameter test system.

The parameter separation study included an error analysis to show the relative effect of calibration and instrument errors, and an analysis of the output signals of the test specimen equivalent circuit in relation to the requirements of the system for good separation of parameters. This study has not been completed, but first results show that the choice of test frequencies and present values of parameter test coil load resistors give an unfavorable condition with respect to separation of parameters. This approach has shown the need to relate in detail typical output signals due to test specimen irregularities of interest to those provided by the equivalent circuit. The new study is not complete, but tentative conclusions based on it will be tested with appropriate changes and measurements in the test system.

Several vacuum tube generator circuits for producing rising exponential waveforms were breadboarded and checked for use with the orthogonal exponential analyzer. These circuits are unstable due to the nature of the function being generated, and require periodic gating or quenching to return

the circuits to definite starting conditions. Both plate and grid circuit gating methods were tried, and best results were obtained with grid circuit gating. The final circuit consisted of three of these circuits having adjustable rates of rise feeding a summing amplifier.

Heat Transfer Testing

Two radiometers for the emissivity-independent heat transfer testing system are nearing completion. Several proposed circuits for continuously giving the ratio of signals from the two radiometers were shown to have serious limitations. Systems Research Operation specialists have been contacted for assistance, and some of their engineering and computer time will be available for developing a suitable ratio circuit.

Components necessary for adaption of Systems Research Operation's Ampex variable time delay unit to signals from the radiometer have been obtained.

X-Ray Stress Analysis

X-ray stress analysis data have been submitted to Operations Research and Synthesis to determine if the small difference between data from pipes annealed one-half hour at 800°F, and from those annealed at 1325°F, is significant.

An invention report, describing the use of X-ray diffraction peaks from sand-blasted metals to produce maximum temperature indicators, has been issued (HWIR-1498). The broad X-ray diffraction peaks from sandblasted metal surfaces return to the sharp peaks of non-deformed metal at the stress relief temperature. This effect is useful, since annealing time is a much less important factor than temperature in showing the sharpness of diffraction peaks from the sand-blasted surfaces. The diffraction peak from one sand blasted surface stress-relieved one-half hour at 1000°F last month was much sharper than expected. Three additional samples were stress relieved under the same conditions. Diffraction peaks from these samples were broad, as expected. Apparently there was an error in the recorded temperature in the furnace used to stress relieve the original sample.

Eddy Current Analysis of Carbon Steel Reactor Piping

Eddy current methods were investigated for classifying carbon steel piping according to the degree of residual stress relief, and for identifying incipient failures in the seam-welded regions of side-bend test specimens of carbon steel process (NPR) piping.

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Piping samples (4-1/2" OD x 6" long) which had been heat treated at 825°F and 1325°F, respectively, were tested. With a specially designed probe in conjunction with a General Electric Metals Comparator, a marked difference in scale reading could be obtained between samples anywhere around their periphery. Samples of smaller pipe (2-5/8" OD x 4" long) which had been heat treated at 825°F (untreated), 1125°F (stress relieved), and 1325°F (annealed) were also tested with the same results, except for one axial strip 1-1/2 inches wide on the stress relieved sample, which produced a consistently ambiguous indication. This sample is undergoing metallographic and chemical analysis in an effort to determine the reason for its characteristics. At this time it seems possible that a test can be developed to provide a negative full scale indication for insufficiently heat treated pipe, a mid-scale reading for properly heat treated pipe, and a positive full scale reading for pipe which has been over heat treated. Twelve additional samples are being prepared to confirm the validity of this test.

Forty-eight side bend specimens were tested with the same equipment in an attempt to detect incipient weld failure. Comparing these data against visual observations made after the specimens were bent resulted in only partial correlation. Minor changes in testing procedure and specimen preparation will be made in an effort to improve the correlation of results between the two tests.

Zirconium Hydride Detection

Equipment has been set up to make measurements of ultrasonic attenuation in solids by the pulse technique. Preliminary investigations have been made to determine how to couple the quartz transducer to the Zircaloy-2 sample. It has been found in the literature that direct mounting, with salol (phenyl salicylate) used as a coupling media, has given repeatable results to within a few percent on remounting the quartz transducer. It has also been found that due to the small size of the Zircaloy-2 samples, there may be problems in measuring the attenuation, due to interference between longitudinal and shear waves.

Four additional Zircaloy-2 pipe samples having small areas hydrided to 50, 100, 500, and 1000 ppm have been ordered by Testing Methods Operation.

In an effort to detect local hydriding in Zircaloy-2, eddy current testing techniques are being applied to measure the Hall coefficient and resistivity of a sample with a single test probe. Work has continued toward optimizing the test sensitivity to changes in the Hall coefficient. To increase the stability of the test and improve the balance point control, a 100 kc crystal-controlled oscillator and an improved balance circuit containing both coarse and vernier controls, have been incorporated into the test system.

The tests are still being conducted primarily on bismuth, since test sensitivity is still not sufficient to provide interpretable results with either Zircaloy-2 or copper--materials with Hall coefficients about 10^{-5} that of bismuth. To gain a better understanding of the variables associated with the Hall phenomenon, a literature search is being conducted to compile the published results of recent investigations in this field. Efforts to locate a comprehensive compilation of this information for a large number of elements and materials have thus far been unsuccessful. In view of this finding, data on materials of engineering interest are now being accumulated. These data will be classified by measurement parameters, converted into a consistent set of units, and tabulated for possible publication.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing Test Angle

The response of a line-focused transducer to a 2-mil-deep notch in a flat Zircaloy plate was measured at 0.5° intervals for ultrasound incidence angles from 13.5° through 30.0° to the plate normal. Peak responses were experimentally observed at the angles tabulated below for two methods of instrument tuning. These are compared with the calculated angles for Lamb wave modes in zirconium for this particular case of a plate depth (D) of 0.0385 inch and nominal transducer frequency (f) of 10 Mc. Attenuation measurements for each observed peak response are in progress.

<u>Transducer & Cable Tuned for Maximum Response for Each Measurement</u>	<u>Transducer & Cable Tuned for Maximum Response at 20.5 Degrees</u>	<u>Theoretical Angles from Zirconium Lamb Mode Curves</u>
14.5 $^\circ$	14.0 $^\circ$	14.00
15.5	15.5	15.5
-	*	17.5
19.0	*	18.5
20.5	20.5	21.5
23.5	23.5	**
-	26.0	25.5
28.5	28.5	29.5

* The velocities of the modes at 17.9 and 18.5 degrees are close to the longitudinal velocity and were not discernible in the data when tuned only at 20.5 degrees.

** Theoretical curves do not show a mode at this point. However, it was experimentally verified independently by two observers.

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Instrument Tuning

When tuned for maximum amplitude at each angle in the tests tabulated above, two peaks could be tuned within the frequency adjustment range of the inductance coil (0.9 - 1.6 μ henries) at some angles. Usually one peak was lower in amplitude. Apparently, in these cases, the FD products required for generation of adjacent modes could be satisfied within the frequency adjustment range. As the incidence angle was increased, both modes continued to be detected at the slightly higher frequencies required to satisfy their respective FD conditions until one mode could no longer be detected within the frequency range of the tuning coil.

Length of Ultrasonic Path

In studies of the effect of metal path length, adjustments were made initially with a constant time maintained between the initial transducer pulse and the reflected signal as observed on an oscilloscope. However, this procedure can introduce errors in comparison of signal amplitudes for different incident angles because of the varying propagation velocities of different modes. A 0.2 μ sec change in transit time may produce up to a 5% change in signal attenuation. To increase the precision of signal amplitude comparisons, the metal path length should be physically determined by mechanical adjustments.

Loading

The test angle measurements were made with water on top the plate and essentially an air load on the bottom. Several layers of tissue were placed on the bottom and covered with sealed plastic to make an air pocket. This increased the signal amplitude 50% compared with water loading on both surfaces.

Transducer Response measurements by the four previously reported methods are complete. Tentatively, reflection measurement from a small ball bearing appears to be the best technique for measuring transducer differences. A report discussing the results is being prepared. Transducers with apparent differences will next be compared by tests with tubing. It is hoped that these latter tests will provide the rest of the bases needed to prepare tentative specifications for the selection of transducers acceptable for fuel sheath tubing testing.

Lamb Wave Propagation Theory predicts that particle motion within the test specimen is elliptical. The axes of the ellipses vary in amplitude throughout the sample thickness. It is therefore expected that the response to defects will depend upon defect depth. With some simplifying assumptions

in the theory, a mathematical expression relating ultrasonic responses to notch depth was derived. This expression was normalized to a set of experimental response measurements for notch type defects ranging in depth from 0.002 to 0.011 inch. The agreement between the theoretical curve and experimental points was very good, suggesting the assumptions made are at least approximately valid. The experimental data, however, are very limited, and more complete studies are necessary to prove whether such methods for predicting response are sound.

Identification of Mode and Propagation Frequency for Pulsed Systems

Group velocities for several modes in plate and tubing samples were measured with the Sperry UM 700 Reflectoscope. The measured velocities, entry angle, and sample thicknesses were used to calculate the instrument pulse frequency and to identify Lamb modes. The calculated frequencies ranged from 8.2 to 11.3 megacycles, indicative of the Reflectoscope frequency band width. This wide band width makes mode identification difficult.

Verification of Theoretical Lamb Wave Curves for Zircaloy Using Narrow Frequency Bands

The theoretical curves of phase velocity, V , versus frequency thickness product, fD , for zirconium have not been experimentally verified, nor has the applicability of the zirconium curves to Zircaloy-2 been demonstrated. The experimental curves for Zircaloy-2 are needed for an understanding of Lamb wave propagation in thin-wall tubing. The V versus fD curves for zirconium show that Lamb wave modes are very closely spaced and the Zircaloy curves are expected to be equally crowded. Thus, signals with very narrow frequency bandwidths should be used for experimental work. Extremely narrow bandwidth is provided by continuous sine wave ultrasound which can be made visible by the schlieren technique.

Existence of Lamb mode propagation can be made visible also. A Lamb mode is made to propagate in a Zircaloy-2 test specimen which is immersed in water. This leakage energy can be observed with the schlieren method which gives a direct verification of mode propagation. The entry angle for a particular mode can be optimized by maximizing the observed leakage energy. The ultrasonic zero entry angle can be obtained either optically, by the schlieren image, or by use of the usual front surface pulse-echo method. Mode entry angles can then be determined mechanically and verified optically. Experimental points for a number of modes were compared to the theoretical zirconium curves. Experimental points were obtained with Zircaloy plate and tubing samples. The experimental data indicate that the general shapes of the theoretical curves are valid for both plate and tubing samples. However, some adjustment appears necessary due to the fact

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that the experimental curves (with only one exception) were consistently higher in phase velocity than the theoretical curves for zirconium. The cause for these phase velocity differences could be the result of slightly different longitudinal and shear velocities for zirconium compared to Zircaloy.

PHYSICAL RESEARCH - 05 PROGRAM

Mechanism of Graphite Damage

A constant temperature enclosure was set up to provide a stable reference temperature at which to measure the resistivity of electron irradiated graphite.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

The third series of diffusion experiments was completed on March 31 at Cape Canaveral, Florida, providing an independent set of data to check the prediction equations for the weather information network and display system installed there. A total of 76 experiments were conducted at the site under a variety of meteorological conditions. About 70 percent of the samples have been assayed at Hanford and basic diffusion parameters computed on the IBM 7090. In addition to successfully providing the required diffusion predictions in support of advanced missile firings, the data comprise an important addition to the Hanford data for evaluating topographic and climatic influences on atmospheric dispersion patterns.

At Vandenberg AFB, the extensive damage done by weather was completely repaired, but delays in completion of the weather information network and display system (by another contractor) will necessitate a third test series at a later date to obtain the independent sample. At the close of the second series on March 31, a total of 80 experiments had been completed.

Ten raindrop size distributions obtained during natural rains during recent months with the Hanford raindrop sampler were reduced to graphical form. This represents only a fraction of those available from some 200 minutes of high quality records available. Each spectrum was based on about 300 drops, resulting in a standard error of 15 percent. The spectra fall naturally into three classes: 1) numerous drops smaller than 0.4 mm and almost no drops larger than 1.2 mm with secondary maxima below 1.0 mm; 2) broader distributions exhibiting two peaks in drop frequency, one below 0.4 mm and the other at about 1.0 mm; and 3) smooth distributions with a maxima at 0.4 to 0.5 mm.

The raindrop spectra taken by the Hanford sampler are in a flux density form, and as such may be used directly in scavenging predictions. Further, the instrument samples the small drop portion of the spectra in light rainfall, which represents the most important drop sizes for scavenging estimates. Deviations from published values of both scavenging efficiency and spectra have been found at Hanford, and continued research to verify these anomalies is required.

The data from all field test conducted at Hanford since 1959 were summarized and cataloged to facilitate analysis. A record, now up to date, will show the data taken, an appraisal of the success of the experiment, the type and quantity of tracer releases, status of the meteorological data reduction, and a listing of instrument failures or other circumstance which might complicate the analysis.

Dosimetry

A patient undergoing P-32 therapy at the University of Oregon Medical School was used to calibrate our bremsstrahlung counter for counting over the head of the subject. Measurements of subjects who regularly eat Columbia River whitefish are being continued to show the yearly cycle of body radioactivity.

A holder for two counters for the whole body counter was designed and is being fabricated. The purpose is to gather information more quickly and thus halve the counting time required. A few measurements with shorter than normal counting time were made with the shadow shield counter to study the deterioration of the pulse spectrum as less information is gathered for it. A shadow shield is being designed for a mobile counter to be mounted on a truck.

A proportional counter was made of low atomic number materials to be used for testing gases for X-ray counting. In such a counter the pulse spectra obtained from plutonium X-rays is not confused by fluorescence X-rays from the counter itself.

Fallout from nuclear tests is being found on the air filters of the whole body counter again. None has been found inside the iron room.

The positive ion Van de Graaff was back in operation after a lengthy shutdown for installation of new equipment. A new r.f. oscillator that had been installed would not work properly and was removed. Other changes have increased the ion output by 75%. With the new ion-getter vacuum pump it has proved possible to eliminate week-end servicing and to reduce consumption of dry ice and liquid nitrogen by $1/3$ and $1/2$, respectively.

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IBM computations were used to see if the scattered neutron flux in the center of the Van de Graaff laboratory, which had been assumed independent of position, might better be represented by a linear function of the distance from the center. Results at another laboratory indicated it might. Our computations showed it was not. The difference is probably due to the larger size of the exposure room in our laboratory.

The availability of the low-contamination deuterium gas-target made possible the completion of the calibration of the polyethylene double-moderator neutron dosimeter. Satisfactory data were obtained, but the gas-target has some reproducibility problems, too.

Several pulse-shape discrimination circuits were built and tested. Their use with LiI(Tl) scintillators showed that pulse-shape discrimination cannot be achieved with such crystals.

The background neutron dose rate in the 329 Building was measured to be 3.3×10^{-8} rad/hr. The slight difference from the value reported for the 700 Area could be due to differences in neutron production and scattering in the building structures. The large decrease from two years ago is probably due to the removal of a small neutron source from the building.

Experimental tests were begun of the tissue-equivalent chambers with neutrons from the Van de Graaff.

The discrepancy in the calorimeter data for the half-life of Sb-124 was found. It was due to one set of high readings that had undue influence on the least-squares fit of the data. The final value for the half-life was 60.20 ± 0.03 days (95% confidence). This very accurate value will be useful in making corrections for the radioactive decay of Sb-Be neutron sources.

Radiation Instruments

Accelerated development and design effort was carried out on various types of pocket-size dose-meters for both visual readout and preselected aural signal levels. The ionization chamber sensor that employs an illuminated quartz fiber and a CdS light detector was extensively modified in an attempt to reduce the present high cost and to provide improved reliability. Sensitivity variation with temperature of the CdS cells remains a problem. The quartz rod mounting for the automatic recharging type ionization chamber sensor was redesigned to provide better rod positioning. A bid was received from a manufacturer of \$15 each in lots of 100 for the automatic recharging chamber. Several low-power miniature transistor binary circuits with sub-miniature light indicators were developed and tested for use with this type

of sensor, and in addition, several micro-module binaries were purchased for possible use. Since the automatic recharging type dosimeter provides a single pulse-out for an approximate accumulated dose of 10 mr, a series of binaries can easily provide readout and signal activation for doses of 10 mr, 20 mr, 40 mr, 80 mr, and 160 mr for example. This approach, which is being actively pursued, should provide a compact and inexpensive dose-meter if large lots can be fabricated. As an alternate to the binary method, a miniature electromechanical register can be employed. In brief, three basic approaches to the pocket dose-meter problem are being actively pursued.

The developed coincidence-count type scintillation counter for the standard HAPO plutonium-collection air filters has eliminated the present pre-counting waiting period requirement to permit decay of the entrapped radon-thoron background products. A continuous air-borne plutonium concentration of 2×10^{-12} $\mu\text{c/cc}$ will result in a filter collection to provide a 2700 d/m rate after 24 hours for an air flow of 15 CFM. This collected value can be checked immediately after filter acquisition in the developed counter even if the radon-thoron concentration has been very high the same period. The present HAPO filter counting method would require an approximate 72-hour waiting period for radon-thoron decay before the same sensitivity value could be attained. Some 300 regular filters from the 308 Building have been successfully counted with the experimental instrument to date. Detection head fabrication and commercial electronic instrument acquisition was started for a final model system.

A complete, detailed review was made of instrument specifications prepared by Instrument and Electrical Development, IPD, for a logarithmic response, six-decade, electrometer-type picoammeter planned for use with a standard HAPO HM chamber as a radiation area monitor. The picoammeter, which is to have two selectable-level alarm trips, is desired by IPD to replace the present linear response electrometer instruments used in the reactor buildings. The desired logarithmic response is to extend from 1 mr/hr to 1000 r/hr for all monitoring locations. In essence, the proposal is to replace one vacuum-tube instrument with another vacuum-tube instrument. A number of detailed technical comments were made concerning the specification.

The experimental scintillation coincidence-count continuous alpha air monitor continued to perform satisfactorily and all necessary operational data have been obtained. Fabrication continues on a final model of the instrument which will be used for field demonstration.

Construction of the 400-channel pulse height analyzer for the positive ion accelerator continued this month.

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Discussions were held with Biology Operation personnel regarding development and design of a cattle thyroid monitor to be especially rugged and reliable for direct use in the field. Basic detector requirements were established and portions of the system will be of the commercial type.

Circuit development was started on a miniature low-voltage DC to DC converter for use in battery-operated portable radiation monitoring instruments. This approach will permit the use of one rechargeable battery to supply all power requirements for the instrument transistor circuits. Although some over-all power efficiency loss will be incurred with this approach, it should considerably reduce the present maintenance costs caused by battery replacement.

A transistor preamplifier was developed for use with organic scintillation detectors and with standard vacuum-tube linear amplifiers. The preamplifier current gain effectively cancels the capacitance loading effects of the detector-to-amplifier coaxial cable. This method improves the signal-to-noise ratio by 20 db and permits easier pulse detection for phototubes used with organic scintillators such as stilbene, anthracene, and terphenyl-in-polyvinyltoluene. The use of the preamplifier requires minimal modification of the main amplifier, and the amplifier remains usable with other inputs.

A majority of the necessary circuit design work has been completed for the Atmospheric Physics experimental portable mast system. All bids were reviewed and accepted for the commercial instruments to be used in the system. Fabrication, which will be considerable, was started on portions of the circuitry.

A stable transistor post amplifier was developed and tested for use in the experimental logarithmic and linear response scintillation area radiation monitors. The amplifier permits the use of 100 microampere meters which are more rugged and reliable than are the present 20 microampere meters. In addition, several types of non-contacting type set-point alarm actuation 100 microampere meters are available. With the added amplifier, the chopper input area monitors have a voltage gain of 10,000 and an equivalent current gain of about 2000. The system is stable from 0°C to +60°C. A standard, inexpensive meter was modified to utilize three photocells for alarm-trip purposes. Two of the photocells provide low and high level alarm activation over the full instrument range of four decades or 1 mr/hr to 10 r/hr. The third cell, which is fixed in place, switches the instrument high voltage through use of a coaxial relay to permit instrument response extension to 1000 r/hr. The method is fail safe in that the cells must be illuminated to prevent alarm actuation; thus, loss of illumination, photocell damage or breakage, etc. will cause alarming. The movement of the meter pointer,

which is equipped with a flag, momentarily shields the photocell from light to cause alarm energizing. One experimental, six-decade logarithmic-response, scintillation area radiation monitor is being fabricated.

Circuitry modification effort continued on portions of the Atmospheric Physics Radiotelemetry System. A number of resolver windings were determined to be actually wired backward and were corrected. Transient noise problems from the stepping switch on the scanner chassis caused appreciable wind direction indication errors, and RC filters were designed and added to alleviate the problem. Several sine-cosine potentiometer connections on the wind direction integrator chassis were found to be reversed. The commercial anemometer wiring drawings were found to be in error with the result that all anemometers had been incorrectly wired into the circuitry. Both items were corrected. A general maintenance procedure guide was prepared and given to the instrument technicians. The guide includes step-by-step analysis procedures and necessary equipment for testing the anemometers, integrators, servos, and output potentiometers. Most of the system schematic drawings have now been corrected.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples as received during the month. The ion detection system of the mass spectrometer for this program has not yet been put into operation.

Design was started on an ion detector to detect heavy ions which have been isotopically analyzed through the detection by a scintillation detector of the secondary electronics produced by post acceleration of the ions to energies of about 50 kv. The detection system is being designed for study on the ion-optic test bench.

TEST REACTOR OPERATIONS

The PCIR was operated on an intermittent basis during the month. There was one unscheduled shutdown which was caused by faulty bypassing technique. The pile modulator was improved mechanically and was tested and calibrated. Various minor maintenance items were completed.

The TTR was operated one night for the University of Washington Graduate Center. There were no unscheduled shutdowns. The critical mass approach tank in the reactor room was used during the remainder of the month. Approach measurements are being made using 1.8% Pu-Al rods in light water moderator.

A solution for the neutron flux as a function of time has been obtained for

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the TTR in support of the revised hazards report for that facility. A solution was obtained for a ramp input of reactivity of four cents per second. The solution was obtained with the IBM 7090, code, HAIREX.

Weather Forecasting and Meteorological Service

The first in a series of scheduled reports on prospective 1962 crests of the Columbia River flow at Hanford was issued on March 21. It shows crest elevations this year should be from 4 to 7 feet under those of last year as indicated in the following table.

	<u>1962 Forecast</u>	<u>1961 Observed</u>
Peak Stage (feet MSL) at: 100-B	404.0-407.0	411.2
100-K	399.5-402.5	--
100-D	393.0-396.0	399.5
100-H	385.5-388.5	392.0
100-F	380.0-383.0	386.8
Richland	348.5-351.5	354.3

The hydrogen and thermogram of the river for 1961, along with normal curves and selected extremes for prior years, and an extreme value analysis of annual floods from 1913 to 1961 were also included.

Consultation service was rendered on meteorological and climatological aspects of environmental safety analysis in reactor siting to APED. Meteorological services, viz., weather forecasts, observations, and climatological services were provided to plant operations and management personnel on a routine basis:

Weather Summary

<u>Type of Forecast</u>	<u>Number Made</u>	<u>% Reliability</u>
8-Hour Production	93	85.2
24-Hour General	62	88.2
Special	200	89.0

Temperatures during March averaged somewhat below normal. Precipitation was only 30 percent of normal.

Instrumentation and Systems Studies

The 200-channel analyzer was loaned to Ceramic Fuels Development, 327

Building, for a series of scintillation-detector measurements. Instructions for operation were given along with a demonstration.

Instructions were given to CPD instrument technicians regarding the adjustment and operation of an Alpha-Beta-Gamma Aurally-Indicating radiation Monitor and regarding the use of adjustment of the Plutonium Wound Monitor.

Equipment purchase specifications were completed for a criticality monitoring system for Finished Products Chemical Technology, CPD.

Actual contaminated garment tests were continued with the experimental Automatic Conveyor Laundry Monitor designed and fabricated for Protective Equipment Decontamination, CE&UO. Spot alpha contamination levels ranging from 450 d/m to 23,000 d/m were easily detected on a number of garments. Similarly, beta-gamma contaminations ranging from 50,000 d/m to 325,000 d/m were detected. The contaminated garments were obtained from the laundry facility after determination of contamination by laborious hand-survey methods. Some meter-scaling difficulties were encountered with the instrument when the alpha detectors were exposed to high level alpha contamination. Such levels caused jamming; however, a transistor backup circuit was designed and tested to provide alarm tripping if scaler jamming occurs. Several minor mechanical adjustments and changes remain to be completed. A complete operation and instruction manual was started.

Fabrication continued in the 328 Building Electronics Shop on the Automatic Columbia River Radiation Monitoring System. This scintillation and transistorized system was designed for Radiation Protection Operation, HLO.

Advice and assistance were rendered to Chemical Effluents Technology, HLO, concerning the design of a zero-suppressed DC voltmeter to be used in measuring the magnetic field strength of the earth.

Fabrication and testing was nearly completed on a transistorized aurally-indicating alpha detection monitor which uses a standard HAPO alpha air proportional probe as a sensor. The instrument was designed for use by both Finished Products Chemical Technology and Control Operation, CPD.

Specifications (HW-73150) were written for the purchase of a Data Logging System to scan and read out thirty-six temperature and pressure points at the HLO Physical Metallurgy K-West radiation effects facility. A study was begun to determine the effort and equipment needed to place the entire control and data acquisition system on a sound basis from the standpoint of reliability, maintenance, and provision for future expansion.

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Proposals were reviewed for supplying a \$10,000 digital voltmeter to Physical Metallurgy, and a letter of recommendation was written explaining the desirability of selecting the high bidder.

An examination of engineering drawings for a shielded enclosure showed several errors in the application of shielding principles. These errors were corrected. The enclosure is to be used to reduce radio-frequency interference from a device in the 326 Building metallographic lab.

Assistance was given in tuning several temperature controllers on the FPD cap welding process. Several tests were run to determine that the process would remain within the specified temperature limits for all normal variations in the capping process.

Work resumed late this month on calibration of micro-displacement readout systems to be used by Physical Metallurgy Operation for in-reactor creep measurements. Preparations are being made to re-locate the reference system on a seismic pad in the 326 Building to eliminate the errors due to building vibration experienced in the 306 Building. Calibration of the transducers used in the third generation creep capsules will resume as soon as the equipment has been set up in the new location.

Optics

A six-inch traverse mechanism has been designed for use in traversing reactor process tubes in B, D, F, H, D, and DR reactors. The unit has sufficient range to read out the most severe bends encountered in reactor process tubes. A shakedown test at 185-F in an ex-reactor tube revealed the target disappeared from view, with severe bending of the borescope for borescope lengths in excess of twenty feet. We also had difficulty with electrical contacts. Modifications of the borescope objective section have been made to correct these faults.

A traverse mechanism has been designed for use to determine the alignment of four-inch lateral casings to be installed under the high level waste tanks in the 200 Areas. This unit is over eight feet long, compared to the one foot length of the process tube traversing mechanism. The two units required for the job will be fabricated in the J. A. Jones shops in North Richland. A closed circuit television camera will be used to read out the pipe deflections indicated by the mechanism.

The application of radiation pyrometry to temperature measurements in a reactor environment was discussed with Barnes Engineering people at their plant in Stanford, Connecticut. The trip was made at the request of Plutonium Fuels Development in support of one of their experiments. A tour of the Barnes Engineering Company facilities revealed a number of interesting instruments, one of which was an infrared micrometer which read the

diameter of cylindrical objects with an accuracy of 0.001 inch when no part of the instrument was closer than 24 inches from the object. The unit on display was built for use in monitoring wire diameter as the wire was extruded out of a die.

During the five week period (February 25-April 1) included in this report, a total of 600 man-hours of shop work was done. The work included:

1. Repair of two microscopes for Biological Analysis Operation, HLO.
2. Fabrication of six ceramic dies for Fuel Fabrication Development, HLO.
3. Fabrication of components of a radiometer for Physical Measurements Operation, HLO.
4. Plating of five quartz fibers with gold.
5. Fabrication of components for an ultrasonic wall thickness probe.
6. Repair of two camera shutters for Radiometallurgy Operation.
7. Fabrication of a microscope objective to be used in high temperature work by Ceramic Fuels Development Operation.
8. Fabrication of four quartz coil mounts for Manufacturing Maintenance, FPD.
9. Fabrication of two quartz lenses for use with a high intensity mercury vapor schlieren source.
10. Recementing and reconditioning of a microscope lens for Plutonium Fabrication Development, HLO.
11. Installation of two non-darkening lenses in the two Redox crane periscopes.

Physical Testing

Service testing work proceeded routinely. A total of 4,734 tests were made on 4,713 items, representing some 43,600 feet of material. The greater part of this material continues to consist of tubular components. Test work included: borescoping; dimensional measurements (micrometric); eddy current; magnetic particle; mechanical tests (bend, hardness, impact, and tensile); metallography (macro and micro examination, and fracture); penetrant (fluorescent O.D. and I.D.); radiography (gamma-ray and X-ray); stress analysis (electric resistance wire strain gages, and X-ray diffraction); surface treatment (alkaline cleaning, conditioning, pickling for autoclaving, steam detergent cleaning, vapor degreasing, and vacu-blasting); and ultrasonic (flaw detection and thickness measurements). Work was done for 26 different HAPC components representing most of the operating departments and service organizations, and other AEC contractors. Advice was given on 46 different occasions on general testing theory and applications.

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Testing and treatment of the NPR Zircaloy-2 pressure tubes was resumed on 21 tubes received during the month to complete the tube order. All of the tubes were processed through testing and treatment with the exception of seven tubes remaining to be examined upon completion of their autoclave cycle. The only other work anticipated on NPR pressure tubes is re-work that may be necessitated by inadvertent damage to the tubes during installation in the reactor.

Twenty-three spare PRTR pressure tubes were autoclaved and subsequently examined. Of these, 17 proved to be satisfactory; the remaining 6 will require re-work to check suspicious areas. The remaining PRTR tubes will be completed upon completion of NPR pressure tube work.

Testing of fuel element sheath tubing proceeded routinely. It appears that the work on available tubes will be completed this month. Except for some clean-up work on a small number yet to be received, all other tubes will be back-logged until the improved ultrasonic tubing test system is available for use.

Field testing activity consisted of X-ray and gamma-ray work at PRTR (rupture-loop, primary coolant, and helium systems), NPR (instrument leads and test coupons), De Laval pump impellers (100-H, 100-D, and 100-DR), and Purex No. 5 concentrator in 200-E Area fabrication shop. Fluorescent penetrant tests were conducted on over-bore nozzles, pump impeller shafts, and reactor nozzle retaining rings. Ultrasonic thickness measurements were made on the heat exchanger in the 308 Building, and piping in the NPR Building. Stress analysis was conducted on inconel pigtails for the BDF reactors.

Considerable effort was expended on the NPR primary loop piping in the 300 Area labs, with work on the following problems: Thirteen comprehensive reports covering the Ladish, Taylor-Forge, and seamless pipe samples that were evaluated have been compiled and issued. Interpretation of Ladish pipe radiographs was completed, a report written and issued, and the film returned to Kaiser Engineers for disposition. To complete the evaluation of HUICO circumferential pipe welds, four samples were sulphur printed. The sulphur prints were made in an effort to establish the cause of apparent dissimilar weld metal in one of the weldments. Negative results were obtained, indicating some other cause for the apparent difference. A report on the evaluation was completed and issued. Work was continued on the evaluation of the use of eddy currents for detection of heat treatment condition. Various samples were sand-blasted and heat treated for the test. Assistance was given in the preparation of impact specimens for further testing of weld quality.

Though not directly related to the NPR primary piping, an evaluation was made of inconel-to-carbon steel weld samples. A complete evaluation was made, including radiography, fluorescent penetrant, and metallographic evaluation.

The electronic system of the infrared fuel-bond heat transfer test is being realigned and an induction heater is being installed as an alternate heat source.

Trial exposures were made with an iridium-192 source in the isotope cell to evaluate the isotope as an energy source to fill the 400 to 600 KV energy gap for radiography.

Experiments were conducted to design a test to improve the image detail obtained when radiographing the inner weld of an I&E fuel element. Previous trials using a panoramic technique with the fuel element and film revolving under a slit proved to be inadequate. An alternate technique is being tried involving sequential exposure utilizing high X-ray tube current with extremely short exposure times and exposing the sample six times around its circumference.

The special transducers for the process tube crack ultrasonic immersion test development were received. A small mechanical carrier to mount and translate the transducer was fabricated. An initial test indicated that a 1/4-inch-long x 0.018-inch-deep electro-machined notch could be easily detected in the inside of a standard section of process tube. All of the components necessary for a mockup test are being assembled to evaluate the system before actual reactor application.

Plutonium Fuels Development Operation, HLO, requested a study to determine if an ultrasonic unbond test could be developed for the fission product transient samples. These samples are co-extruded, aluminum-clad plutonium core tubes. Unbonds greater than a 1/8-inch x 1/8-inch square are rejectable. Since the cladding thickness is only about 0.20 inch, conventional ultrasonic pulse-echo techniques cannot be used because of insufficient instrument resolution. A through transmission technique using a transmitter and receiver crystal appeared to be practical. The through transmission method was tried and it proved to have adequate sensitivity to detect a 1/8-inch-diameter artificially-fabricated unbond. Actual fuel samples will be tested, radiographed, and sectioned to reveal the reliability for detecting natural unbonds.

ANALOG COMPUTER FACILITY OPERATION

The major analog computer problems considered during the month were:

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1. NPR Secondary Loop
2. Instrumentation for Old Reactors
3. Speed of Control
4. NPR Simulator
5. "C" Column Simulation

The class on Analog Applications in the Chemical Industry given to members of CPD and Chemical Development Operation in HLO was concluded.

A presentation was given on new techniques in analog computation to Dr. George Marr of Electronic Associates.

Two class sessions on analog computation are planned for presentation to members of Operations Research and Synthesis Operation and others. The sessions are intended to provide class members with the philosophy of analog computation.

The optimization of a function of two variables with an analog computer was demonstrated to be feasible with the automatic plotting of an inverted parabola on an x-y curve plotter. Possible applications exist in process control problems.

The GEDA was 86 percent and the EASE 90 percent operational during the month.

<u>GEDA</u>	<u>EASE</u>	
151	153	Hours Up
23	22	Hours Scheduled Downtime
<u>1</u>	<u>0</u>	Hours Unscheduled Downtime
175	175	Hours Total

Trouble experienced with support equipment delayed the completion of several analog problems. Support equipment includes recorders, transfer function analyzers, and readout equipment.

The EASE 1159 multipliers were exceptionally noisy this month and are causing the problem being run to be disrupted periodically.

The routine maintenance has, in general, been very good.

All function generators on the GEDA are now working.

INSTRUMENT EVALUATION

With the successful completion of nearly one year of continuous charge-discharge cyclic tests on one type of Ni-Cd rechargeable battery, one portable radiation monitoring instrument is being modified to incorporate the rechargeable batteries for direct field tests.

One scintillation transistorized area monitor, incorporating both logarithmic and linear ranges, was fabricated and successfully tested for use in the 326 Building. Except for the necessity of replacing one wired-in-reverse tantalitic capacitor, the instrument tested satisfactorily in all respects.

Except for several light shield pinholes which needed spotting (a simple, two-minute task), both the scintillation transistorized hand and shoe counters (one for alpha, beta, and gamma and the other a background-compensated beta-gamma unit) worked correctly during the month. All HAPD drawings and specifications necessary for off-site fabrication were completed for the combination alpha-beta-gamma monitor and the HAPD drawings are nearly completed on the background-compensated beta-gamma monitor. Both units have been accepted by RPO for general HAPD use. The drawings and specifications were cooperatively prepared with members of RPO. Four scintillation alpha-beta-gamma counters are being ordered for use in the 327 Building.

The improvement of ground-point connections has apparently solved the sporadic oscillation problem on the aluminum-chassis Model II Scintrons of which 30 were purchased from an off-site fabricator for HAPD use. Most of the instruments have now been in satisfactory continuous use for three or four months with only one component failure of a high voltage transformer. This operational life equates to one component failure for about 100 instrument-months of operation.

All laboratory evaluation tests were successfully completed on the prototype scintillation transistorized portable alpha "poppy" monitor. HAPD drawings were started and RPO plans a large purchase in July, 1962, to replace the present obsolescent portable vacuum-tube alpha "poppies".

Evaluation tests were completed on one prototype portable transistorized BF₃ neutron monitor fabricated to HAPD Design Specification G-503766-1 by a Seattle manufacturer. Thirty of the units have been ordered by RPO to replace the obsolete vacuum-tube BF₃ portable instruments. Several errors were noted and corrected in the prototype; however, general performance was satisfactory.

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Recalibration tests were performed on one scintillation logarithmic response area monitor designed two and one-half years ago for use on reactor elevators. The monitor, with two other identical units, has been in continuous use for two years on the 105-F reactor elevator. These units were of the original design in which the phototube anode current drives the meters directly. Even after two years' use at anode currents averaging about four microamperes, the unit was easily calibrated with an increase in high voltage. The calibration change was a negative 50% in 24 months for an average change of 2% per month. This is excellent service, considering the application conditions and the high average anode current. The results demonstrate the usefulness of phototubes in long-term radiation area monitors.

Two commercial "pocket-pencil" dose-rate monitors were received for evaluation tests. These units are a modified version of the ORNL chirping and flashing pocket pencil; however, these start to sound and flash only above a factory-set dose rate. One of the units is set for a 2 mr/hr trigger level and the second for 75 mr/hr. Such an idea seems applicable to many IPD and CPD dose-rate "alarming" problems.

for PFQ by RSPaul
Manager

PHYSICS AND INSTRUMENT RESEARCH
AND DEVELOPMENT

PF Gast:RSP:mcs

CHEMICAL RESEARCH AND DEVELOPMENT OPERATIONRESEARCH AND ENGINEERINGFISSIONABLE MATERIALS - O2 PROGRAMIRRADIATION PROCESSESReactor Radioisotope Reduction Studies

The two K-Area reactor water treatment plants have reduced their alum feed rates to levels which produce a floc with a zeta potential of from -3 to -5 mv in a test of the effectiveness of this treatment for radioisotope parent removal and of the feasibility of determining water quality by the zeta potential method. At present this requires about 9 ppm alum. At the same time, the plants have shortened their filter runs somewhat in an attempt to keep the turbidity of the water to 0.003 ppm or less. These conditions represent those optimum as determined in laboratory tests for producing water of a pH of 6.6 and using the alum floc - sand and anthrafilt filter techniques. Reductions in the P-32 content of the effluent water from these reactors should result.

Laboratory studies were made with the aluminum turnings columns which are used to simulate reactor process tubes in a test of effect of silicate concentration and pH. A low alkalinity silicate was used and the tests indicated that its effectiveness in reducing phosphate ion adsorption increased throughout the concentration range of 10 to 50 ppm studied. Minimum adsorption of phosphate ion was observed between the pH's of 7 and 8 with a rapidly increasing adsorption with decreasing pH. At a pH of 5 the adsorption is about an order of magnitude greater than at a pH of 7. These tests indicate that the silicate effectiveness in reducing P-32 formation in reactor effluent water could be as much as 50 percent greater in reactors operating at a pH of 7 than in those at a pH of 6.6, depending on the silicate concentration used.

The effectiveness of a sand and anthrafilt filter simulating the process water filters in removing radiophosphorus-containing materials was tested in the small laboratory water treatment apparatus. Water spiked with P-32 as the phosphate was treated by alum flocculation and then passed through the sand and anthrafilt filter. At the start of the process 89 percent of the P-32 was removed. After five hours, the removal had increased to 95.8 percent. Additional filtration through millipore filters removed some additional P-32 so that the combined removal efficiency was greater than 99 percent for the entire time of the test. These data indicate that the water treatment process results in the formation of P-32 on particulate material which is only partially removed by the present filters. Studies on improved filter media have been started.

Soil Permeability of NPR Decontamination Wastes

Soil permeability studies of NPR decontamination waste liquors continued. The permeability of a combined waste adjusted with sodium hydroxide to pH 9 and filtered was 50 percent that of water. As reported last month, the permeability of a combined waste adjusted to pH 13 and filtered was 40 percent that of water. Soil permeability of a three-step waste was 65 percent that of water. None of the wastes investigated to date has exhibited a tendency to plug or otherwise adversely affect crib disposal.

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Disposal to Ground

Work progressed on evaluating the distribution of macroscopic river arrival times of water draining from the 100-N disposal crib. Two evaluation methods are being used. The first is an approximate method which uses two-dimensional flow patterns already available from an earlier 100-N Area study. The second method is a rigorous analytical treatment of the three-dimensional flow from the crib. Computer programming is underway on this latter, more realistic evaluation. The approximate method yielded a minimum macroscopic travel time from the crib to the river of 32 days. By the 35th day about 60 percent of the initial volume discharged will have reached the river, and about 98 percent will have entered the river 37 days following discharge. The results of this study, together with radionuclide migration rates and sorption data, are being used to estimate the river contamination that might result from various fuel element rupture frequencies and degree of severity.

Ground Water Temperature Studies

Ground water temperature profiles obtained for project monitoring wells during the current month were essentially the same as those noted last month. Minor temperature increases of from 0.2 - 0.5 C were noted in riverbank wells which normally are colder than the average ground water temperature of 16 - 17 C. The increase is attributed to the flow of ground water into the wells during this period of minimum river level. Temperatures in these wells are expected to decrease during the next few months as the river level increases and the aquifer is recharged with cold river water. Ground water temperature and elevation data collected during the recharge cycle will be useful in evaluation of the effects of river level on local ground water conditions.

Recent estimates show that the ground water under 100-B Area constitutes about 5 percent of the intake coolant flow at the 181-B pump house. This value is in good agreement with the 6 to 10 percent ground water contribution estimated previously from radionuclide concentration and water temperature data collected in November, 1961. Similar studies are in progress for downstream reactors. Also, the effects of an increasing river level on the ground water contribution to coolant intake will be followed closely in the coming months.

Airborne Particulates in Reactor Operations

Some 39 air samples were taken during tube replacement operations at 100-D. The radioactive particulate material collected was characterized as to solubility, isotopes present, and air concentrations. Solubility of radionuclides present was 3 - 5 percent, in essential agreement with previously reported results. Airborne nuclides identified were found in the approximate concentrations:

P-32:	5×10^{-10}	$\mu\text{c/cc}$
Cr-51:	1×10^{-10}	$\mu\text{c/cc}$
Zn-65:	1×10^{-10} to 3×10^{-9}	$\mu\text{c/cc}$
Fe-59:	3×10^{-11} to 4×10^{-10}	$\mu\text{c/cc}$
Sc-46:	1×10^{-11} to 1×10^{-9}	$\mu\text{c/cc}$
Mn-54, Co-60, Sb-124:	1×10^{-12} to 1×10^{-10}	$\mu\text{c/cc}$

The mass median particle diameter was determined to be three microns. Graphite dust collected during broaching contained $2.5 \mu\text{C}$ C-14 per gram.

The foregoing data are to be used for reassessment of the internal radiation hazard if particulates are breathed during tube removal operations.

Effluent Monitoring

Assembly of the As-76 prototype was completed. Final checkout of the sampling and electronic systems is in progress. Calibration of the detector was started. Stability of the detection system was improved by substituting a transistorized preamplifier for the conventional White cathode follower. Essentially no heat is generated by the transistorized device, and consequently more stable performance should result.

SEPARATIONS PROCESSES

Disposal to Ground

Test core-drilling to evaluate technique and equipment is being done in uncontaminated soil adjacent to the 216-Z-9 Recuplex crib. Core recovery has not been satisfactory in the unconsolidated sand due to blockage of the bit. Successful cold drilling must be demonstrated before coring of contaminated sediments in the bottom of the crib is undertaken. Additional equipment and technique modifications are being made in an effort to achieve maximum core recovery.

A tritium concentration of $2 \times 10^{-4} \mu\text{C/cc}$ was detected in a water sample from well 699-20-E8, located eight miles north of the 300 Area and one-half mile from the Columbia River. This well evidently intercepts the relatively permeable aquifer known to exist in this region. Ground water temperature measurements along with tritium and gross beta emitter analyses from wells intervening toward the 200 East Area substantiate this as one of the preferential paths for the 200 East Area contaminated ground water movement. Shoreline samples taken at frequent intervals from the river directly east of well 699-20-E8 have been submitted for tritium analyses.

Iodine Sampling in Separations Areas Stacks

The study to evaluate the efficiency of sampling Redox and Purex stacks for I-131 was completed. Samples drawn from the 50-foot level at Purex were an average of 10 percent higher in I-131 concentrations than were samples drawn from near the top of the stack. Line adsorption losses likely account for this difference.

Retention on Charcoal of Oxidizer-Evolved I-131

Low I-131 removal efficiencies were experienced for an experimental charcoal adsorber located in a jumper from the oxidizer vessel vent. Assistance was given CPD in sampling the stream to confirm that the charcoal was giving much lower removal than anticipated and to yield information helping to explain this. A small charcoal cartridge placed in a sampling line from the oxidizer vent also showed low retention, about 85 percent. Additional tests of the retention on charcoal of I-131 evolved from an oxidizer solution are planned to explain the anomalous behavior.

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Purex Solvent Extraction Studies

Satisfactory operation of the Purex Plant 2D and 2E columns has not been achieved since installation of the six percent free area nozzle plate scrub cartridge several months ago. The decontamination provided by the 2D column has been considerably lower than that demonstrated before installation of the cartridge, and the column has been subject to unpredictable flooding when the frequency was raised to improve the DF. A decrease in 2E column efficiency and capacity has paralleled the mal-operation of the 2D column. The pattern of poor performance also coincides with the yearly cycle of generally poor C-column performance which has been attributed in the past to a demineralized water impurity. A short program has been initiated in the glass Demonstration Unit pulse columns using plant cartridge configurations in an attempt to duplicate and clarify the plant performance.

Highlights of the findings thus far are the following:

1. Capacities of the Demonstration Unit 2D and 2E columns met or exceeded those obtained in the original development programs (see HW-69946 and HW-70059). At the normal plant volume velocities and temperatures, stable operating frequencies about 30 cycles per minute greater than those obtainable in the plant were achieved.
2. A specific batch of Purex demineralized water obtained just after a recent shutdown (March 12, 1962) had only a minor effect on the dispersion characteristics or the flooding frequency (generally less than five cycles/min) when used as either 2DS or 2EX.
3. Attempts to make "bad" water by adding such components as liquid detergent, ground resin fines, or sodium silicate - all in the 100 ppm range - were only marginally successful. The flooding frequency with such additives was lowered less than 10 cycles/min and was still about 20 cycles/min greater than that of the plant.
4. The capacities of the extraction and scrub sections of the 2D columns were virtually the same; a tight emulsion layer built up above the top of the scrub section at about the same frequency that signs of instability were observed in the lower third of the nine-foot-high extraction section. Further increases in frequency caused complete flooding of the extraction section while the position of the "interface" at the top of the scrub section tended to remain stable (the normal interface in the 2D column is at the bottom).
5. Temperature was found to have a large effect on the 2D column performance. At ambient temperature, the flooding frequency was about 20 to 30 cycles lower than it was at 50 - 60 C. When the scrub stream was not heated, the equilibrium temperature at the top of the column was about 40 C. It was observed that the tight emulsion layer formed at the top of the scrub section when the column was operated at the instability threshold at ambient temperatures would dissipate when the temperature at the top of the column reached about 30 C from heat added via the 2DF and 2DX. This temperature effect may be the cause of unexpected instability frequently observed in startup of the Plant 2D column.

6. Operation of the 2D column with a 2 M HNO_3 scrub and a low uranium loss to simulate HA column performance also demonstrated a close match between extraction and scrub flooding frequencies. Instability in the extraction section was more pronounced at the feed point, however, and was somewhat easier to break by reducing the frequency.
7. A sidelight, which may have no relation to plant performance, was observed in the 2D tests: A brown impurity in the feed, evidently colloidal ferric hydroxide, was observed to reflux in the lower portion of the extraction section. This behavior was especially pronounced near the instability threshold and was, in fact, an excellent harbinger of flooding. Also near the instability threshold, this brown material could be seen backmixing into the scrub section. The reflux and entrainment behavior of the hypothesized colloidal iron is believed due to conversion of the dispersed phase aqueous into interfacial films -- that is, a change from aqueous drops dispersed in the continuous organic phase to organic drops enclosed in aqueous films. No real phase inversion could be seen until the instability had progressed much further.
8. A quick comparison of the sandwich cartridge developed by DeMier and Jansen (HW-56450, HW-68846) with the plant-type 23 percent free area nozzle plate 2E column showed that it provided about 20 to 30 cycles/min more frequency rangeability than the present cartridge at no apparent loss in efficiency. Use of such a cartridge might provide more flexibility for coping with the evidently recurring water problem.

Reduction of Uranyl Nitrate to Uranium(IV) Nitrate with Aluminum

Studies on possible methods of introducing hydrazine stabilized uranium(IV) solution (a substitute for iron(III) sulfate) to the Purex 1BX partitioning column for a plant test have ruled out introduction of an aqueous stream at the top or bottom of the column due to distribution character. Additional batch contact tests to study adding the material to the column feed stream tank have also been discouraging.

In lieu of column modifications which would be unwarranted for just test purposes, one remaining alternate is to introduce the uranium(IV) as an organic stream. In support of this concept, a series of laboratory scale experiments has been initiated to determine the feasibility of producing high concentration of uranium(IV) by the reduction of uranyl nitrate with aluminum metal for this single plant test. The extraneous salts introduced in the aqueous would not be objectionable since the uranium(IV) nitrate is transferred to the organic before use.

Highlights of the studies are:

1. High yields of uranium(IV) nitrate can be obtained by a controlled reaction in a solution at least 0.5 molar in aluminum nitrate and containing hydrazine or sulfamic acid as a holding reductant.
2. The reaction proceeds equally well with aluminum dust or with a gross excess of aluminum turnings.

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3. Comparable results can be obtained with either hydrazine or sulfamic acid as the holding reductant.
4. With a deficiency of nitric acid, a fine, black precipitate forms. The precipitate does not readily re-dissolve in dilute nitric acid.
5. In the absence of aluminum nitrate, the reaction starts after an induction period. The length of the induction period is an inverse function of temperature. Once initiated, the reaction is very rapid and difficult to control if a high uranium(IV) nitrate concentration is desired.
6. The reaction rate appears to increase with temperature. In one run, the temperature was allowed to rise above 50 C. The reduction then proceeded rapidly raising the solution temperature to the boiling point.
7. The reduction rate is not always reproducible. In a pair of identical runs, the uranium(IV) ion concentration in one run was three times that in the second run after two hours. After four hours, the uranium(IV) ion concentrations were essentially the same.
8. The uranium(IV) ion concentration obtainable appears to be limited only by solubility considerations.
9. Measurements of the uranium(IV) nitrate oxidation rates ranged from 0.4 to 3.2 percent per day.
10. The uranium(IV) nitrate can be readily extracted into a 30 percent TBP organic phase. A black precipitate can form in the aqueous phase if the initial free nitric acid concentration is insufficient to compensate for that extracted into the organic phase.
11. The stability of the uranium(IV) nitrate in a 30 percent TBP organic phase is comparable to that in an aqueous phase when the organic is in contact with aqueous containing a holding reductant.

Ion Exchange Recovery of Plutonium from RMC Line - Continuous Task I Oxalate Supernates

The RMC-Line flowsheet involves continuous precipitation and filtration of plutonium oxalate from a 2.7 M nitric acid solution. The waste contains 0.3 to 0.4 grams of plutonium per liter, and this plutonium is currently recovered by a procedure which involves several oxalate-kill steps, a thermal concentration, and solvent extraction. Replacing this sequence of operations with a simple anion-exchange process would be desirable, provided the oxalate and fluoride present in the feed do not interfere seriously with the absorption reaction.

A series of laboratory-scale runs have been carried out with plant waste to test the feasibility and operability of the scheme.

The feed was butted to 7.2 M in nitric acid and nitrite added to oxidize the plutonium to +4. Additional oxalate was also added to insure that an excess was indeed present. In the first two runs the feed plutonium concentration was low (by a factor of ten) due to post precipitation of plutonium oxalate in the

container in which the feed was received. This precipitate was dissolved with nitric acid prior to the third run, and the feed so prepared was stable with respect to plutonium precipitation. The process was found to be fully operable. Although the capacity of the resin for plutonium from an oxalate-fluoride solution was lower, by about a factor of two, than would be obtained under ideal conditions, the capacity is more than adequate for the intended application. Details will be summarized in an informal report as soon as all analytical work is completed.

Solid State Electromigration Studies

Equipment has been designed and tested for solid state electromigration studies. It consists essentially of a Pyrex envelope with ground fittings at each end into which are inserted water cooled electrode clamp assemblies. The specimen to be used is installed in one clamp which is inserted into the envelope. The other clamp is engaged and tightened with a screwdriver through a side-arm in the side of the envelope. With a 5/16-inch x 5-3/4-inch x 0.030-inch Zr specimen in place, temperatures of ~1600 C were attained at 4.8 v and 86 amp. The system was maintained at temperature for 18 hours. (No analytical data were obtained; however, marked grain growth was observed.)

WASTE TREATMENT

Solvent Extraction Removal of Fission Products from Purex FTW

Batch contact and mini-mixer-settler studies were continued to define operating conditions for the extraction of strontium, cerium and promethium from Purex FTW by D2EHPA-TBP-Spray Base solvent. Both study techniques showed that cerium and promethium are extracted more slowly than strontium from acetate buffered, HEDTA complexed feeds. This kinetic effect is much greater for promethium than for cerium. However, the effect is sufficiently great for cerium that cerium losses during extraction runs in the mixer-settler consistently exceeded strontium losses even though equilibrium distribution ratios for cerium were much greater than for strontium under extraction conditions. Experiments to determine if similar kinetics apply to the extraction of cerium and promethium from citrate complexed feeds are in progress.

The oxidation state of chromium in FTW is of interest in this program since chromium(VI) is extracted to an appreciable extent by D2EHPA-TBP-SSB solvents even though complexing agents such as HEDTA are present. In laboratory-scale formaldehyde treatment of 3.8 M HNO_3 - 0.02 M $\text{Na}_2\text{Cr}_2\text{O}_7$ solution, the chromium was reduced to Cr(III) as soon as the first formaldehyde was added. Chromium(III) appears to be stable in the final FTW solution.

Solvent extraction feed solutions, prepared by making synthetic FTW 0.186 M in HEDTA and 0.35 M in sodium acetate, adjusting to pH 4.2 and tracing with Fe-59 and Cr-51, were subjected to Co-60 gamma radiation. Integrated doses of 2.4×10^7 , 4.9×10^7 and 8×10^7 R were given separate portions of the solution. Distribution ratios for iron and chromium between the irradiated solutions and 0.4 M D2EHPA - 0.2 M TBP - SSB were measured. After the highest exposure, iron and chromium distribution ratios were about six times higher than those obtained with a non-irradiated control. However, the distribution ratios were still less than 0.02. It is estimated that the radiation doses given correspond to 10, 20 and 33 hours exposure, respectively, in actual Purex FTW. It is concluded that HEDTA is adequately stable to radiation in FTW to be used as the complexing agent.

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Radiolysis in Strontium Nitrate - Citric Acid - Nitric Acid Solutions

A study of the radiolytic destruction of nitric acid and citric acid and the precipitation of strontium during exposure of strontium nitrate - citric acid - nitric acid solutions to Co-60 gamma radiation has been completed. Solutions containing initially 2.0 M citric acid, 0.075 M $\text{Sr}(\text{NO}_3)_2$ and 2.0, 1.0, 0.5 or zero M HNO_3 were irradiated. Additional nitric acid, equivalent to the amount initially present, was added at exposure intervals of 4×10^8 R in one series and 8×10^8 R in another. Decomposition of citrate was essentially independent of citrate concentration and was increased only slightly by the presence of nitric acid. Average decrease in citrate concentration was 0.4 M for each 4×10^8 R received by the solution. Solids which contained strontium were formed in all of the solutions as radiation proceeded. Also, the pH of all solutions increased. The volume of solids produced, as well as the amount of strontium precipitated, increased markedly with increasing initial nitric acid concentration. Precipitated strontium could be resolubilized in all cases by addition of nitric acid. However, organic solids not soluble in nitric acid were also present in the irradiated solutions.

Product Binding Agents

Elemental sulfur mixed with strontium carbonate filter cake and heated above the melting point (120 C) of the sulfur produces upon cooling a tightly cemented mass. Ten weight percent sulfur gives a lightly cemented cake that is porous and friable. Equal amounts of sulfur and SrCO_3 give a dense, hard, non-porous cake. Above the melting point of the sulfur, the SrCO_3 is still retained, being wet with sulfur, but, if the cake is exposed, sulfur is lost by evaporation and oxidation.

A hard, non-dusting cake of SrCO_3 containing 0.035 gram of LiF per gram of SrCO_3 has been made by sintering a filter cake at 700 C.

Previously reported strontium oxide cake fluxed with lithium fluoride lost 17 percent of its weight between 500 C and 900 C when heated in an inert gas stream. The gas evolved upon acid dissolution of the cake was essentially pure CO_2 . The source of CO_2 in the cake has not yet been definitely determined. In other studies of the LiF fluxed strontium oxide, excellent monolithic castings have been made with this material.

Filtration Canister

In the present prototype loading station, the fission product precipitate is deposited in the canister as a filter cake on the centrally located, stainless steel mesh filter. The precipitate is calcined in place by heating the canister to 800 C. Corrosion and heat damage to the filter mesh have caused failure of the filter after three cycles of calcination of strontium peroxide. Techniques are being sought to permit removal of the filter from the cake during calcination.

The loading station equipment was slightly modified to allow reverse flow of an inert gas through the filter during calcination with the off gas being withdrawn through a line parallel to the slurry feed line. This arrangement failed in operation because the expanding off gas (mostly steam) displaced the filter cake upward, jamming the ports in the loading head. It is apparent that forward flow of the off gas into the filter is necessary to retain the filter cake in its proper position.

Remote Welding

Studies of remote welding techniques continued with the welding of 23 sample cap closures with various joint preparations. Flush, pointed raised face, flat raised face, and "V" groove joints were investigated. The flat raised face and "V" groove are preferred.

Since helium leak detection is proposed for canister inspection, attempts were made to weld under helium atmosphere to avoid the necessity for a separate helium introduction system. It was found that helium shielding gas results reduced tungsten electrode life and gave poor arc stability.

Bulk Waste Packaging

In continued studies of filtration packaging of bulk FTW via ammonia precipitation, filter cake volumes of one cubic foot per ton uranium equivalent were obtained in two runs. A cake thickness of 1.1 inches was easily obtained in the first run. Filtration rate was 20 gph/sq.ft. with a 3 gpm jet. The second run was less encouraging. When cake thickness reached about 1/16 inch, the flow rate was too low to provide slurry transfer. Consequently, a water pulse blow back once every 60 seconds was used to clean the filter. Even with this provision, the overall filtration rate was still low (less than 4 gph/sq.ft.). Filtrate was cloudy throughout the second run. A too rapid precipitation rate is thought to be the cause for unsatisfactory performance.

Analysis of the filtrates from bench scale tests reported last month show that only about 30 percent of the precipitated aluminum was redissolved when the pH of the ammonia precipitated slurry was increased to 12 by caustic addition. Due to the large final waste volumes with this process, no further work is planned.

Stabilization Furnace Concept

Following high temperature stabilization of packaged waste or bulk fission products, the canister must be cooled to some lower temperature suitable for handling. Assuming an inductivity heated canister, a simple method of cooling would be by forced air, flowing downward around the canister to a blower, exhausting through in-cell filters to the ventilation tunnel. For example, a canister 18 inches in diameter, 4 feet long containing 400,000 curies of Sr-90, could be air-cooled from 1600 F to 400 F in less than one hour if an overall heat transfer coefficient of 10 Btu/hr-ft²-F can be obtained. Calculations indicate that for a one inch annulus between the furnace liner and the canister, flow rates of 10, 30, 60 and 80 gfm will result in heat transfer coefficients of 4, 6.6, 9.9, and 10 Btu/hr-ft²-F, respectively. Radiation to the cold furnace liner will also assist in heat dissipation.

Waste Tank Bottom Detector

An ultrasonic depth measuring device was successfully tested in a tank containing water of known depth. Tests with a one-megacycle barium titanate transducer were designed to determine the effects of crystal orientation and large liquid volumes on the reflected sound pulses in a non-radioactive environment, prior to similar tests in a Purex waste storage tank. After numerous instrumentation adjustments, measurements with an accuracy of 1 to 2 percent were obtained for a ten foot depth.

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Several high temperature transducers were obtained and preliminary laboratory evaluations were completed. Specifications are being prepared for a lower frequency (20 to 50 kc) depth sounder which is expected to have the test combination of performance characteristics for determining sludge depth in waste tanks.

Cesium Removal and Purification from Purex Formaldehyde-Treated Waste

A flowsheet for Cs-137 removal and purification from Purex formaldehyde-treated waste was conceived and studied. The process involves sorption of Cs-137 and sodium on clinoptilolite, removal of the sodium with dilute nitric acid, elution of the cesium with 2 M ammonium acetate, and volatilization of the ammonium acetate (by boiling) to leave a solid residue of cesium salt.

Previous laboratory experiments indicated that a column of clinoptilolite would remove the cesium from 100 column volumes of 82 gal/ton "1965" FTW to 50 percent breakthrough or 80 column volumes to one percent breakthrough. Recent research shows that 98 percent of the sodium on the bed can be removed with 10 column volumes of 0.25 M HNO_3 . (At equilibrium there is about 8 times as much sodium on the bed as there is cesium.) Less than one percent of the cesium is lost in the nitric acid wash. The loss of Cs-137 could be further reduced by loading only to the initial cesium breakthrough, and by maintaining same direction of wash flow as used for loading. More than 97 percent of the Cs-137 can be eluted with about 30 column volumes of 2 M ammonium acetate.

More than 99 percent of the ammonium acetate was volatilized in the temperature range 100-200 C. Less than 0.3 percent of the Cs-137 was found in the condensate. Ammonium carbonate solutions can also be used for elution and will volatilize at low temperature.

Cesium Removal with Ammonium Phosphomolybdate

A method of reducing the mole ratio of sodium to cesium in high-salt wastes with ammonium phosphomolybdate (APM) was investigated. A simulated Purex high-level waste supernatant solution was diluted 1:1 with water, and 8 g of APM and 14.3 ml of 10 M NH_4NO_3 were added per liter of diluted waste. The APM was precipitated along with more than 95 percent of the cesium by the addition of 200 ml of 70 percent HNO_3 per liter of waste. This was the minimum amount of acid required to precipitate the APM. The APM settled well and was filtered on a medium porosity sintered glass filter. The APM was then dissolved in a small volume of dilute sodium hydroxide solution.

The cesium in this solution was loaded on columns of clinoptilolite and Decalso. This method increased cesium loadings on clinoptilolite and Decalso by factors of 10-20 over that of the untreated waste. The experiments also indicated that the effluent from the clinoptilolite or Decalso columns could be used as a source of APM for subsequent batches of waste.

TRANSURANIC ELEMENT AND FISSION PRODUCT RECOVERY

Assistance to Purex Head-End Fission Product Recovery

In late February and early March, during change-over from conventional LW to formaldehyde treatment, difficulties were encountered in processing strontium

through the Purex head-end equipment. A series of runs, using the same lead-carrier flowsheet which had worked well before, produced essentially no product (average recoveries below 20 percent). The indication was that precipitation occurred normally, but that the precipitate dissolved during the in-centrifuge caustic-carbonate metathesis. No operational cause for this behavior could be found and minor flowsheet changes failed to correct the difficulty, which was suspected in some way to be due to formaldehyde.

A series of B-Cell studies was accordingly initiated with plant feed and following the plant flowsheet as closely as possible. The plant behavior was duplicated and was found to be due to the formation, during the sulfate precipitation step, of a very bulky (10X normal volume) iron and tartrate containing precipitate. This largely dissolved on subsequent treatment with caustic-carbonate solution to form a strongly complexing solution which dissolved the sulfate precipitate. The cause was traced to an abnormally high iron content in the feed (0.7 M vice 0.3 to 0.5 M) coupled with a high precipitation pH of about 2.5. Precipitation at a lower pH eliminated the trouble and restored plant strontium yields through the carbonate product to 80 percent or better. The reason for the high iron content is not known.

Other B-Cell runs during the month were aimed at better defining conditions for precipitating rare earth oxalates from Purex waste (for rare earth recovery or as a first step in an improved strontium recovery flowsheet). With proper pH and oxalate concentration, about 90 percent of the cerium was precipitated, leaving the strontium in solution. Behavior of promethium was almost identical to that of cerium. In supporting laboratory studies, neither peroxide nor nitrite had any affect on the oxalate precipitation of either cerium or promethium.

A report, "Peroxyacetate Separation of Cerium(IV) from Trivalent Rare Earths in Purex Process Solutions," by L.F. Lust and H.H. Van Tuyl, HW-73135, was written summarizing the hot cell work on the peroxyacetate process.

Technetium Recovery

Analytical work is in progress to define the path of technetium-99 in the Hanford Redox and Purex plants. Samples from most of the sampling points in Redox have been analyzed but fail to give a satisfactory material balance across the process, perhaps because no samples of organic streams or of solvent recovery wastes were taken. It is clear that little or no technetium leaves the plant with the uranium or plutonium products and that it must, therefore, be lost either to the aqueous or organic wastes. Further samples are being obtained to clarify the matter.

Many of the Purex samples are from organic streams, and a method has been devised to quantitatively transfer the technetium to an aqueous phase to allow standard methods of analysis to be used. The sample (TBP in Shell Spray Base) is passed through a cation exchange column, which absorbs uranium, plutonium, and most of the fission products, and then through an anion column to absorb the technetium. The columns are pretreated with methanol to remove water from the resin. After absorbing the sample, they are thoroughly washed again with methanol to remove all traces of TBP and diluent, the technetium is then eluted with 8 M HNO_3 .

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A report was written summarizing the laboratory development and pilot plant demonstration of the ion exchange process which was developed for recovering purified technetium from Purex alkaline waste tank supernate. The title is "Recovery and Purification of Technetium-99 from Hanford Wastes," F.P. Roberts, F.M. Smith and E.J. Wheelwright (HW-73121).

A-Cell Decontamination

A-Cell decontamination work was started on February 12. Two weeks were spent removing waste and general cleanup prior to cell flushing operations. On February 26, the cells were officially released and general flushing operations were started. The cell has been given a series of nine surface flushes. All tanks, columns and lines were flushed repeatedly during this period. Approximately 3,000 gallons of flush solution were used during the flushing process prior to opening the cell door.

On March 20, A-Cell door was opened, a general reading of one rad per hour was encountered, with points up to 100 rads per hour. These high readings were in areas which were inaccessible to in-cell spray equipment or in areas where faulty valves were located, making it impossible to flush either the valves or associated piping. Removal of faulty valves and flushing from the rear face reduced the general reading to approximately 500 mrad per hour.

Almost all of the in-cell equipment has been removed. All work is progressing satisfactorily. Experience in decontaminating A-Cell allows several conclusions which may be of value in the design of future cells and equipment. Most important was that the stainless steel cell liners and equipment cleaned up very well and greatly shortened the time required to decontaminate the cell. Use of substitute materials, and particularly painted mild steel, proved a false economy. (The column support racks were constructed of "unichromed" mild steel, became quite hot, did not decontaminate well, and were difficult to dispose of.) The linear polyethylene gaskets in the Lapp feed pump and columns were in excellent condition, as was the stainless steel diaphragm of the pump, which appeared to be as good as when installed.

Cesium Solvent Extraction

The extractability of rubidium, potassium and sodium by DPA-NB (nitrobenzene) solutions was measured by use of the appropriate radioactive tracers. The separation factors of cesium from rubidium, potassium, and sodium were 5.4, 28, and 2545, respectively. Although the extraction behavior of rubidium and potassium more nearly resembles that of cesium than that of sodium, the differences are great enough that it should be fairly simple to separate cesium effectively from rubidium and potassium in counter-current equipment.

Although present formaldehyde treated waste (FTW) is very nearly identical in composition to 1965 FTW, except for a modest decrease in nitric acid content, future ("1965") FTW will reflect the use of uranium(IV) reductant, titanium tube bundles, and other changes which should markedly reduce its salt content. Such a solution is more favorable for the extraction of cesium and other fission products (mainly because of lower sodium and iron content) than is present waste. Cesium extraction coefficients from (tartrate complexed) 1965 FTW and from the Semiworks waste which would result from using 1965 FTW (and which would also

contain acetate, citrate and HEDTA) were identical and were higher than those obtained from current FTW by a factor of over three -- about what one would expect on the basis of sodium content. The extraction coefficient into 0.02 M DPA-NB was 23, high enough for a batch operation to be practical. Use of a higher DPA concentration or dilution of the aqueous would result in even higher extraction coefficients.

Although the above experiments suggest the feasibility of recovering cesium by DPA extraction of the waste from D2EHPA strontium recovery, other experiments showed that both D2EHPA and TBP interfere seriously with the extraction. These materials at a concentration of 0.015 M lowered the extraction coefficients of cesium into 0.01 M DPA-NB by factors of three to six. Thus, it is essential that these impurities be absent, and extraction of cesium prior to strontium would be preferable. Experiments are in progress to determine whether cesium can be extracted from strontium-extraction feed without adjusting its composition or pH.

Kinetic experiments showed no rate problem in either the extraction or stripping reactions. In both cases, the cesium partition coefficients achieve over 90 percent of the equilibrium value in less than half a minute.

The solubilities of the DPA salts in nitrobenzene and of nitrobenzene and DPA in the various aqueous waste compositions are being measured, because of the importance of these data in determining solvent losses and possible storage problems, etc. The solubilities of HDPA, NaDPA, KDPA, and CsDPA in nitrobenzene were 0.03, 0.21, 0.056 and 0.086 M, respectively. Implication of these data is that a more concentrated and "potent" DPA-NB extractant might be prepared by using sodium salt rather than the hydrogen form of DPA. Thus, the extraction coefficient of sodium from 103A supernate into saturated (0.21 M) NaDPA-NB was 23, compared to a value of 4.4 into 0.025 M DPA-NB. The solubility of nitrobenzene in aqueous solutions was determined using a spectrophotometric technique. Solubility in water was 1.8 g/l, in good agreement with literature values. Solubility in dilute acid (≤ 1 M) was about the same, but increased at higher acidities. The solubility of nitrobenzene in alkaline solutions of high sodium content (such as 103A supernate) was lower by a factor of ten or more, a surprising behavior which, if confirmed, indicates that extraction losses may be very low indeed. Aqueous DPA solubilities have not yet been measured, but are believed to be quite low.

Compounds believed to be o -CF₃ - phenylpicrylamine and o -nitrophenylpicrylamine were synthesized during the month by a somewhat different method than used previously. Cesium extraction with "these" compounds is much higher than with the earlier ones. Cesium extraction with the o -CF₃ compound approached that of DPA and solubility of the hydrogen form in nitrobenzene was over three times that of DPA. However, both the o -CF₃ and o -nitro extractants required a higher pH for effective cesium extraction than did DPA. These results cast doubt on earlier measurements with synthetic compounds. A proof of structure is being sought.

Extraction of Np(IV) from Purex Formaldehyde Treated Waste with D2EHPA

Studies of the feasibility of recovering neptunium from Purex plant FTW by a single batch extraction with a D2EHPA - TBP - diluent solvent continue to show

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promise. Over 97 percent of the neptunium in a synthetic FTW solution containing 3×10^{-5} M Np(IV) and 0.5 M hydrazine was extracted in a single contact with one-tenth volume of 0.04 M D2EHPA - 0.02 M TBP - Shell Spray Base. Under these conditions, little, if any, extraction of contaminating monovalent and divalent cations occurred. For the trivalent ions Ce(III), Cr(III) and Fe(III), decontamination factors were 4×10^4 , 800 and 200, respectively. Decontamination factors for ruthenium and zirconium were 2.4×10^4 and 3.6, respectively. These latter two factors should not be accepted without some reservation because of possible differences in chemical species between spiked isotopes and plant solutions.

Cesium 12-Tungstophosphate

Cesium 12-tungstophosphate is an attractive candidate for storing or shipping radiocesium. $\text{Cs}_2\text{HP}(\text{W}_{12}\text{O}_{40}) \cdot 15\text{H}_2\text{O}$ is precipitated nearly quantitatively by addition of 100 to 200 percent excess 12-tungstophosphoric acid to an acidic cesium solution. The waters of hydration can be removed by heating and the anhydrous cesium 12-tungstophosphate is thermally stable to at least 800 C. Precipitated cesium 12-tungstophosphate is finely divided--nearly colloidal in nature. It settles slowly and is filterable only on the finest filters at very slow rates. The filter cake has a bulk density of 5.5 g/cc, containing about 0.5 g Cs/cc.

The precipitate can be flocculated by slow addition of a 0.1 percent solution of certain flocculating agents (Separan 2610 from the Dow Chemical Company or Aerofloc 550 from the American Cyanamid Company) to the extent of about 0.03 grams of flocculating agent per gram of cesium in the precipitate. The flocculated precipitate settles rapidly but is still difficult to filter because of easy re-dispersal of the precipitate.

The cesium 12-tungstophosphate precipitate has been agglomerated by the addition of a one percent gelatin solution to the extent of about two grams of gelatin per gram of cesium in the precipitate followed by digestion at 100 C for 15 minutes. This agglomerated precipitate does not settle, but is easily filterable on a vacuum filter with a thin precoat of diatomaceous earth filter aid. Future work will be directed toward decreasing the amount of gelatin required and/or finding some other pre-filtration beneficiation technique for the precipitate.

ANALYTICAL AND INSTRUMENTAL CHEMISTRY

X-Ray Spectroscopy

The possibility of using X-ray spectrographic determinations of aluminum and silicon in the Al-Si canning bath was evaluated. It was found that problems with non-homogeneity within a sample and with sample preparations are such that the desired accuracy cannot be obtained by this method.

EQUIPMENT AND MATERIALS

Hydraulic Equipment

A canned motor pump proposed for jumper installation operated smoothly and without incident for over 1100 hours pumping hexone. On inspection, no measurable wear of the bearings or journals was noted.

A special tapered shaft pump developed for Hot Semiworks application operated smoothly and quietly for 270 hours at a speed of 1800 rpm, pumping water at a controlled rate of 2 gpm against a head of nine feet. Some slight bearing noise indicated a possibly damaged lower bearing, but no untoward effects were noted on pump behavior. However, after the motor windings were replaced and the speed of the pump was doubled to develop additional head, the bearing failed after approximately two hours. New bearings have been ordered and will be installed on receipt. Operation prior to failure was relatively smooth with the pump delivering 2 gpm against a head of 55 feet. Additional head and capacity are expected with installation of a fully shrouded impeller to be designed and constructed. This impeller will also be installed at the same time as the new bearings.

Mechanical Equipment

In search of radiation resistant jumper gasketing material for B-Plant service, additional tests have been performed to determine linear polyethylene's resistance to flowing at 75 C. Two tests were performed in which an 0.08 inch thick, two inch pipe size gasket was closed in a flange with 250 ft.lb. torque. The gasket was subjected to 75 C for 24 hours. While still hot, the torque required to increase the pressure on the gasket was measured as 75 ft.lb. in test No. 1 and 87 ft.lb. in test No. 2. A Teflon® gasket was then subjected to the same test and at the end of 24 hours the torque required to increase the pressure on the gasket was 240 ft.lb. Linear polyethylene use in the B-Plant connector would be unsuitable as a Teflon substitute for temperatures over room temperature.

Sonic Energy Application

In continuation of studies reported last month, new information indicates the output of a modified stem-jet whistle to be about 152 db whereas the output of a commercial device was determined to be about 168 db. The commercial whistle, however, had a higher air consumption rate and was found to be only 73 percent as efficient as the modified stem-jet with relation to foam destruction, being able to destroy about 29 cfm of dense detergent foam as generated in an eight inch diameter tube.

Stress Cracking of Mild Steel in Alkaline Nitrate Systems

Stress cracking of mild steel in waste storage tanks at SRP has prompted a study of potential stress cracking of mild steel in HAP0 waste tank environments and means of preventing such attack. Initial studies have been toward prevention of stress cracking by applied cathodic currents. Polarization curves have been determined for mild steel in contact with solutions simulating several HAP0 stored wastes and with solutions commonly used in stress cracking studies. Incidence of cracking in stressed mild steel samples in contact with nitrate solutions and subjected to anodic and cathodic current flow is also under study. Stressed and notched C-ring corrosion test samples have been prepared from 1.5 inch 1020 steel pipe and are currently being exposed to test solutions. Three plates of mild steel, three feet square and 0.5 inch thick, have been ordered.

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These plates will be welded under conditions conducive to high stress levels and exposed to test waste solutions. Two of the plates will be given a spheroidizing heat treatment to increase stress cracking tendency. Cadmium rods will be attached to one plate parallel to weld beads to study the effect of coupling.

Corrosion of 304-L in Concentrated $\text{UNH-Al(NO}_3)_3$ Solutions

Sensitized 304-L coupons were exposed in the liquid, vapor and interface of boiling solutions of 3.5 M UNH - 2.3 M $\text{Al(NO}_3)_3$ with nitric acid concentrations of + 0.1, -0.2 and -0.6 M. Samples were exposed for single periods ranging from 237 to 397 hours. Corrosion rates ranging from 0.1 to 4.7 mils/mo were observed. The maximum rate occurred on the liquid phase sample in the 0.6 M acid deficient solution. Corrosive attack here was of a very selective intergranular nature.

Corrosion of Titanium and Tantalum by Purex Decontaminating Solutions

Samples of A-55 titanium and commercially pure tantalum were exposed at the interface to the following commonly used decontaminating solutions: caustic tartrate, oxalic acid - nitric acid, alkaline permanganate and nitric acid - ferrous sulfamate. The only samples which exhibited measurable corrosion rates were the titanium samples exposed to oxalic acid - nitric acid and nitric acid - ferrous sulfamate solutions. In no instance was the observed corrosion rate greater than 0.5 mils/mo.

PROCESS CONTROL DEVELOPMENT

C-Column Mathematical Model

A mathematical model describing the operation of the C-Column solvent extraction system has been selected for extensive study. This basic model, consisting of two simultaneous differential equations, has the necessary characteristics for adequate fitting of the experimental data obtained to date. Three unknown parameters are present: an aqueous backmixing coefficient (or longitudinal dispersion coefficient), a mass transfer coefficient, and an empirical exponent to which the organic phase volumetric fraction is raised. Variables in the equations are flow rate and uranium concentration of each phase, volumetric fraction of organic, equilibrium uranium concentrations, and distance along the column.

Work has been concentrated on determining numerical values for the unknown parameters of the model such that the equation accurately describes the observed performance of the experimental column. Also, simplifications of the basic model are being investigated to obtain the best form for use with computer control. Initial values of the parameters were obtained using analog techniques; subsequent refinement of the values is obtained by an iterative technique on the IBM-7090. The 7090 routine to fit the data to the model using a Gaussian least squares method was improved with the result that a complete iteration can be performed in 30 percent of the time previously needed.

REACTOR DEVELOPMENT - 04 PROGRAMPLUTONIUM RECYCLE PROGRAMSalt Cycle Process

Precipitation Reactions in Molten Chloride Salt Solutions - Laboratory studies of the precipitation of single and mixed oxides from molten chloride salt solutions have been continued, with results as follow:

1. Plutonium Oxide

Two different methods have been used for the precipitation of plutonium oxide from 2.6 LiCl-KCl (1 w/o PuCl_3) at 575 C. Sparging the melt with 55 percent O_2 - 45 percent Cl_2 produced a black material with discernible crystal faces. Using a water-saturated helium sparge, an olive-green compound was formed. Both were shown by X-ray diffraction studies to be PuO_2 , the black oxide being formed of larger crystallites than the other.

2. Plutonium-Uranium Oxides

Mixed plutonium and uranium oxides have been precipitated from 2.6 LiCl-KCl (4.3 w/o Pu(III), 1.4 w/o U(IV)) at 575 C, by sparging with water-saturated helium. X-ray diffraction studies showed the black, finely divided solid to be a solid solution of PuO_2 and UO_2 , with a plutonium/uranium ratio of about 5/3.

3. Plutonium-Thorium Oxides

Solid solutions of the dioxides of plutonium and thorium have been prepared by sparging a 2.6 LiCl-KCl solution of their chlorides with an O_2 - Cl_2 mixture. With a melt containing 1 w/o Pu and 5 w/o Th, the product was green-colored and had a plutonium/thorium ratio of 0.4. Precipitation from a melt initially containing 2 w/o Pu and 2 w/o Th gave what appeared to be a mixture of two PuO_2 - ThO_2 compounds. As estimated from X-ray diffraction peak angles, these compounds had plutonium/thorium ratios of four and one. This mixture was black in color.

4. Uranium-Thorium Oxides

A solid solution of about 9 w/o UO_2 in ThO_2 was prepared from a solution of thorium and uranium(IV) chlorides in LiCl-KCl. The product, formed by the action of a moist helium sparge, was dirty gray in color and extremely finely divided.

Co-Deposition of Uranium and Thorium Dioxides - Oxide preparations containing from 1.85 to 48 w/o ThO_2 in UO_2 have been made by electrolytic codeposition from LiCl-KCl solutions of UO_2Cl_2 and ThCl_4 . The higher ThO_2 concentrations (above 14 w/o) were achieved only with the use of a Cl_2 or HCl gas sparge during the electrolysis. Cl_2 was found to be more effective than HCl, probably because it reacts more readily with UO_2 than with ThO_2 , whereas HCl attacks both. The deposits produced in the gas-sparged melts were comprised of fine needle-like whiskers, with no voids visible in a cross-section cut.

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Cerium(IV) Stabilization by Complexing Ions - A study has been made of the effect of varying concentrations of fluoride and sulfate ions upon the relative stability of the tri- and tetravalent states of cerium (as a stand-in for plutonium) in molten NaCl-KCl (1.2 w/o Ce(III)). Using Cl_2 gas as an oxidant, it was found necessary to add enough fluoride to give a fluoride/cerium ratio of four or greater before cerium(IV) could be detected. The fluoride effect was, however, nullified by the presence of uranium in the melt. Sulfate ion in molar concentrations up to ten times the cerium was ineffective in stabilizing cerium(IV).

X-Ray Diffraction Studies - Since there is disagreement in the literature over the value of the lattice constants for the face-centered cubic structures of ThO_2 , UO_2 and PuO_2 , standard X-ray diffraction patterns were obtained for nearly stoichiometric samples prepared here. The lattice constants for these preparations were measured as 5.598, 5.469 and 5.395, respectively, and were used in further studies of preparations of the mixed oxides of thorium, uranium and plutonium. It was observed in these studies that UO_2 and ThO_2 apparently are completely miscible with each other. Results of work on the PuO_2 -containing mixtures are summarized in the section on precipitation reactions in molten salts. In all this work, X-ray fluorescence measurements were useful in providing a rapid estimate of the relative uranium-thorium-plutonium contents of the mixture.

Engineering Process Development - Electrolysis studies with a KCl-LiCl melt contained in a 2 liter pot were carried out at 535 C. The starting material, $\text{UO}_2\text{Cl}_2\text{-H}_2\text{O}$, was dried in the melt with HCl at a flow rate of 280 cc/min for 24 hours. Duranickel electrodes were used with a potential of 1.5 VDC and approximately 5 amps. Current efficiencies for a 16-hour period averaged 56 percent with current densities of 0.75 - 1.0 amps/in². During this period severe corrosion of the electrodes at the vapor-liquid interface was observed.

Four UO_2 deposits were made with starting uranium concentration (calculated) of 30, 26, 24.6 and 19.0 percent, and using a HCl sparge to insure a dry melt. Each deposit was extremely dense with large, flat crystal faces. Crystal size decreased with initial uranium concentration. No laboratory analysis of the UO_2 has been obtained; however, streak tests indicated an O/U ratio of 2.0 - 2.01 for the first three deposits, and 2.01 - 2.02 for the fourth.

Alternating 60 cycle current for heating purposes was applied across two 1 inch x 1/4 inch graphite electrodes immersed 3-1/2 inches in a melt similar to those described above. After 16 hours at 4 VAC and 38 amps, no UO_2 deposition was observed. In a similar run a direct current of 1.5 volts and 8 amps was superimposed on the electrodes carrying AC. A UO_2 deposit was obtained which was physically similar to those obtained with direct current only. These results encourage the consideration of alternating current as a method of heating molten salts while electrodepositing UO_2 .

Process Control Instrumentation - The concentration of uranium in its several valence states is an important variable in the salt cycle process. As part of a program to develop salt cycle process control systems, an instrument has been devised to measure concentration of uranium(IV) and the combined concentrations of uranium(III) and (VI). A probe type filter photometer has been fabricated, consisting of a quartz tube and prism arrangement. Light is

projected down the tube, through the molten salt, through a reflecting prism and back up the tube when the resultant intensity in various spectral bands is measured. Absorption in the regions of 1100 and 500 μ m are used to determine, respectively, uranium(IV) and the combined value of uranium(III) and (VI). The special interference filters needed have been obtained and tested. Following assembly of the instrument components, the photometer will be tested and calibrated in a small pot containing molten salts with known uranium concentrations. The instrument should be adaptable for continuous analysis of constituents other than uranium.

Dissolution of PRTR Fuels

Irradiated PRTR spike fuel alloy shows markedly different behavior toward activation and dissolution in nitric acid - mercury(II) nitrate systems than has been observed with the non-irradiated alloy. Decladding and dissolution of a 5-inch segment of irradiated spike fuel (Zircaloy-4 clad plutonium-aluminum-nickel alloy) was done in the 222-S high level cubicle. The segment was from a fuel assembly which had received a total exposure of 45 MWD per assembly or 60 percent of goal. Removal of the Zircaloy-4 jacket with ammonium fluoride-ammonium nitrate proceeded as expected. However, the declad fuel alloy failed to activate and dissolve rapidly in nitric acid-mercury(II) nitrate solutions, with or without ammonium fluoride, under conditions which readily produced activation and rapid dissolution of similarly declad non-irradiated alloy.

Activation of the irradiated alloy was achieved at low acid (1 M) but the conditions may not be appropriate for use in Redox because of increased hydrogen evolution. Reasons for the differing behavior of irradiated and non-irradiated alloys and means for assuring activation of the irradiated alloy under conditions acceptable for Redox are being explored in further tests.

UO₂ Leaching from PRTR Fuel Pieces

Nitric acid leaching of UO₂ from cut pieces of Zircaloy-clad, swaged, PRTR fuel rods has been investigated during routine pilot plant dissolution of uranium metal. The pieces of UO₂ fuel rods were included in three different dissolver runs. In all cases the length of the UO₂ core was 22 inches and the inside diameter of the Zircaloy cladding was 1/2 inch.

In the first run, two pieces of fuel rod with the UO₂ exposed at both ends were placed in the dissolver along with 1200 pounds of uranium metal. One piece was placed at a 45 degree angle to the dissolver bottom and the other piece was positioned horizontally. During the dissolution cycle, the pieces were completely immersed in boiling dissolver solution (about 300 liters) for 10 hours. The initial concentration of the dissolvent was 9 M HNO₃. The final concentration of the dissolver solution was 2.26 M U and 1.2 M HNO₃. Inspection of the Zircaloy tubes showed complete removal of the UO₂ cores.

During a shutdown about midway through a second dissolver run, a piece of fuel rod with the UO₂ exposed at both ends was immersed horizontally in the dissolver solution. After immersion, the dissolver solution was maintained at boiling for five hours. The initial dissolver solution concentration was 1.6 M U and 3 M HNO₃ and the final concentration was 2.3 M U and 1 M HNO₃. Inspection of the Zircaloy tubing showed that one end had been placed against

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the dissolver coil. The UO_2 was removed to a depth of 10.5 inches from the unobstructed end and to a depth of 1.5 inches from the end partially blinded by the dissolver coil.

In a third run, two pieces of fuel rod with the UO_2 exposed at only one end were placed in the dissolver with 500 pounds of uranium metal. One piece was positioned vertically and the other horizontally. The exposed UO_2 was at the top of the vertical piece. During the dissolver cycle the pieces of fuel rod were completely immersed in boiling dissolver solution (about 300 l) for eight hours. The initial concentration of the dissolvent was 12 M HNO_3 . The final concentration of the dissolver solution was 2.1 M U and 1.0 M HNO_3 . Inspection of the Zircaloy tubes showed complete removal of the UO_2 cores.

RADIOACTIVE RESIDUE FIXATION

In-Cell Calciner

Testing and minor modification of the mock-up of the in-cell calciner continued during the month. Steady-state temperature profiles were determined and found to agree closely with design calculations. A run with simulated Purex FTW went well except for excessive air in-leakage in the off-gas system, which will have to be remedied. The leak in the water cooling system of the Tocco unit was finally repaired, the unit connected to the melt-pot induction coil through a coaxial cable, and the combination tested. About 9 kw, or 60 percent of rated power was delivered. While this may be adequate, an attempt is being made to improve the situation by constructing a new induction coil with a different number of turns. The problem of how to seal the pots to the column is believed to have been solved. The initial plan was to use a bolted flange connection; however, this would be difficult to make and break with the manipulators. Modified Marmon "Conoseal" flanges, but with a "Flexitallic" rather than a "Conoseal" gasket, have been tried and appear very satisfactory. The bottom of the column is being modified to use this type of flange. The pot will be secured with a one-bolt clamp which can be readily opened or closed with a small impact wrench. Remote welding experiments indicate that pots with this type flange can be permanently sealed (after filling) by remote welding.

More details of the in-cell equipment and photographs of the mock-up are included in the "Radioactive Residue Fixation Quarterly Report for January, February, March, 1962."

Synthetic Zeolites

Two acid-resistant, synthetic zeolites, Zeolon and Linde AW-500, recently became available commercially in pelletized form. Zeolon is a synthetic mordenite ($2\text{H}^+ \cdot \text{Al}_2\text{O}_3 \cdot 10\text{SiO}_2 \cdot 6\text{H}_2\text{O}$) and Linde AW-500, a synthetic chabazite ($\text{Na}_2\text{O} \cdot \text{Al}_2\text{O}_3 \cdot 4\text{SiO}_2 \cdot 4\text{H}_2\text{O}$). Cesium kinetics and capacities of H^+ -based Zeolon and Na^+ -based Linde AW-500 were determined and compared with clinoptilolite with the following results:

1. at pH 3.0, Linde AW-500 cesium capacity was twice that of clinoptilolite with an influent containing 0.2 N $\text{CsCl} + \text{HCl} + \text{Cs-134}$; Zeolon cesium capacity was forty percent greater than clinoptilolite with the same influents;

2. Linde AW-500, in the particle diffusion region of cesium concentration, loads at about three times the rate for clinoptilolite; Zeolon and clinoptilolite cesium loading rates are comparable under the same conditions; and
3. at pH 1.4, the cesium loading rates of clinoptilolite and Linde AW-500 were unchanged from those at pH 3.0; under the same circumstances, Zeolon cesium loading rates were decreased to one-half that of clinoptilolite.

Linde AW-500 and its binder are not guaranteed by the manufacturer to be stable in solutions of pH less than 2.5 (Linde Division of Union Carbide Corporation). Zeolon, on the other hand, was designed for use as an acid catalyst (by the Norton Company) and is therefore quite stable in up to 50 percent acid solutions. The cesium selectivity of the three zeolites will be measured, in multi-cation containing solutions, and compared.

Intermediate-Level Wastes

Laboratory studies show that ion exchange gels, prepared by mixing a small amount of hydrous cobaltous cobalticyanide with hydrous zirconium phosphate, have greater cesium and strontium selectivity than zirconium phosphate gel alone. A mixed gel having a mole ratio of five zirconium to one cobalticyanide gave a cesium K_d of 6×10^4 compared to 835 for zirconium phosphate alone. In column experiments with trace cesium and strontium in $0.012 \text{ M } \text{NH}_4\text{NO}_3$ the mixed gel sorbed more than ten times as much cesium and strontium as zirconium phosphate.

Condensate Treatment

Analysis of the condensate collected during the steam regeneration at 250 F of the activated carbon from MPP Run 22 indicates that only about 25 percent of the hydrocarbons and none of the tri-n-butyl phosphate were removed from the bed. Probably the organic material which was removed by the regeneration procedure was from the interstices of the bed, originally retained by filtration rather than adsorption. Much higher steam temperatures are thought to be needed to effectively remove the adsorbed organic. An attempt to use a commercial grade of Anthrafilt (a crushed and screened anthracite coal) as filter media in the form of a packed bed proved unsuccessful because of early plugging of the bed. The column plugged due to cake formation at the top of the column.

Several adsorbents were studied for removal of trace ruthenium from Purex tank farm condensate which had been put through a cation exchange column. Duolite S-10 resin which contained metallic copper gave an initial ruthenium DF of 10 but fell to about two in a few hundred column volumes. Iron-based Amberlite IR-120 and iron filings mixed with crushed calcite gave ruthenium DF's of about two. Essentially no ruthenium was removed by the mineral olivine, or by an anion exchange resin.

Release of Radiocesium from Clinoptilolite

The release of adsorbed Cs-137 from clinoptilolite at elevated temperatures was shown to be linear over an eight hour period for temperatures below 1100 C.

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This linear release indicates that surface loss of radiocesium from the clinoptilolite granules was controlling for temperatures below the fusion point. The release rate was approximately 6×10^{-5} percent per hour at 590 C. The equation for calculation of the percent release for temperatures below 1100 C is:

$$R = 875 t \cdot e^{-\left[\frac{1.416 \times 10^4}{T}\right]}$$

where: R = percent release

t = hours

T = temperature, Kelvin.

Above the melting point of clinoptilolite at 1350 C, Cs-137 release followed the parabolic rate law which indicates the release is controlled by diffusion. For the special case when release occurs from the top surface of a disc with an area to volume ratio of $14 \text{ cm}^2/\text{cm}^3$, the percent release R' per cm^2 in t hours followed the equation:

$$R' = 0.35 t^{1/2}$$

From rough estimates of surface to volume ratios of the solidified melts, the apparent diffusion coefficient at 1350 C was calculated to be in the order of $10^{-10} \text{ cm}^2/\text{sec}$.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Geology and Hydrology

A magnetic tape record file for Hanford monitoring well data was outlined for the Electronic Data Processing Operation. The file will contain radioisotopic analysis histories, well locations, depths, and other physical information pertinent to the evaluation of ground water contamination and hydrology problems. This record should measurably speed up the scanning of data necessary for preparing routinely issued maps and reports. The tape will also be searched for input data essential to the calculation of project permeabilities.

The four Priest Rapids basalts flows (immediately underlying the Priest Rapids Dam) and the overlying Beverly Member of the lower part of the Ellensburg Formation were identified in the White Bluffs area from deep well logs. Tracing of these rock units westward beneath the Hanford Works toward their previously known occurrences in Umtanum Ridge will help define the extent and continuity of the Saddle Mountain basalt flows that lie above the Beverly Member. The irregularity of the Saddle Mountain basalt flows where studied at Priest Rapids indicates that their hydrologic characteristics may significantly complicate those of the Ringold Formation where it overlies those basalt flows.

Field Apparatus Development and Application

The 2000 cycle band pass filter for the magnetometer was obtained from the U.S.G.S. Geophysics laboratory and installed in the magnetometer converter. Equipment

operation now appears satisfactory. With assistance from the Nuclear Instrument Development Unit, an audio-frequency amplifier and control circuitry were developed to modify the magnetometer converter for geophysical use. A power supply unit for the magnetometer was designed and is being fabricated. Also, an enclosure for the magnetometer head is being designed to permit extending the sense coils behind a vehicle.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Radiation Chemistry

Recent literature data, together with our protection indices data, provide information from which absolute rate constants can be calculated for reactions between hydroxyl radical and all the compounds whose protection indices have been evaluated. The rate constant for the reaction between hydroxyl radicals and the erioglaucine dye anion was calculated to be $4.5 \times 10^9 \text{ l mole}^{-1} \text{ sec}^{-1}$, indicating that the reaction is diffusion controlled. Since this reaction occurs as fast as diffusion will allow, more reactive scavengers of the hydroxyl radical cannot be expected to be found. Emphasis will now be placed on searching for less toxic compounds which diffuse more rapidly.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition in Conduits

Deposition of ZnS in a 60-foot long, 1/2-inch diameter tube was measured as a function of average velocity in the tube. With previously measured size distributions for the particles used, a comparison was made between actual and predicted deposition on the tube wall. The predicted deposition was that calculated from a regression equation developed in an earlier study. The following table gives the predicted and actual deposition for several velocities.

<u>Velocity</u> <u>cm/sec</u>	<u>Percent Deposited</u>	
	<u>Predicted</u>	<u>Observed</u>
585	51	62
740	62	74
900	72	85
1030	76	88
1180	82	88
1470	88	93

The agreement is good considering the assumptions which were made regarding the size distribution of the particulate material entering the tube.

The above velocities are well into the turbulent range. A few experiments in a laminar flow indicate that deposition will be much lower, perhaps negligible. Further experiments are underway to obtain firm deposition data in this region of flow.

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Aerosol Generation Techniques

Progress was made toward design and assembling of parts for a spinning disc aerosol generator. The spinning disc generator should materially assist in particle studies through permitting fairly narrow size ranges of particles to be prepared, a vital need in particle research.

Manager
Chemical Research and Development

WH Reas:cf

BIOLOGY OPERATION

A. ORGANIZATION AND PERSONNEL

R. L. Uhler was appointed Acting Manager of Plant Nutrition and Microbiology effective March 1, 1962.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

A study was started at the 1706-KE laboratory testing the effect of reactor effluent on migrating-size chinook salmon fingerling. Percentages of 3, 5, and 7 are being tested. The largest number of young salmon occurring in this section of the Columbia River are reared in the river's main stem and tributary streams above HAPO. They are exposed to Hanford effluents when they pass downstream on their migration to the ocean. The major migration occurs during April to July.

Columnaris

A series of tests have been started to determine if strains of columnaris collected from last year's population density study show an increase in virulence as the season progressed. Preliminary data from eight different strains collected from 7/31/61 to 10/10/61 show that the strains were virulent from the start of the season. No increase in virulence between early and mid-season strains have been noted.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Phosphorus-32

Preliminary data have indicated the feasibility of utilizing whole-body monitoring for estimating P^{32} burdens in miniature swine. The counting rate per μc for P^{32} in the pig is less than that obtained for Sr^{90} . The difference in counting rate was attributed to differences in energy released as Bremsstrahlung by the two isotopes (the Bremsstrahlung energy released being directly related to the maximum energy of the beta particle).

In an animal killed one day after intravenous administration of a large dose of P^{32} , wide variations in P^{32} distribution in the bone were readily detectable utilizing a miniature beta probe described below.

Strontium

At the end of four weeks of feeding $0.5 \mu\text{c Sr}^{90}\text{-Y}^{90}/\text{g}$ food to yearling trout, the average total beta activity for carcass and operculum (bone) was $0.076 \mu\text{c/g} \pm 0.008$ and $2.2 \mu\text{c/g} \pm 0.21$, respectively. The activity of the food was increased to $1.0 \mu\text{c/g}$ food at week 7.

Although no detection of radiation damage is expected at the present body burden of Sr^{90} , measurements on plasma chloride and plasma freezing point depressions have been made. The average of 124 meq chloride/liter determined for the Sr^{90} fish is no different than normal values.

Some interesting results were obtained on the effect of the chelating agent EDTA on absorption of strontium and calcium on the intestine of the rat. Perfusion of the small intestine in vivo with 5 mM solutions of EDTA in the presence of Sr^{85} and Ca^{45} tracers resulted in a sharp reduction in Ca^{45} absorption but little effect on Sr^{85} absorption. The ratio of Sr^{85} to Ca^{45} absorbed which is normally in the range of 0.3 to 0.5 was changed in the presence of EDTA to approximately 2.0. Studies were initiated to determine the ratio of Sr^{85} to Ca^{45} in subcellular components of the liver. Slightly more Sr^{85} was found in the mitochondrial fraction than calcium, while the reverse was true for the microsomal soluble fractions.

A preliminary study was performed to assess the feasibility of using a miniature beta probe for obtaining information on radiation dose rates in bones from animals fed Sr^{90} . A 1/8 inch diameter by 1/8 inch thick NE-102 scintillator was used to make measurements over the sawed surface of a tibia from an animal which had received 3.1 mc Sr^{90} per day for just over 100 days. These results were compared with measurements made at the surface of a plaster of paris source which contained a known quantity of Sr^{90} for which dose rates could be calculated.

Considerable variation was noted in the radiation dose rates at different locations at the surface of the sagittally sawed bone. The highest dose rates of 300 to 400 rads per day were detected in the bone which had been formed during the period of isotope administration. The lowest dose rates, 75 to 150 rads per day, were found in the less dense spongy bone and the bone which had been formed for the most part previous to isotope administration.

Comparative Toxicity of Sr^{90} , Ra^{226} and Pu^{239}

Female miniature swine administered $6.4 \mu\text{c Ra}^{226}/\text{kg}$ six months previously are exhibiting elevated blood urea nitrogen levels as compared to control values. Animals that received single doses of Sr^{90} and Pu^{239} ($64 \mu\text{c}$ and $1.3 \mu\text{c/kg}$, respectively) at the same time have normal values. Previously high blood urea nitrogen levels were noted in male swine injected with comparable doses of Ra^{226} .

Technetium

Studies were initiated to determine the absorption, distribution and retention of Tc^{99} in rat. These studies to date have been principally limited to the development of suitable methods for controlling valence state and assuring against loss of activity in the digestion processes. There are indications that the valence state is not a very important factor in determining distribution and that technetium absorbed is quite rapidly lost from most tissues. The kidney retention is high after one day. Concentration of technetium in the thyroid is being investigated.

Iodine

A seven-year-old ewe fed $1.5 \mu c I^{131}$ /day since being born to and weaned from a ewe on a similar I^{131} regimen was euthanized after showing symptoms suggestive of CNS lesions. A poorly differentiated follicular carcinoma of one lobe of the thyroid was observed with extensive metastases in lymph nodes, lung, liver, kidney, adrenals, brain, pituitary, and bone.

Neptunium and Plutonium

A direct comparison of the absorption of neptunium nitrate and plutonium nitrate in rats indicates absorption of 0.12 per cent for neptunium and an absorption of less than 0.01 for plutonium. There would appear to be a genuine and substantial difference in the absorption of these elements from the intestinal tract. The possible effect of the mass of material fed on absorption will be evaluated as soon as neptunium-239 becomes available to us.

Comparative toxicity studies with Pu^{238} , Pu^{239} , and Np^{237} have been instituted to determine the extent of chemical factors as compared to radiation factors. Neptunium administered intravenously as the nitrate was toxic within five minutes to 2 out of 3 rats at the 10 mg/kg level. The third rat died after 72 hours. Animals receiving 5 mg/kg survived without acute effects. Neptunium administered as the citrate is not acutely toxic at levels as high as 30 mg/kg body weight.

A study has been initiated to determine the retention and distribution of neptunium in the adrenals. Preliminary studies have indicated that there is a translocation and concentration of activity within the adrenals that may be demonstrated by autoradiographic means.

Three additional one-year-old lambs were injected with Np^{237} citrate at dose levels previously used. One ewe given 12 mg/kg died 69 hours post-injection. Post-mortem findings of hemorrhage and extreme congestion of the liver were similar to those found in a ewe given the same dose previously. Twenty-four hours prior to death, liver function, as measured by I^{131} Rose Bengal blood clearance, was severely impaired. Two ewes which received 3 mg/kg of Np^{237} survived, as did two other ewes given the same

dose previously. Although 3 mg/kg does not appear to be a lethal dose in one-year-old ewes, it does produce reversible damage to the liver as shown by the results of I^{131} Rose Bengal liver function tests in four ewes.

Limited results from histological section of ewes dying from Np^{237} toxicity show severe necrosis and degenerative changes in the liver. The brain also exhibited degenerative changes. Whether the brain damage is directly due to Np^{237} or secondary to other functional damage in the body, e.g., liver, is not known.

In a continuation of our studies on Pu^{239} in skin, a series of injections were made using equivalent μ c injections of Pu^{238} and weight equivalents of Np^{237} . At 12 and 28 days post-injection, the 1 μ c Pu^{239} sites resembled the 5 μ c Pu^{238} sites and the neptunium sites showed only minimal changes. These observations still leave undefined the contribution of radiation and chemical changes to the effects noted following Pu^{239} intradermal injections. The use of other plutonium isotopes and other forms of Pu^{239} is indicated.

A single animal was killed which had been intradermally injected with 24 μ c of Pu^{239} 8 days and 1 day previously and 12 μ c of Pu^{239} 1/2 hour previous to death. Various tissues were collected for radioanalysis, autoradiography, and histology. Preliminary results indicate that approximately 0.8 μ c of Pu^{239} was translocated to the liver. Since this seemed unusually high, further studies are being performed. Initially two males were injected with 1 μ c and 5 μ c Pu^{239} at various sites (one animal, 12 sites at 5 μ c per site; one animal, 24 sites at 1 μ c per site) and held for 7 days. Samples of the regional lymph nodes, liver, skeleton, and kidney were sampled for radioanalysis. Two additional males were injected and samples of their tissues collected for radioanalysis 24 hours post-injection.

Inhalation Studies

After exposure to $Ce^{144}O_2$, rats were treated with aerosols of Neo-synephrine, DTPA, or Neo-synephrine plus DTPA. After three weeks the rats were killed for comparison of the Ce^{144} contents of the tissues with those of control rats which had been exposed only to $Ce^{144}O_2$. The results are given in Table 1.

Table 1. Test of Pulmonary Clearance Agents
(Per cent of untreated controls)

Tissue	Neo-synephrine	DTPA	Neo-synephrine + DTPA
Lung	25	7	10
Liver	25	7	10

Both materials were effective in increasing the clearance of Ce^{144} from the lungs. This was accomplished without build-up in other tissues, such as the liver, which usually slowly accumulates Ce^{144} following inhalation. (The liver burden was 0.001 of the lung burden.) These results have been confirmed in a second experiment and represent our most successful attempt to find a method for removing radioactive particles from the respiratory tract. These agents are now being tested on dogs after exposure to plutonium dioxide aerosols.

Bone Marrow

Mice (C3H males) protected against the lethal effects of 900 r total body X ray by injection of bone marrow cells (C3H x 101) were sacrificed at intervals and liver and spleen analyzed for acid phosphatase activity. Approximately two-fold increases in liver acid phosphatase concentration (activity per unit wet weight) was observed two weeks following irradiation and bone marrow injection. The acid phosphatase is localized in the lysosomal fraction of parenchymal and Kupfer cells of the liver. This increase in acid phosphatase concentration seemed to parallel the hepatomegaly which was most apparent at two weeks and which declined at three weeks. Changes in spleen phosphatase activity are difficult to interpret owing to the drastic cellular changes seen, with a diminished spleen weight at three days, a two-fold increase at two weeks and a return to controls at three weeks. A more systematic study of the relationship with secondary spleen and phosphatase activity and the response of the animal to foreign matter is being initiated.

Cellular Biology

Work on identifying the 260 mμ absorbing materials which leak out of X-rayed yeast cells was resumed. Elution patterns from anion exchange resins show at least 4 different components. Absorption spectra and 280 to 260 ratios indicate the main component is an adenilic acid residue. A second fraction is tentatively identified as a uridine compound.

Rhodotorula yeast seems promising in radiation studies since it does not clump appreciably. From initial sensitivity experiments the yeast is haploid and has an LD₅₀ of about 20 KR and an oxygen effect of 2.7. The LD₅₀ for this yeast is three times higher than the Saccharomyces used in other studies.

Plants

Young pea seedlings are being used in an attempt to differentiate effects of transpiration on ion uptake and translocation. During the early stage of development when transpiration is at a minimum, no translocation of Cs^{137} to the cotyledon was observed during an eight-hour uptake period. Trace amounts of Cs^{137} were observed in plumule after brief uptake periods, i.e., in one minute.

Roots of beans grown in various nutrient solutions of K were found to vary in the amount of Cs retained against rinsing solutions. Roots grown in 0.2 and 0.02 mM K solutions released similar amounts of Cs, 2 per cent, whereas roots grown in 2 mM K released 20 per cent of the Cs¹³⁷. No difference was found in effectiveness of H, Cs or K ions as exchange ions.

Rattlesnake Springs

Preliminary observations were made of the plankton community in the new impoundment. A relatively simple association consisting of three algal species and four or five zooplankton species was present. Alkalinity values of pond water have ranged from 133 to 141 ppm as CaCO₃.

Columbia River

River water samples collected during February and March showed increasing turbidity and alkalinity. A majority of dissolved ions, especially iron, phosphate, and chloride, also increased during this period; boron decreased steadily.

A definite algal bloom was observed on March 22.

Fallout

Radioiodine was not detectable in several deer and caribou thyroids from Colorado, Maryland, and Alaska.

Project Chariot

Biomass determinations of an Eriophorum tussock community showed Sphagnum moss contributed 297 ± 118 (SD) g air dry weight per 0.5 square meter of ground area, compared to 244 ± 108 g of vascular plants. The extent of this community in the Arctic tundra and the effective concentration of fallout radionuclides by Sphagnum moss combine to create an important reservoir of fallout materials.

Insects

Electrophoretic measurements indicated a sexual difference in protein structure of meal moth larvae. Subsequent studies will be made of the effect of ionizing radiation upon this property.

HA Kornberg
Manager
BIOLOGY OPERATION

HA Kornberg:es

C. Lectures**a. Papers Presented at Society Meetings**

None

b. Off-Site and Local Seminars

Thompson, R. C. A perspective on fallout. March 6, 1962.
Washington State University, Institute on Dairy Science, Pullman,
Washington.

Hanson, W. C. Seasonal patterns of stable and radioactive iodine
in thyroids of native jack rabbits. North American Wildlife
Conference, Denver, Colorado, March 13, 1962.

Bair, W. J. Retention, translocation, and excretion of inhaled
plutonium in dogs. Oak Ridge National Laboratory, Health
Physics Division, Oak Ridge, Tennessee. March 29, 1962.

c. Seminars (Biology)

Mahlum, D. Dennis. Cell energy metabolism. March 6, 1962.

Farner, Donald S., Washington State University, Pullman, Washington
(Exchange Seminar). Photoperiodic mechanisms in animals.
March 20, 1962.

Noyan, Ahmet. Relative importance of the blood and lymphatic systems
in water resorption in the small intestine. March 27, 1962.

Mraz, Frank R. Intestinal absorption of calcium-45 and strontium-85
as affected by the alkaline earths and pH. March 27, 1962.

d. Miscellaneous

Erdman, Howard E. Biology at Hanford. March 16, 1962. High school
science fair, Yakima, Washington.

Dockum, Norman L. Autoradiography. HAPO Staff Meeting, March 28, 1962.

D. Publications

Documents: None

Open Literature: None

OPERATIONS RESEARCH AND SYNTHESIS OPERATION
MONTHLY REPORT - MARCH, 1962

ORGANIZATION AND PERSONNEL

Bonnie L. Pierce was promoted from Clerk to Secretary following completion of the shorthand requirements.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Fuels Preparation Department

The updating of the empirical models relating fuel element distortion during irradiation to reactor variables is almost completed. These relationships will now be based on over 12,000 fuel elements. Over-all agreement with existing models is very good, but in some instances the models have been simplified by the elimination of some terms. Results of this analysis, plus a thorough discussion, will be contained in a formal document to be issued jointly with FPD personnel.

A design was made of a production test intended to investigate, under carefully controlled conditions, the effects of certain (d) ingot impurities on in-reactor performance. The basic problem involves investigating three impurities at three levels, and the design proposed is a cube plus two octahedra design with four center points, charged in a balanced lattice design of block size 5.

A document was issued setting forth in detail the analysis of the output from the MERCY program that is made in order to separate and identify the error sources contributing to a given total measurement error at some measurement station. These additional calculations will be incorporated in the MERCY program. It is intended to simplify the output from the program to the extent that the output itself will comprise the routine reports to be issued displaying the total measurement error, and the contributing sources at the various stations.

Existing sampling plans for specific ingot impurities were reviewed for effectiveness. It was recommended that different plans be adopted in view of the poor power of the existing plans.

An analysis was made of some small amounts of uniform corrosion data from Quality Certification tubes in order to compare roughly different vendors with respect to uniform corrosion attack.

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Consulting assistance was provided in connection with the analysis of data from a 2 x 3³ designed experiment which evaluated the effects of certain welding variables on weld width and weld penetration.

An analysis is being made of warp data from 13 recent NPR extrusions which permitted estimation of the effects on warp of A, B, and C billets plus other extrusion variables. This analysis is being made together with FPD personnel, and a joint report will be issued.

A report was issued presenting the results of an analysis performed on wall thickness data from some 115 NPR outer extrusions. Real time associated effects were found, but it was not possible to clearly identify the variables causing these significant differences using these production type data.

Some further vector 1 warp data (the vector describing the change in warp of an NPR fuel element after Beta heat treating) were analyzed in order to test the significance of certain proposed hypotheses.

Experimental data on the relationship between film density and NPR fuel cladding thickness are being studied in order to develop a computer program which will routinely calibrate density and thickness and thus provide a fuel cladding acceptance method.

The study on the measurement biases and imprecisions existing in measuring bumper shoulder heights was completed. This was performed in order to determine if the specifications on the rail height were too rigid in view of the existing measurement capabilities.

Discussions were held as to the size and nature of an experiment proposed to compare cleaners on the basis of component wettability ratings. A memo was written discussing the proposed test and giving appropriate power curves.

Assistance was given in the statistics of a tester designed to detect uranium contamination in the bonding layer. Knowing the background count and the count of a standard having a given amount of contamination, the testing times required and the rejection levels were established for given sets of risks of failing to detect a given contamination, and of erroneously culling out a good fuel element.

Irradiation Processing Department

Some further assistance is being provided in connection with describing internal process tube corrosion. Attention is directed towards how best to utilize existing probolog data in intelligently arriving at such an internal corrosion model.

As requested, existing sampling plans for incoming coal shipments are being reviewed to determine what amounts of compositing are necessary.

A meeting was held with IPD financial personnel to present and discuss the results of the overtime study, and review the status of the computer simulation study. It was pointed out that inadequacies in the recording and maintaining of reactor technical data were delaying the study, and it was agreed that an analysis of source data requirements for the simulation model would be the first study objective. A report delineating these source data requirements will be issued within a few weeks.

An analysis was made of total count data from a canning line known to have operated out of process specifications for an undetermined amount of time. Subsequent to this, assistance was provided in formulating a plan to take additional samples from the suspect lot to assist in reaching a decision on the lot disposition.

Work continued on the problem of estimating the probability of detecting defects in NPR welded primary piping. Current efforts are directed toward providing a statistical analysis of data obtained from guided bend tests of specimens of welded pipe.

Work continued on the provision of a reliability analysis of various reactor safety systems employing a "k out of n" type of trip logic. Probability graphs for various devices based on a four-state reliability algebra were drawn and sent to interested IPD personnel.

Chemical Processing Department

In view of certain observed discrepancies in the data, a review is being made of existing means of controlling ^{240}Pu content in meeting the specifications on the final product. This control is maintained now by 100 percent neutron counting supplemented by mass spectrographic measurements on a random sample basis.

Dimensional measurement data for balls and plugs were readied for submission to the computer. An analysis will be performed to identify sources of variability in the recorded measurements.

An investigation is being made into the feasibility of a study of the probability distribution of transportation accidents severity. The further delineation of risks in the shipment of radioactive materials by alternate railroad, motor vehicles, and ship routes will depend upon the existence of records adequate for the construction of appropriate models.

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HW-73202

An analysis was made of work sampling data in order to test for the significance of the "day" effect.

An EDPM Program is being written which converts manufactured parts gauge data from the design coordinate system to one that is more convenient for specific local gauges.

Mathematical expressions have been developed and are being evaluated which express the self-interaction coefficients for annulus-shaped containers used in storing fissile materials. These coefficients are of primary importance in criticality studies.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HLO

2000 Program

Further statistical analysis was done on data from the twenty pilot study experiments run on the pulse column last summer. A multiple regression analysis was completed which correlated mass transfer indices, fraction of total uranium throughput transferred and absolute transfer rate with a quadratic function of six column control variables, pulse amplitude, pulse frequency, feed concentration, distance from the flooding curve, capacity energy ratio, and extractant stream acidity.

Consideration was given to the question of scheduling the sequence in which ports are sampled during an experimental run so as to minimize the effect of removing the absorptiometer sample on later sampling in the sequence. A solution was obtained to the problem of determining the schedule which maximizes the minimum distance between adjacent sampling ports.

Theoretical studies are being made on the heat conduction properties of a proposed annular-shaped container for high thermal source materials. A mathematical model has been developed and the appropriate numerical calculations are well under way to study the solutions as a function of the variable design parameters.

Mathematical investigations are in progress to study the fluid-flow patterns which might be expected in underground waste storage tanks if they were equipped with a proposed axially-located circulating device.

Further discussions were held with personnel engaged in creep testing to formulate the creep raw data package of the mechanical properties raw data file to be built in conjunction with the program storage and retrieval system. A tentative format has been set up for the display of creep raw

data and defining variables on magnetic tape. To assay the appropriateness of this format and its compatibility with the storage and retrieval system, several sets of experimental data will be arrayed in this format and compatibility with the storage and retrieval system checked.

The analysis of the data from the chromate-dichromate autoclave treatment test was completed. As a result of this experiment, sets of autoclave conditions were found for use in preparing fuel elements for reactor testing.

A visit to the Manufacturing Services Laboratories in Schenectady was made for the purpose of checking out for the first time some of the 7090-generated magnetic tapes on the prototype Gorton Controls under development there. On the basis of those tests which were possible to complete, the technique of preparing tapes appears to be both feasible and practical. An agreement with Manufacturing Services has been made to prepare one more tape especially designed to test the characteristics of the controlling mechanisms. It is planned to run these tests in mid-April.

4000 Program

Irradiation Effects on Structural Materials

Several discussions were held concerning the experimental design for the various physical properties measures to be investigated in the forthcoming irradiation effects on structural materials AEC Program to be directed by the Hanford Laboratories. Currently, designs are being constructed for physical properties tests on Hastelloy X, the initial material to be investigated in the program. For each of the physical properties, tensile, impact, stress to rupture, creep and burst designs are being constructed which cover the range of independent variables currently and in the foreseeable future of interest.

Swelling Studies

A program has been written to fit fuel element swelling data to an empirical model for the swelling of uranium fuel elements. The model which is nonlinear in the parameters has been programmed for use with the NELLY nonlinear least squares program.

Further programming was done of statistical procedures for detailed analysis of micrograph pore size distributions. The routine evaluation of void fraction density estimates for photographs of uranium samples continues.

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HW-73202

Fuels Research

A study is being conducted of the characteristics of particle size sieving from a probabilistic and statistical standpoint. A mathematical model was developed which predicts particle size distributions as a function of the initial composition, screen sequential position and screening time. Several experiments were designed and are currently being carried out. The data will be used to test the model and to determine the characteristics of the material in question.

Further work was done on the precision and accuracy studies. Two additional experiments were designed to test the within day-between-day effects of the quinalizarin method of estimating trace boron content and to determine relative biases between this method and the azure C method. Further statistical work was done on the trace plutonium content data to determine the precision of the method as a function of sample size and pre-dissolution treatment of the material.

5000 Program

Actinide Element Research

Investigation was begun into the feasibility of writing a program to determine crystal indices of powdered crystals of actinide compounds by means of X-ray diffraction. A program for cubic crystals has been written, debugged, and is in the process of being applied to actual data.

Division of Research Programs

Work continued on the grid structure phase of the search and detection problem. Log time and log distance models were correlated with program sample data using the new computer search and planer plot routine.

The definition of the replacement for the RCA-IRA Program continues. Discussions have been held with data processing personnel and a definitive definition of approximately half of the over-all program has been completed. Laboratory blank evaluation was initiated with present IRA file data.

6000 Program

Biology and Medicine

Work was begun on programming a multicompartiment model to find rate estimates. The program will be used first on a retention study dealing with fish.

Other

Multiple regression relationships are being found between Pu excreted in the urine and the time after intake for various sets of conditions.

The first of a series of three lectures on probability and statistics was given to participants in the Engineering "A" Course.

Carl A. Bennett

Manager
Operations Research and Synthesis

CA Bennett:dgl

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PROGRAMMING OPERATIONMARCH, 1962REACTOR DEVELOPMENT - 04 PROGRAMPLUTONIUM RECYCLE PROGRAMSummary of Bred Fuel Value Calculations

At AEC request, the values of plutonium and U-233 in five different simulated reactor concepts with various uranium price schedules and various fuel inventory charges are being calculated with the MELEAGER CHAIN code. The reactor types simulated in this study are the APWR, the BWR, the HWR, the GCR, and the OMR as described in HW-72217, "Fuel Cycle Analysis for Successive Plutonium Recycle - I. Results for Five Reactor Concepts." In HW-72217, the July 1, 1961, uranium price schedule is used. In this study the July 1, 1961, as well as other uranium price schedules are used with use charge rates of 4.75, 7, 10, and 12.5 percent. The batch irradiation cases have been completed for successive recycle of the bred fuel (plutonium or U-233) from the previous cycle with and without incremental fuel fabrication costs. Similar computations for graded irradiation are well under way. In addition, preparations are complete to compute selected recycle cases without limitations as to the amount of bred fuel available for recycle.

Combined Cycles

The Combined Cycle Analysis is concerned with computation of minimized fuel costs using two alternative fertile materials, U-238 and Th-232, each enriched with five alternatives (U-233, U-235, and three different isotopic compositions of plutonium) in various reactor lattices with various jacketing materials so as to span most thermal reactors under consideration today. Batch and graded cycles are computed for several different neutron leakage and control allowance levels.

The basic block of information for the Combined Cycles Studies is nearing completion (see Table I). This basic block of information will be surveyed for areas of extraordinary interest, which will receive closer scrutiny by means of MELEAGER CHAIN code and special computer runs as appropriate. In addition, it will be necessary to assess the impact of the conservatism of MELEAGER code on fuel cycle costs and consequent bred fuel values for closely spaced lattices. A special version of MELEAGER is being debugged to allow unique flexibility for such studies of the model used. Currently, the standard to be used is a modification of SPECTRUM V (HW-71953) which is a combination of the General Atomics codes SPECTRUM and SIGMA GAS. SPECTRUM V calculates the thermal neutron flux spectrum for an infinite homogeneous medium using scattering kernels for a monatomic heavy gas with a Maxwellian distribution of velocities. For the work with MELEAGER, scattering kernels for light water and graphite will also be used. SPECTRUM also calculates weighted average cross sections for all of the materials individually. In this study 75 energy groups are employed from essentially zero to 4.2 ev to embrace the resonance regions of interest to plutonium fueled cores.

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HW-73202

TABLE I

STATUS OF COMBINED CYCLE COMPUTATIONS - MARCH 1962

Assigned Series No.	Fuel Combinations	Plutonium Compositions Used for Fissile Enrichment					Degree of Completion			
		Pu-239		Pu-240	Pu-241		Pu-242	MELEAGER %	PROTEUS* %	QUICK and MINIMIZER %
		%	%	%	%	%				
1,000	U-235 in uranium			None			90	90	80	
2,000	U-233 in depleted uranium			None			90	90	80	
3,000	Plutonium in depleted uranium	70	18	11		1.0	90	90	80	
4,000	Plutonium in depleted uranium	95	5				90	90	80	
5,000	Plutonium in depleted uranium	31.3	33.6	25		10.1	90	90	80	
6,000	U-235 in Th-232			None			85	85	75	
7,000	U-233 in Th-232			None			85	85	75	
8,000	Plutonium in Th-232	70	18	11		1.0	85	85	75	
9,000	Plutonium in Th-232	95	5				85	85	75	
10,000	Plutonium in Th-232	31.3	33.6	25		10.1	85	85	75	

* Extends MELEAGER data.

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Cross sections are being added to the SPECTRUM V library for Pu-241 and Pu-242 to allow handling of well burned plutonium in SPECTRUM V code. Thus, it will be possible to derive empirical values of k for the special MELEAGER deck for fuel compositions and enrichments of interest to the Phoenix Fuel Study, U-238 Average Spatial Concentration Study, Zoned Spectrum Reactor Study, and the closely spaced lattices analyzed in the Combined Cycles Studies.

A preliminary report has been outlined which will identify the crucial problems involved in determining the impact of Fuel Geometry, Reactor Physics, Fuels Management, and Economic Climate on bred fuel values and fuel cycle costs. Techniques are already adequate to treat many of the parameters with sufficient adequacy. Computing strategies being employed to enhance the accuracy of treatment of the remaining parameters will be described. Hand calculations have been carried out to make certain that the inventory isotopes are being handled correctly among MELEAGER, PROTEUS, and QUICK codes for the ten fueling schemes; and that costs are being calculated correctly.

Variations of U-238 Spatial Concentration

A computer code has been written to allow direct communication with the MELEAGER burn-up code which is of particular interest to the Combined Cycles and Variable U-238 Spatial Concentration Studies. This new code calculates lattice parameters for a specified fuel element geometry and fuel type and translates these into the input parameters necessary for the MELEAGER code. A special output tape can be used to punch the necessary change cards to make a complete new MELEAGER case for each lattice case run. In addition to providing a convenient means of communication between a fuel element description and a burn-up code, the code calculates buckling and leakage for five reactors ranging from 100 MW thermal to 10,000 MW thermal as a function of six specific heat transfer rates ranging from 250,000 to 10,000,000 BTU/ft²,hr to afford a wide perspective of reactor potential. Several other quantities useful in engineering and economics evaluations of reactors are calculated. (Example: The clad surface of the fuel regions in sq ft/ft of cell is calculated.)

The availability of this code is important to the U-238 Spatial Concentration study. It is now possible to use more easily the proper flux and volume weighted thermal utilization for each fuel element and moderator geometry and each enrichment. The amount of time necessary to make a single thermal utilization calculation has been so great that in large surveys, such as the Combined Cycles study, it was necessary to parameterize the nonfuel absorption term "Sigma Non-Fuel" or SNF⁽¹⁾ corresponding to an average thermal utilization rather than use a value specifically calculated for each case.

The variation of SNF with changing SDFV⁽²⁾ is illustrated by the following cases:

H₂O Moderated Cases

<u>SDFV</u>	<u>SNF</u>
1.368	0.139
4.076	0.177

Graphite Moderated Cases

<u>SDFV</u>	<u>SNF</u>
2.22	0.104
6.63	0.120

$$(1) \text{ SNF} = \frac{\sum_k \sum_i \phi_k V_k \text{Ncc}_i \sigma_{\text{oa}_i} \xi_{\text{ai}}}{\sum_k \phi_k V_k} \frac{1-f}{f}$$

where the summation is made over each isotope "i" for each fuel region "k"

ξ_i = Westcott g for the "i"th isotope in the "k"th region

Ncc_i = Nuclei/cc of "i"th isotope

σ_{oa} = 2200 m/s absorption cross section of the isotope

k = Thermal neutron flux of the "k"th region

V_k = Volume of the "k"th region

f = Thermal utilization factor

(2) SDPV = the slowing down power volume ratio and is defined by the equation

$$\text{SDPV} = \frac{(\xi \sum_s V_m \phi_m)}{V_f \phi_f}$$

$\sum_s$ = Macroscopic scattering cross section of moderator

V_m = Moderator volume

V_f = Fuel volume

ϕ_m = An average flux in the moderator

ϕ_f = An average flux in the fuel

ξ = Logarithmic energy decrement the average over all the neutrons of the logarithm of the energy change per collision.

Typically, an increased SNF will increase fuel costs. Since minimum fuel costs were obtained with Pu-240 rich plutonium fuels at maximum density only when SDPV was large, it could be expected that inclusion of the proper SNF with a change in SDPV may make reduced density plutonium fuels appear attractive under more conditions than previously seemed to exist. In particular, as reported in February, pressurized and boiling water reactors are typified by SDPV values of 1.5 to 2.5 wherein SNF is lessened; i.e., the product of p_f in the formula $K = 3 \epsilon p_f$ is maximized. Under these circumstances, reduced U-238 spatial concentrations definitely increased the value of plutonium used as enrichment if the relative Pu-240 concentration was great.

Zoned Fueling of Plutonium Enriched Reactors

As described on pages 105-109 of HW-72217, the incremental cost of fabricating plutonium bearing fuel elements can be minimized by "zoning." That is, mixing the plutonium with low composition uranium will concentrate the available plutonium into fewer elements thus reducing the total FEFJ cost of the load. Two popular zoning criteria are to make the initial enrichment of the fuel the same or to make the initial fission cross section of the fuel the same in both zones. If the fuel elements are placed in the reactor such that both types of element "see" the same flux then heat transfer considerations dictate that the second of these criteria be used. If, however, the plutonium fuels are positioned in low flux regions for power flattening then the uniform initial enrichment should be used.

A computer program was written to calculate the zone size for each zoning scheme for two "depleted" uranium compositions. The results for the reactors run in the MELEAGER CHAIN computational series are shown in Table II. Note that as a result of the high fission cross sections of Pu-239 and Pu-241, the uniform initial fission cross section condition requires that the plutonium be contained in $1\frac{1}{2}$ to 2 times as many fuel elements as does the uniform initial enrichment requirement.

In the MELEAGER CHAIN calculations, the net delta FEFJ cost is assumed constant throughout the recycle series, which is equivalent to assuming that the premium FEFJ cost per fuel element decreases. This inflates the values calculated for the plutonium generated in the series and, as a consequence of the PUVS solution, raises the initial plutonium value. In order to eliminate this effect, a routine to calculate the zone size will be programmed into QUICK. This routine will contain the additional refinement of calculating the uranium enrichment in the uranium only elements as this is necessarily different from the enrichment when fully mixed. The uniform initial enrichment criterion with minimum cost uranium enrichment will be used.

TABLE II
ZONE SIZES IN VARIOUS REACTORS FOR
PLUTONIUM MIXED WITH NATURAL URANIUM

<u>Reactor</u>	<u>Recycle Step Number</u>	<u>Percent of Elements Containing Plutonium</u>	
		<u>Uniform Fission Cross Section</u>	<u>Uniform Enrichment</u>
APWR	2	32.3	15.9
	3	39.2	20.4
	5	41.6	22.1
BWR	2	48.7	27.5
	3	53.7	32.0
	5	64.1	42.4
HWR	2	52.6	34.5
	3	59.9	41.0
	5	62.8	43.6
OMR-PRIME	2	37.0	19.1
	3	46.1	26.1
	5	52.4	31.8

Code Development

Extremely long lived fuels that have relatively constant composition for a significant portion of the in-reactor time had been poorly curve fit by PROTEUS which is used to extend basic burn-up data from codes such as MELEAGER. The PROTEUS code was modified so that these isotopes will be fit in two parts. That is, in the portion of the in-reactor time in which the isotopic composition varies, a third order polynomial will be used, while in the second portion a linear relation will be used. The boundary is the time at which the slope of the curve is one-tenth of the initial slope. This time will be, in general, different for each isotope in the case. In most cases the boundary will not be reached and the curve fit will be as in previous versions of the code (i.e., polynomial fit over the entire time).

The PUVF code was modified so that the iterated endpoint calculation will be done with every recycle step after the second considered to be the final step. This will be a considerable aid in choosing the best set of values.

Some difficulties in finding the minimum fuel cost were found during MELEAGER CHAIN computations for graded cases. The problem was caused by the reduced uranium price and low enrichments necessary for graded cycles. These two factors made the fuel cost curves essentially flat in the region of the minimum, which in turn made determination of the actual minimum increasingly difficult. To correct this, the minimum hunting routine was altered to neglect spurious minimums caused by slight fluctuations in the basic fuel cost fits.

With the increased interest in Phoenix fuels, an addition to the plotter code was made which will hunt for a known Phoenix composition in any physics case being plotted. The plutonium compositions (in atomic percent) are plotted as a function of fuel exposure, and the composition and the exposure at which conditions to produce a Phoenix composition are satisfied are listed. Along with the Phoenix composition, the allowable deviation from the specified composition is supplied as input.

Some changes were made in QUICK separation subroutine to eliminate plutonium and uranium conversion charges when a fixed separations charge per unit weight or volume of fuel is used. On the other hand, when the AEC model separations plant is employed, the conversion costs are used as described in the AEC Cost Handbook. Also, the QUICK fuel cost code may now be used optionally on a \$/cc of fuel basis instead of \$/pound of fuel basis as used in the past.

A modification to MELEAGER CHAIN has been made which allows QUICK to minimize using previously calculated physics results stored on tape. With this method small changes in cost parameters may be iterated without rerunning a complete chain which reduces run time by a factor of four.

Changes were made in MELEAGER to correct for the difference in fission energies of the fissile isotopes. Although checking is still taking place, results indicate that for recycle cases the change in plutonium value is not detectable. However, the fuel costs of both uranium and plutonium recycle case were lowered.

Plutonium Requirements for Mixed Oxide Fuel Elements in PRTR

An initial ηk_{eff} (approximately k_{00} as evaluated using Westcott cross sections) product value of 1.275 for plutonium-enriched natural UO_2 Mark I fuel elements was selected as a standard. This is the value of ηk_{eff} obtained by assuming that 95 percent of the equilibrium amounts of fissile and fertile plutonium isotopes, formed by irradiation of U-238 in the PRTR, are recycled and mixed with the same amount of U-238 plus naturally occurring U-235; in addition the presence of Pu-242 in amount equal to 5 percent of the total plutonium was assumed. Under these conditions the amount of plutonium per ton of natural uranium is 5.96 kilograms, of which 37.4 percent is Pu-240. For shorter irradiations the concentration of Pu-240 is lower and a smaller amount of plutonium is needed to attain the "standard" initial reactivity.

Plutonium needs to meet the initial $\eta \epsilon f$ requirement vary from about 2.5 to 6.0 kilograms of plutonium per ton of uranium as the plutonium Pu-240 isotopic content ranges from about 5 to 37 weight percent. Plutonium and U-235 fuel isotopic contents were calculated, by Differential Digital Analyzer, as functions of exposure using the Westcott cross sections with constant $r = 0.08$ and a constant neutron temperature of 130 C. These burnup calculations show that as feed Pu-240 plutonium content ranges from 5 to 37 weight percent, irradiation exposure may be increased from about 3300 to 5000 MWD/ton of initial uranium for fuels discharged at a constant $\eta \epsilon f$ value of 1.165. Corollary to the preceding statement, at 5000 MWD/ton uranium exposure the $\eta \epsilon f$ value at discharge will range from 1.12 to 1.165 as the initial Pu-240 content varies from 5 to 37 weight percent. Typical feed and discharge data are summarized in Table III.

If the three sets of data for "Kilograms of Plutonium per Ton of Uranium" and "Weight Percent Pu-240," shown in Table III for the initial condition and two different discharge conditions are plotted on the same graph, it is observed that all points fall very closely onto the same curve. This indicates that the decline in reactivity with irradiation is caused by depletion of U-235 and accumulation of fission products; recycle of plutonium into the same amount of natural uranium will restore the reactivity to the initial value essentially independent of the initial plutonium composition or of the amount of irradiation. However, it is also indicated that any process losses of plutonium will have to be compensated by the use of new fissionable material. (This correlation is valid only for low concentrations of Pu-242.)

TABLE III

INITIAL AND FINAL FUEL COMPOSITIONS FOR TWO DIFFERENT
DISCHARGE CONDITIONS FOR MIXED OXIDE FUEL IN PRTR

Required Initial Pu for $\eta \epsilon f = 1.275$		Discharge at $\eta \epsilon f = 1.165$				Discharge at MWD/T = 5000			
Wt. % Pu-240	Kg Pu per Ton U	MWD/T	Wt. % Pu-240	Kg Pu per Ton U	Wt. % U-235	$\eta \epsilon f$	Wt. % Pu-240	Kg Pu per Ton U	Wt. % U-235
0	2.23	-	-	-	-	-	-	-	-
5.3	2.55	3300	15.6	3.27	0.48	1.12	19.1	3.58	0.39
15	3.18	3500	21.5	3.75	0.48	1.13	23.3	3.96	0.41
21.6	3.74	3700	25.4	4.11	0.48	1.135	26.5	4.27	0.42
25	4.08	4000	27.8	4.51	0.48	1.145	28.5	4.60	0.43
37.4	5.96	5000	36.7	6.08	0.48	1.165	36.7	6.08	0.48

Salt Cycle Reprocessing

Work on the Salt Cycle reprocessing computer code was devoted to assembling the economics routine. The complete program including all the subroutines is now being assembled and should be ready for initial trial runs about the middle of April. The code has been written to provide a choice between lease or purchase, and if desired a further choice can be made of leasing all uranium above a specified enrichment with purchase at all lower enrichments.

OTHER ACTIVITIESHanford Science Colloquium

The Hanford Science Colloquium scheduled for April 3, 1962, was cancelled at the request of the speaker.

Manager,
Programming

WK Woods:jm

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF MARCH 1962

A. ORGANIZATION AND PERSONNEL

Effective March 1, Louella M. Robb was reactivated into External Dosimetry Operation. Arlene D. Maxfield resigned from Radiation Monitoring March 2. On March 5, Bonita F. Ray was transferred to Analytical Laboratories. Melvin L. Brinkerhoff transferred to Chemical Processing Department effective March 19. Frances S. Glick was reactivated and Ronald L. Olson was hired effective March 26. On March 30, Louella M. Robb and Ralph C. Dozer retired from External Dosimetry Operation.

B. ACTIVITIES

Occupational Exposure Experience

Two new cases of plutonium deposition, each less than 0.5% mpbb, were confirmed by bioassay data during the month. The total number of plutonium deposition cases that have occurred at Hanford is 285, of which 205 are currently employed. The two new deposition cases resulted when two HLO Plutonium Metallurgy employees were exposed to air-borne plutonium during a plutonium carbide melting operation at the 231-Z Building.

Nasal smears of about 600 d/m plutonium were found for a CPD operator after his entry to a room in the 234-5 Building in which a condenser had exploded in an open-faced hood. Severe skin contamination from plutonium occurred in six separate incidents at the 234-5 Building, involving a total of five CPD employees and two construction employees. Skin contamination levels to 10^5 and 10^6 d/m occurred. All were decontaminated successfully. In one instance, a skin eruption noted in a contaminated skin area was examined at the Whole Body Counter, but no activity above the detection limit was found. Special urine samples were requested in each instance.

About 3:00 p.m., March 20, an employee noticed water coming from a blanked-off flange in the E-4 line (hood air exhaust) in room 46 of the 231 Building. A quick check revealed water running through the filters into the furnace hood in room 47-B. The building was immediately evacuated while the investigation continued. Water was found in two glove boxes and on the floor in room 46. At the start of the 4-12 shift, the contaminated water seeped through a hole in a bag in room 47-B and contaminated about 75 square feet of floor area. The source of the water was a leak in the cooling jacket of a vacuum pump in room 47-C.

Twelve square feet of 108-F Building laboratory floor space were spot contaminated to greater than 40,000 d/m following the breaking of a 15 ml glass vial containing approximately 10 microcuries of plutonium-239. Contaminated floor surfaces were taped down immediately and confined to the laboratory floor area. All skin surfaces of the one employee involved were surveyed with no contamination detected. A special bioassay sample was requested. This event was formally recorded as Radiation Occurrence Report 62-0-17.

Dose rates experienced at the Plutonium Recycle Test Reactor during the extended shutdown resulted in little or no restrictions on scheduled maintenance operations. Flash dose rates to 5 r/hour and radiation beams to 47 rads/hour were encountered over open process tubes during inspection. Maintenance on primary pumps was accomplished with body and hand dose rates of 100 mrad/hour and 5 rads/hour, respectively. Leak checking in A Cell encountered dose rates to 500 mr/hour. Entries into A Cell while operating at 5 Mw revealed a maximum dose rate of 1.25 rem/hour, including 1 r/hour. Purification of tritium-contaminated heavy water resulted in several positive tritium bioassay results, ranging to 34.9 $\mu\text{C T/l}$. The maximum radiation exposure for the month from internally deposited tritium was 321 mrem. Sixteen employees received a dose greater than 25 mrem from tritium assimilation.

An explosion of an aluminum-encapsulated graphite sample at the 3730 Building Hot Shop allowed small fragments of graphite with dose rates to 200 mrad/hour to be dispersed over approximately 50 per cent of the floor area. Air samples collected during the incident indicated air concentrations of $8.4 \times 10^{-9} \mu\text{C FP/cc}$. Respiratory and protective clothing precautions precluded any personnel contamination problems.

Greenhouse techniques were employed at the High Level Radiochemistry Facility to facilitate decontamination and removal of equipment from A Cell. A beta-gamma ratio of 10 to 1 was established from film studies performed prior to entry. Dose rates within the cell ranged to 10 rad/hour with a general field of 1.5 rad/hour at the cell door. Prior to decontamination, dose rates ranged to 500 rad/hour.

Distribution of the radiation exposure report cards to Hanford employees was made on March 22. No significant inquiries or comments resulted following the distribution.

Environmental Experience

Concentrations of fallout, as measured in Pacific Northwest air samples, decreased for the second month in succession. Average values for the first three months this year were 6.1, 5.3, 4.4 $\mu\text{C/m}^3$, respectively. The actual concentrations during March ranged from 1.4 to 8.5 $\mu\text{C/m}^3$.

A survey of the beaches and islands between Richland and Ringold revealed a maximum dose rate of 1.1 mr/hour and ground contamination up to 7500 c/m. These values are about the same as measured last summer and fall.

A total of 107 fish, mostly whitefish and suckers, were taken from Columbia River sample locations at Priest Rapids, Hanford, Ringold, Richland, Burbank, and McNary Dam. One hundred and thirty-two tissue samples were submitted for analysis. Seventy-four produce samples were attained for radiochemical analysis. These included milk from the Ringold, Riverview, Benton City, Mesa, and Eltopia areas, as well as composite samples obtained from the Twin City Creamery. Milk samples totaled 125 gallons. Four pounds of Willapa Bay oysters, four pounds of ground round steak, 40 pounds of wheat, and 50 sets of beef thyroid glands were also obtained.

The Hot Semiworks facility, 201-C, has shown an increase in stack emissions with an increase in process activity. The daily emissions for the second half of March ranged from 0.2 to 6.7 mc/day (total beta) and with 20 to 50 per cent of the total radioactivity attributable to Sr^{90} . Emissions of this magnitude are at the high end of the recommended operational control for Sr^{90} of 5 mc/week and indicate the need for improvement. Action taken by Purex Technology management on this problem has been to install a double filtration system in the off-gas vent.

Studies and Improvements

Tentative limits for thorium and its daughters in effluent vapors emitted from the 306 Building stacks and roof vents were set at the respective 40-hour occupational MPC_a .

Three aerial monitoring flights were made, two over the project and adjacent lands and waters to collect background data and keep emergency equipment in a state of readiness. The third flight followed the Columbia River to the ocean.

A revised vegetation sampling program for sagebrush was put into effect on April 1, 1962.

A crucial part of the CPD hazards analysis is the rate of release of fission products from uranium during oxidation. After the comment drafts were issued, corrections to the release rate data were presented which resulted in drastic changes to all but one of the completed drafts. In addition, the maximum percolation rate used in the analysis of the waste storage tank was increased to the point where the postulated accident to this facility can no longer be considered incredible as was previously concluded.

The radiological consequences of an accidental excursion involving the Physical Constants Testing Reactor (PCTR) were evaluated. The analysis shows that only those personnel in the reactor building and within a distance of 100 meters would be significantly exposed.

The 328 Technical Shops completed their part of the fabrication of the new personnel film badge dosimeter processing machine. The machine is now in the Pipe Fitter Shop where it is being fitted with the appropriate hydraulic cylinders for mechanical operation. A six-second processing cycle is used. During this cycle, the machine: (1) Indexes the turntable; (2) X-ray codes the identifying payroll number on the film; (3) Supplies the dosimeter to the processing station; (4) Releases the dosimeter lock and withdraws the slide from the dosimeter body; (5) Removes the film packet from the slide; (6) Slides a new film packet on the slide; (7) Replaces the slide and new film packet into the body of the dosimeter; and (8) Removes the dosimeter from the processing station.

An evaluation of the film from 100-F Area checked well when the automatic densitometer readings were compared to the manually read film. The total radiation dose for 100-F Area film manually read was 15.3 r plus 1.4 rads of beta; the automatic densitometer total was 15.7 r plus 1.2 rads of beta.

The experimental silicon diode neutron dosimeters were exposed to 100 Kev and 200 Kev neutrons to determine their low energy response. The response to 200 Kev neutrons was about the same as that measured previously for higher energy neutrons. The diodes did not detect 1.2 rads of 100 Kev neutrons. A sharp drop in the silicon cross section at 140 Kev reduces the sensitivity about a factor of ten. The total cross section at 100 Kev is about one-tenth the cross section at 200 Kev. It appears that about 200 Kev is the practical lower limit of the diode energy response.

The performance of the Hanford neutron badge with various thicknesses of lucite added to the front of the badge was studied. Although there was some indication in the literature that the addition of lucite to the front of the badge would produce greatly increased slow neutron dose, this was not observed in the Hanford experiment. Lucite thicknesses as great as three inches were investigated. The PuF_4 source was used as the neutron source. The fast neutron dose determined by track counting gradually decreased as the lucite thickness increased. The slow neutron dose observed fluctuated slightly but did not significantly increase at any lucite thickness.

A similar experiment using a phantom was conducted to verify the response of the neutron dosimeters when worn on the front or the back of the phantom. A plutonium fluoride source was used to irradiate the front of the phantom to a total dose of 980 mrem of fast neutrons. The neutron dosimeters on the front of the phantom appropriately interpreted this dose; however, the dosimeters positioned at the back of the phantom showed no detectable dose. No increase in thermal neutrons was noted at the back of the phantom. Although there may be some point within the body where the neutron doses from the front and the back of the body should be summed, complete penetration of the body by a substantial fraction of the incident neutrons was not observed. Practical reasoning would indicate that the neutron dosimeter should be worn on the side of the body receiving the highest neutron dose. Except for unusual circumstances, this may be assumed to be the front of the body.

Commonly used neutron sources were reviewed to determine the percentage of their total dose that is delivered by neutrons with energies above 0.8 Mev.

CALCULATED DOSE ABOVE 0.8 MEV
FOR VARIOUS NEUTRON SOURCES

<u>Source</u>	<u>% Dose Above 0.8 Mev</u>	<u>Average Energy (Mev)</u>
PoB	100%	2.8
RaBe	100%	3.6
PuBe	98.72%	4.5
PoBe	98.39%	4.3
PoF	91.14%	1.4
Fission	89.27%	1.9
PuF_4	82.18%	1.4

Neutrons with energies of 0.8 Mev and above are monitored in the Hanford neutron dosimeter with NTA film. To assure the Hanford neutron dosimeter is accurately recording fast neutron dose, it is desirable to calibrate these dosimeters with a neutron spectrum similar to field spectra. To assure the neutron dosimeter does not underestimate fast neutron dose, it is desirable to calibrate with a neutron energy slightly less than that encountered in the field. The PuF_4 neutron source currently used for NTA film calibration meets both of the above criteria better than other neutron sources currently available.

Calibration of the Royco Particle Analyzer continued. Certain electrical modifications in the instrument were required in order to make the internal calibration circuitry perform properly. After these modifications, the instrument was calibrated with 0.557, 0.814 and 1.17 micron latex spheres. The Royco Particle Analyzer was used in the 234-5 Building and in the 308 Building where the particle size distributions in the work areas were measured. In general, it was observed that the number of particles decreased as the particle size increased. The maximum number of particles observed was in the size range from 0.32 to 0.40 microns and varied from 14 to 16 particles per cc. This is the smallest size range which can be monitored with the Royco Particle Analyzer.

The manufacturer's first prototype of the new portable BF_3 neutron monitoring instrument was received. Physical appearance, circuit layout and workmanship were of good quality. Nucleonic Instrumentation evaluation of the performance of the instrument indicated that it met the Hanford performance specifications. A letter requesting appropriate minor revisions was forwarded to AEC Purchasing with our approval to proceed with the fabrication of the complete order of 30 instruments.

Victoreen r-meter No. 1512 was sent to the Bureau of Standards for recalibration. They have indicated that they will be unable to calibrate this r-meter set until the middle of May or June.

During the month, seven Emergency Monitoring Kits were exchanged and serviced.

C. VISITORS AND VISITS

The Radiation Protection Operation staff had the following visitors during the month:

Dave Wagstaff - Oregon Board of Health, Salem, Oregon

E. R. Mercer - Agricultural Research Council, Department of Agriculture,
Letcombe, Berkshire, England

R. E. Wood - APED, San Jose, California

Dr. R. M. Chatters - Radioisotopes Laboratory, Division of Industrial Research,
Roger C. Brown Washington State University
Allen R. Mettler
Charles E. Simpson

A group of about 50 students from Waitsburg, Washington and Arlington, Oregon high schools visited the 3745 Building on March 13.

Twenty-eight high school students and teachers from Lewiston, Idaho toured External Dosimetry and Exposure Records.

Visits made by members of the Radiation Protection Operation staff during the month were as follows:

- A. R. Keene - Attended Subcommittee meeting on Registration of Technical Advisory Board on Radiation Control in Seattle.
- R. F. Foster - Presented lecture on "Behavior of Radionuclides in Food Chains - Fresh Water Studies" to U. S. Public Health Service Radiological Health Training Activities course in Cincinnati.
- L. F. Kocher - Discussed design specifications on new personnel film badge dosimeter with Product Engineering Company, Portland, Oregon.
- A. R. Keene - Attended steering committee meeting for the coming hearings on "Radiation Protection Standards Including Fallout", Joint Committee on Atomic Energy, Washington, D. C.

D. RELATIONS

Six suggestions were submitted by the personnel of the Radiation Protection Operation during the month of March. Three suggestions were adopted and four were rejected. The total amount of suggestion awards received was \$125.00.

The medical treatment injuries for the Radiation Protection Operation for the month of March were 4; the frequency for the month was 1.63. The total injuries for the year is 7; frequency for year to date is 1.08.

No security violations occurred during the month of March.

Two orientation talks were presented to Biology employees and five orientation talks were presented for 300 Area customer groups. A total of sixty-five persons have attended the first three sessions of the RPO Disaster Monitoring training course. Included in this group are representatives from all the product departments, from HLO, and from the AEC.

A practice session for the Radiological Emergency Staff was attended by eleven members. A nuclear excursion was assumed to have occurred in the Purex facilities. As a result of the critique following the exercise, a check list was prepared to stimulate the initiation of action and two status boards were prepared to record results attained from the various programs.

Safety meetings were held throughout the various components during the month.

E. SIGNIFICANT REPORTS

- HW-72691 2 - "Summary of Radiological Data for the Month of February, 1962", by R. F. Foster.
- HW-73212 - "Monthly Report - March, 1962, Radiation Monitoring Operation", by R. D. Tillson.
- HW-71702 - "Gamma Calibrations and Evaluation Techniques for Hanford Beta-Gamma Film Badge Dosimeters", by R. H. Wilson, V. M. Milligan, C. M. Unruh and H. V. Larson.
- HW-71764 - "A Personnel Dosimeter Filter System for Measuring Beta and Gamma Doses in Mixed Radiation Fields", by L. F. Kocher.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

<u>External Exposure Above Permissible Limits</u>	<u>March</u>	<u>1962 to Date</u>
Whole Body Penetrating	0	0
Whole Body Skin	0	0
Extremity	0	0
<u>Hanford Pocket Dosimeters</u>		
Dosimeters Processed	2,780	7,836
Paired Results - 100-280 mr	5	21
Paired Results - Over 280 mr	2	3
Lost Results	0	0
<u>Hanford Beta-Gamma Film Badge Dosimeters</u>		
Film Processed	9,679	28,152
Results - 100-300 mrad	173	1,176
Results - 300-500 mrad	12	125
Results - Over 500 mrad	11	51
Lost Results	52	82
Average Dose Per Film Packet - mrad (ow)	10.9	9.92
- mr (s)	33.06	27.37
<u>Hanford Neutron Film Badge Dosimeters</u>		
<u>Slow Neutron</u>		
Film Processed	740	3,564
Results - 50-100 mrem	0	1
Results - 100-300 mrem	0	1
Results - Over 300 mrem	0	0
Lost Results	3	3
<u>Fast Neutron</u>		
Film Processed	209	926
Results - 50-100 mrem	38	141
Results - 100-300 mrem	40	171
Results - Over 300 mrem	0	1
Lost Results	0	10
<u>Hand Checks*</u>		
Checks Taken - Alpha	8,120	79,453
- Beta-Gamma	32,454	143,829
<u>Skin Contamination*</u>		
Plutonium	2	26
Fission Products	27	125
Uranium	1	8
Tritium	0	0

*CPD Not Available.

<u>Whole Body Counter</u>	<u>Male</u>	<u>Female</u>	<u>March</u>	<u>1962 to Date</u>
GE Employees				
Routine	17	0	17	62
Special	2	0	2	38
Terminal	12	1	13	42
Non-Routine	20	4	24	83
Non-Employees	5	0	5	6
Pre-Employment	<u>1</u>	<u>2</u>	<u>3</u>	<u>3</u>
	57	7	64	234

Bioassay

Confirmed Plutonium Deposition Cases	2	2*
Plutonium - Samples assayed	409	1,435
- Results above 2.2×10^{-8} $\mu\text{c/sample}$	14	53
Fission Product - Samples assayed	702	1,882
- Results above 3.1×10^{-5} $\mu\text{c/sample}$	9	14
Uranium - Samples assayed	129	599
Biological - Samples assayed	150	172

Uranium Analyses

<u>Sample Description</u>	<u>Following Exposure</u> <u>Units of 10^{-9} $\mu\text{c U/cc}$</u>			<u>Following Period</u> <u>of No Exposure</u> <u>Units of 10^{-9} $\mu\text{c U/cc}$</u>		
			<u>Number</u>			<u>Number</u>
	<u>Maximum</u>	<u>Average</u>	<u>Samples</u>	<u>Maximum</u>	<u>Average</u>	<u>Samples</u>
Fuels Preparation	6.0	2.9	34	5.6	2.5	29
Fuels Preparation**	0	0	0	0	0	0
Hanford Laboratories	71.9	7.1	26	10.0	3.2	31
Hanford Laboratories**	0	0	0	0	0	0
Chemical Processing	28.2	15.2	2	6.7	3.6	4
Chemical Processing**	0	0	0	0	0	0
Special Incidents	2.1	2.1	1	0	0	0
Random	2.7	2.4	2	0	0	0

<u>Tritium Samples</u>	<u>Maximum</u>	<u>Count</u>	<u>Total</u>
Urine Samples			
> 5.0 $\mu\text{c/l}$	34.9	109	
< 1.0 $\mu\text{c/l}$		<u>17</u>	
Samples assayed			201
D ₂ O Samples			
Moderator	435.6	6	
Primary Coolant	107.0	6	
Reflector	391.8	<u>4</u>	
			16
Other Water Samples			
#4138 Med. 2.0%	3.11 $\mu\text{c/ml}$		<u>122</u>
			339

* The total number of plutonium deposition cases which have occurred at Hanford is now 285, of which 205 are currently employed.

** Samples taken prior to and after a specific job during work week.

CalibrationsNumber of Units Calibrated
March 1962 to Date

Portable Instruments

CP Meter	969	3,089
Juno	258	832
GM	548	1,694
Other	205	612
Audits	100	307
	<u>2,080</u>	<u>6,534</u>

Personnel Meters

Badge Film	1,296	4,356
Pencils	0	4,780
Other	247	1,144
	<u>1,543</u>	<u>10,280</u>

Miscellaneous Special Services

Total Number of Calibrations	1,963	2,930
	5,586	19,744

*Carl M. Unruh*for the Manager
RADIATION PROTECTION

CM Unruh:bh

FINANCE AND ADMINISTRATIONACCOUNTINGCost Accounting

Preparation of the Budget for FY 1964 and Revision of Budget for FY 1963 was completed with the submission of required schedules and research and development proposals to Contract Accounting. A summary of the research and development proposals submitted by Hanford Laboratories is shown below:

Research and Development Proposals

<u>AEC Program</u>	<u>Number of Proposals</u>	(In Millions of Dollars)			
		<u>FY 1963</u>		<u>FY 1964</u>	
		<u>R & D</u>	<u>Equipment</u>	<u>R & D</u>	<u>Equipment</u>
<u>Current Programs</u>					
02 Program	2	\$.2	\$	\$.2	\$
04 Program	22	13.7	1.7	14.7	1.9
05 Program	4	1.4	.2	1.4	.1
06 Program	<u>12</u>	<u>3.3</u>	<u>.3</u>	<u>3.6</u>	<u>.3</u>
	<u>40</u>	<u>18.6</u>	<u>2.2</u>	<u>19.9</u>	<u>2.3</u>
<u>Proposed Programs</u>					
02 Program	2	.1		.2	
04 Program	21	3.6	.9	5.9	.7
05 Program	3	.3	.1	.5	.1
06 Program	<u>5</u>	<u>.4</u>	<u>.1</u>	<u>.5</u>	<u>.1</u>
	<u>31</u>	<u>4.4</u>	<u>1.1</u>	<u>7.1</u>	<u>.9</u>
Total	<u>71</u>	<u>\$23.0</u>	<u>\$3.3</u>	<u>\$27.0</u>	<u>\$3.2</u>

Summary data were also compiled on all phases of Hanford Laboratories' budget for use at the HAPO General Manager's Review held on March 26, 1962.

Financial Plan #11 was issued by HOO-AEC. Significant items pertaining to Hanford Laboratories' business are as follows:

1. Additional funds to cover the expanded program on Irradiation Damage to Reactor Metals as planned by HOO-AEC and Hanford Laboratories were not included in the financial plan under the Reactor Fuels and Materials classification. As a result, HOO-AEC has requested reclassification of certain research and development activities within the Reactor Fuels and Materials Program to the Plutonium Recycle Program.

2. The financial plan included an additional \$30,000 for operation of the DR-1 Loop thereby enabling continued operation for the remainder of the fiscal year.
3. Funds were not included in the financial plan to support the additional effort on the Specific Fuel Cycle Analyses Program requested by Washington-AEC. HOO-AEC has assured Hanford Laboratories that funds will be obtained to cover the requested work.
4. As requested by Hanford Laboratories, the financial plan included an increase of \$140,000 in the Special Reactor and Other Special Materials inventory account balance for June 30, 1962, to cover increased zirconium holdings.
5. The financial plan included an additional \$13,000 for research and development sponsored by the Division of Biology and Medicine.

The operating cost control budget for March 1962 was adjusted to reflect the changes included in Financial Plan #11, including the arrangements made by HOO-AEC on the Reactor Fuels and Materials Program. Other revisions in the control budget were:

1. Increase of \$50,000 in CPD sponsored process technology funds.
2. Increase of \$23,000 in IPD sponsored O2 Program research and development for work pertaining to the irradiation of NPR graphite samples in the GETR.
3. Reduction of \$50,000 in O6 Program research and development in response to a request from Washington-AEC to release funds, if possible, for utilization by another AEC site which is badly in need of funds.
4. Transfer of \$46,000 from O6 Program research and development to O6 Program capital equipment to finance equipment requirements for the remainder of FY 1962.

Special requests established during the month were as follows:

<u>Accounting Code</u>	<u>Description</u>
.4V	Participation by Hanford Laboratories with the University of Washington in conducting a summer institute in radiation biology for high school

Accounting
CodeDescription

and college teachers. Incidental costs approximating \$400 will be billed to the University of Washington.

.4W

Review of APED activities in the area of critical mass safety problems by C. L. Brown. Consultation to begin March 12 for a period of two weeks at San Jose with an additional week required for writing report. Estimated costs are \$1,300 plus travel and subsistence.

.4Y

Consultation by J. J. Fuquay with APED on the environmental safety analysis of the Humboldt Bay reactor. Two days of consulting are anticipated with estimated costs of \$300 plus travel and subsistence.

General Accounting

The volume of travel in Hanford Laboratories is currently running at about the same level as the corresponding period of FY 1961. Total trips fiscal year to date is below FY 1961, reflecting a lower level of travel in the first four months of this fiscal year.

Following is a summary of the status of letters or agreements covering specific actions requiring AEC concurrence:

<u>No.</u>	<u>Title</u>	<u>Status</u>
AT-104	Fission Products Dispersal Handbook	Unapproved
AT-198	Miniature Swine for Hammersmith Hospital	Approved 3-2-62
AT-221	Seattle World's Fair - Library Exhibit	In process
AT-223	University of Washington Primate Research Center - Executive Committee	In process
AT-225	University of Washington Request for Data	Approved 3-16-62
AT-227	Participation of Spencer H. Bush in National Academy of Sciences Committee on Metallic Materials	Approved 3-9-62

<u>No.</u>	<u>Title</u>	<u>Status</u>
AT-228	Processing of Film Demonstrating the Alsi Canning Process	Still held by AEC
AT-229	Space Allocation for Universities	Approved 3-16-62
AT-237	Summer Institute in Nuclear Energy for Engineering Faculty	In process

The reconciliation of the physical inventory of movable catalogued equipment in the custody of Chemical Research and Development Operation is progressing on schedule with only equipment items valued at \$3,699 still unlocated.

Hanford Laboratories inventory investment at March 1, 1962 was \$25.8 million as detailed below:

(Amounts in thousands)

SS Material	\$24 429
Reactor and Other Special Materials	1 047
Spare Parts	<u>351</u>
	<u>\$25 827</u>

At March 1, 1962, the reserve for Spare Parts Inventory was \$72,706, which is short \$5,071 of the required 25 per cent established by the AEC. To bring our reserve up to the required amount, Cost was charged with \$5,000 and the reserve credited during March business.

Costs on the following projects were transferred from unclassified account to classified plant accounts:

CGH-785	In Reactor Studies Equipment-105 KW	\$ 321 110
CGH-834	Additional Heat Transfer Facilities 189-D	749 735
CGH-923	Spectroscopy Laboratory - 325 Bldg.	<u>92 585</u>
		<u>\$1 163 430</u>

A total of 137 items valued at \$134,014 was received at the Laboratory Equipment and Material Pool during the month of March. This included two items comprised of U-238 powder-wafers and scrap valued at \$72,422 and Hastelloy valued at \$8,913. Twenty-seven items valued at \$10,516 were loaned or transferred in lieu of placement of requisitions; 31 items valued at \$80,731, including UO₂ powder valued at \$59,770, were withdrawn by custodians. There

are currently 903 items valued at \$522,893 physically located in the storage pool of which 36 items valued at \$86,304 are held for the convenience of others.

A survey is currently being made of equipment which has been in the storage pool in excess of 24 months without activity. Items for which no definite need is foreseen will be excessed. During March, 144 items (\$9,255) were excessed, and 79 items (\$30,143) are awaiting transportation to the excess yard.

Materials in the Laboratory Pool at month end were:

Beryllium	1035 grams	\$ 592
Gold	2402 grams	3 218
Palladium	2180 grams	2 485
Platinum	4245 grams	11 765
Silver	6968 grams	486
Zirconium	16 443.3 pounds	<u>219 297</u>
		<u>\$337 844</u>

All other material held for the convenience of others	<u>\$249 589</u>
	<u>\$587 433</u>

Heavy water losses chargeable to operating cost for the month of March amounted to \$7,650 representing (1) PRTR D₂O BPID loss of 335 pounds valued at \$4,650, (2) 1,486 pounds of scrap generated during March valued at approximately \$3,000. The D₂O average unit price for the month used to compute BPID loss was \$13.88 per pound. During March 13,473 pounds of D₂O having a value of \$185,119 were received from SR00.

The actual D₂O system loss for the month of March was 835 pounds; however, 500 pounds were returned to inventory following removal of oil emulsion and re-analysis. The net for the month was 335 pounds.

Action as indicated occurred on the following projects during the month:

New Money Authorized HLO

CGH-955*	Reactivation of the H-1 Loop, 105-H Building	\$10 000
CGH-957	Small Particle Technology Laboratory, 325 Bldg.	40 000

* Authorized from HLO 04 Equipment Fund Allocation

Physical Completion Notices Issued

AEC-167* Plutonium Recycle Test Reactor
CAH-902* Uranium Scrap Burning Facility
CAH-921* Geological and Hydrological Wells - FY 1961

* AEM Services only.

Construction Completion and Cost Closing Statements Issued

CAH-896* Stress-Rupture Testing Facility
CAH-914* Rattlesnake Springs Radioecology Facility
CAH-921* Geological and Hydrological Wells - FY 1961

* AEM Services only.

An additional allocation of \$12,000 for Miscellaneous Capital Work Orders Under \$20,000 was received during the month. Total allocation is now \$147,000 of which \$127,000 has been expended and committed.

On March 20, 1962, HOO-AEC advised HAPO that certain Miscellaneous Capital Work Orders falling within "sensitive" areas must be discussed informally and agreements reached with HOO-AEC prior to issuance of authorization to perform the work. The "sensitive" areas as defined by HOO-AEC are:

1. Grounds improvements and/or landscaping - all areas
2. Building additions in the 300 Area
3. 300 Area temporary maintenance shops and storage areas

Revised OPGs issued in March are listed below:

<u>OPG</u>	<u>Title</u>
22.1.12	Test Reactor and Auxiliaries Operation Organization
6.4	Indemnification
3.6.1	Plan for Communication in Event of Serious Accident or Emergency at Hanford
3.6.3	Release of Information to the Public
3.6.5	Plant Visits and Tours
3.3.1	Grievance Procedure
3.6.4	Public Information Releases and Articles for Publication or Oral Presentation
3.1.9	Emergency Loan Plan for Ill or Laid-Off Pension Plan Participants
3.4.2	Induction of New Employees

A complete review of OPG 22.3.1, Delegation of Approval Authorization is nearing completion.

The following contracts were processed during the month:

CA-321	FI Badgley
CA-324	Yuba Consolidated Industries Incorporated
CA-320	R. Wells Moulton
DDR-134	Oregon Metallurgical Corporation
SA-207	Great Lakes Carbon Corporation
SA-208	Speer Carbon Company
SA-206	Union Carbide Corporation
DDR-142	Van's Metal Spinning Company

Twenty-four revisions were received to the AEC Manual. The contents of these revisions are covered in a separate report to the Section Managers.

Personnel Accounting

Payroll Statistics

Number of HLO Employees
Changes During Month

	<u>Total</u>	<u>Exempt</u>	<u>Nonexempt</u>
Employees on payroll at beginning of month	1 434	670	764
Additions and transfers in	30	7	23
Removals and transfers out	<u>26</u>	<u>9</u>	<u>17</u>
Employees on payroll at end of month	<u>1 438</u>	<u>668</u>	<u>770</u>

Overtime Payments During Month

	<u>March</u>	<u>February</u>
Exempt	\$ 3 367	\$ 4 068
Nonexempt	<u>20 842</u>	<u>16 771</u>
Total	<u>\$24 209</u>	<u>\$20 839</u>

Gross Payroll Paid During Month

Exempt	\$ 629 995	\$ 620 495
Nonexempt	<u>500 884</u>	<u>398 104</u>
Total	<u>\$1 130 879</u>	<u>\$1 018 599</u>

Participation in Employee Benefit Plans at Month End

	March		February	
	Number	Per Cent	Number	Per Cent
Pension	1 289	99.5	1 283	99.4
Insurance Plan-Personal	361		364	
-Dependent	1 067	99.8	1 059	99.9
U.S. Savings Bonds				
Stock Bonus Plan	88	38.9	91	39.6
Savings Plan	75	5.2	77	5.4
Savings and Security Plan	1 079	89.3	1 078	89.7
Good Neighbor Fund	969	67.4	961	67.1

Insurance Claims

	March		February	
	Number	Amount	Number	Amount
<u>Employee Benefits</u>				
Life Insurance	0	-0-	0	0
Weekly Sickness and Accident	15	\$ 1 033	13	\$ 1 240
Comprehensive Medical	73	5 096	63	4 761
<u>Dependent Benefits</u>				
Comprehensive Medical	141	11 939	132	11 628
Total	229	\$18 068	208	\$17 629

On March 9, 1962, each weekly salaried nonbargaining employee was informed of the change in his salary structure effective April 2, 1962. Individual computations were prepared by Personnel Accounting for the 585 employees involved, showing weekly rate as of April 1, and new weekly rate as of April 2, 1962.

The following indicates the approximate annual increase in payroll for each of our weekly salaried groups:

Weekly Salaried Nonbargaining	\$138 200
Weekly Salaried Bargaining	44 200
Total Annual Increases	\$182 400

Employee Purchase Plan booklets, Appliance Catalogues and Medical Expense Record forms were delivered to each employee on March 16, 1962.

AUDITS AND PROCEDURES

A Contract and Accounting Operation auditing proposal to eliminate departmental identification on letterheads was rejected.

Two additional units of punched card automatic data processing equipment were recommended for purchase. The machines, a verifier and a printing card punch, are presently in use at the 700 Area Bioassay laboratory of Radiation Protection Operation.

A study report of capital budgeting and authorization procedures employed in the Laboratories was issued by the Business Systems Development Operation. Evaluation of recommendations contained therein is being made by Finance and Administration Operation management.

TECHNICAL ADMINISTRATION

Employee Relations

Ten nonexempt employment requisitions were filled during March; 21 remain to be filled.

Professional Placement

Advanced Degree - Eight Ph.D. applicants visited HAPO for employment interviews. Six offers were extended; two acceptances and six rejections were received. Current open offers total six.

BS/MS - Forty-two program offers and five direct placement offers were extended; offers accepted - ten and one, rejections received - fourteen and one, respectively. Current offers open total 125. Eleven schools were visited for recruiting purposes.

Technical Graduate Program - Eight Technical Graduates were placed on permanent assignment; two members terminated from the rolls. Current program members total 44.

Technical Information

A revised format for the HAPO Technical Interchange Report was determined and departments requested to submit contributions.

A revised directory of Technical Information services and an announcement of the availability of various journal style guides were issued.

Microfilming of HW documents in Files vault which were more than 18 months old was completed - 48,213 documents.

FACILITIES ENGINEERINGProjects

At month's end Facilities Engineering Operation was responsible for 11 active projects having total authorized funds in the amount of \$2,672,600. The total estimated cost of these projects is \$7,581,000. Expenditures on these projects through February 28, 1962 were \$952,000. The attached project report details project status.

The following summarizes the status of Facilities Engineering Operation's project activity in March:

Number of authorized projects at month's end - - - - - 11

Number of new projects authorized - - - - - 2

CGH-955 - Reactivation of the H-1 Loop - 105-H Building

CGH-957 - Small Particle Technology Laboratory

Projects completed - - - - - 1

CAH-924 - 200 KW Induction Heating System, 306 Building

New project proposals submitted to the AEC - - - - - 3

CGH-957 - Small Particle Technology Laboratory

CGH-958 - Plutonium Fuels Testing and Evaluation
Laboratories, 308 Building (1)

CAH-959 - Graphite Machining Shop

New projects awaiting AEC authorization - - - - - 2

CAH-917 - Field Service Center - Atmospheric Physics

CAH-959 - Graphite Machining Shop

Project Proposals complete or nearing completion - - - - - 2

Low Level Radiochemistry Building (was re-written during
the month and is now ready for routing for signatures)

Geological and Hydrological Wells - FY 1962 (being routed
for signatures)

(1) Subsequently returned at our request for inclusion of more
detailed justification.

Services

Engineering services during March were provided to the following activities with satisfactory progress shown on each:

Laboratory Modifications - 321-A, 325 and 3706 Buildings
Controlled Environment Facility - 108 Building
Split Half Machine - Critical Mass Laboratory
Secondary Water System - PRTR
Cell Door Drive Modification - Whole Body Monitor
Electrical Load Study - 108-F Building
Humidity Recorder - 144-F Building
Relocate Induction Heating Equipment - 314 Building

Thirty-one equipment requisitions were issued during the month with a value of \$22,000. Total value of material and equipment procurement in process is \$575,000.

Plant Engineering effort was expended on:

Pressure system code work (Identification sketches for PRTR
Rupture Loop, Third Party Inspection of thermal hydraulics
laboratory piping)
209-B Compressor and Motor Control Center
307 Proportional Sampler
325 Ceramic Fuels Area Air Conditioning
Criticality Alarm Central Annunciator

Maintenance and Operation

Costs for February were \$149,300. Costs to date are 98.5% of forecasted expenditures. Improvement maintenance costs for February were \$11,600.

The following tabulation summarizes waste disposal operations:

	<u>February</u>	<u>January</u>
Concrete Barrels	4	12
Loadluggers - Hot Waste	3	3
Milk Pails (up to 2-20-62)	0	23
Gatling Gun	0	1
Crib Waste	220,000 gal.	320,000 gal.

Drafting

The equivalent of 140 drawings were completed during the month. Major jobs in progress are:

Fission Product Packaging, TF-23 Test Loop, 280 Ton Extrusion Press Installation, Welding Apparatus for 8 Ft. Fuel Element, Automatic Densitometer "As Built", PRTR Shim Rods, Vibration Compaction, A-C Columns - 321 Building, Fuel Element Rupture Test Facility - PRTR.

Construction

There were 64 existing J. A. Jones Company orders at the beginning of the month with a total unexpended balance of \$121,230. One hundred forty-six new orders, five supplements and adjustments for underruns amounted to \$124,009. Expenditures during the month on Hanford Laboratories' work were \$103,148 (includes C. O. cost). Total J. A. Jones backlog at month's end was \$142,081.

	<u>Summary</u>		<u>CE&UO</u>	
	<u>No.</u>	<u>HL Unexpended Balance</u>	<u>No.</u>	<u>Unexpended Balance</u>
Orders outstanding beginning of month	63	\$118 991	1	\$2 239
Issued during the month (Inc. Supp. and Adj.)	146	124 009		
J. A. Jones expenditures during month (inc. C.O. costs)		101 318		1 840
Balance at month's end	79	141 081	1	399
Orders closed during month	130	91 046*		
*Face value of orders closed				

Construction activities completed during March were:

- 141-H Building - Concrete block wall enclosure
- 141-C Building - Entrance enclosure
- 141-C Building - Concrete pads and feeding stations
- 209-E Building - Paint fire hydrant post barricades
- 307 Building - Continuous sampler facility
- 325 Building - Install 1220 Dynapack unit
- 325 Building - Office and foyer partitions

J. L. Boyd
for Manager
Finance and Administration

W Sale:whm

SEMI - MONTHLY PROJECT STATUS REPORT						DATE 3-31-62	
GENERAL ELECTRIC CO. - Hanford Laboratories						FUNDING	
PROJ. NO. CAH-822	TITLE Pressurized Gas Cooled Facility					4141 Operating	
AUTHORIZED FUNDS		DESIGN \$ 43,000	AEC \$ 50,000	COST & COMM TO 3-11-62		\$ 1,104,273	
\$ 1,170,000		CONST. \$ 1,127,000	GE \$ 1,120,000	ESTIMATED TOTAL COST		\$ 1,170,000	
STARTING DATES	DESIGN 8-19-59	DATE AUTHORIZED 2-2-62*	EST'D. COMPL. DATES	DESIGN 4-29-60	PERCENT COMPLETE		
	CONST. 10-17-60	DIR. COMP. DATE 6-30-62		CONST. 6-30-62	WT'D.	SCHED.	ACTUAL
ENGINEER TR&AO-MEEO - DP Schively					DESIGN	100	100
					TITLE I	100	100
MANPOWER					GE-TIT. II		
FIXED PRICE					AE-TIT. II		
COST PLUS FIXED FEE							
PLANT FORCES					CONST.	100	91
ARCHITECT - ENGINEER					PF	1.4	0
DESIGN ENGINEERING OPERATION					CPFF	22.1	99
GE FIELD ENGINEERING					FP	6.6	100
					Govt. Ed.	69.9	90
SCOPE, PURPOSE, STATUS & PROGRESS							
<p>The contract for design-fabricate of the new heater (DDR-143) has been signed by Struther-Wells. Detail design is scheduled to start immediately with shipment scheduled approximately eight weeks following.</p> <p>A recent telephone call to Bristol-Siddeley has disclosed that the two blowers in their possession will not be ready to test for five to six weeks despite our willingness to relax design point conditions. Expediting efforts have been increased.</p> <p>*Initial authorization date was 12-18-58.</p>							

SEMI - MONTHLY PROJECT STATUS REPORT						DATE 3-31-62	
GENERAL ELECTRIC CO. - Hanford Laboratories						FUNDING	
PROJ. NO. CAH-842	TITLE Critical Reactivity Measuring Facility					58-e-15	
AUTHORIZED FUNDS		DESIGN \$ 45,000	AEC \$ 178,000	COST & COMM. TO 3-18-62		\$ 182,267 (GE)	
\$ 400,000		CONST. \$ 355,000	GE \$ 222,000	ESTIMATED TOTAL COST		\$ 400,000	
STARTING DATES	DESIGN 11-17-59	DATE AUTHORIZED 11-30-61*	EST'D. COMPL. DATES	DESIGN 2-1-61	PERCENT COMPLETE		
	CONST. 10-3-60	DIR. COMP. DATE 3-15-62		CONST. 3-15-62	WT'D.	SCHED.	ACTUAL
ENGINEER TR&AO-MEEO - WS Kelly					DESIGN	100	100
MANPOWER					TITLE I		
FIXED PRICE					GE-TIT. II		
COST PLUS FIXED FEE					AE-TIT. II		
PLANT FORCES							
ARCHITECT - ENGINEER					CONST.	100	100
DESIGN ENGINEERING OPERATION					PF		
GE FIELD ENGINEERING					CPFF	60	100
					FP	40	100
SCOPE, PURPOSE, STATUS & PROGRESS							
<p>Construction is 100% complete with exceptions. These include cleanup, piping, gas storage bottle station, and such corrective work as may be uncovered by the design tests which are in progress.</p> <p>*Initial authorization was on 4-9-59.</p>							

GENERAL ELECTRIC CO - Hanford Laboratories

DATE 3-31-62

****Includes delivery charges, inspection and contingency.**

SEMI-MONTHLY PROJECT STATUS REPORT							DATE 3-31-62	
GENERAL ELECTRIC CO. - Hanford Laboratories								
PROJ. NO.	TITLE					FUNDING		
CGH-858	High Level Utility Cell - 327 Building					0290		
AUTHORIZED FUNDS		DESIGN \$	50,000	AEC \$	-	COST & COMM TO		3-18-62 \$ 348,918
\$ 400,000		CONST. \$	350,000	GE \$	400,000	ESTIMATED TOTAL COST		\$ 1,000,000
STARTING DATES	DESIGN 11-1-59	DATE AUTHORIZED 4-6-61*		EST'D. COMPL. DATES	DESIGN 2-15-61	PERCENT COMPLETE		
	CONST. 5-15-61	DIR. COMP. DATE 6-1-62			CONST. 6-1-62		WT'D.	SCHED. ACTUAL
ENGINEER						DESIGN	100	100
FEO - KA Clark						TITLE I		
MANPOWER				AVERAGE	ACCUM MANDAYS	GE-TIT. II	95	100
FIXED PRICE				7	675	AE-TIT. II		
COST PLUS FIXED FEE						Vendor	5	100
PLANT FORCES						CONST.	100	100
ARCHITECT-ENGINEER					35	PF		
DESIGN ENGINEERING OPERATION					930	CPFF	100	100
GE FIELD ENGINEERING				1	32	FP		70
SCOPE, PURPOSE, STATUS & PROGRESS								
<p>This project will provide facilities to prepare specimens from irradiated materials for use in determining their physical and mechanical properties and involves the installation of a cell in 327 Building.</p> <p>Current estimate of Title I and II costs is \$62,000. Detailed design started 4-1-60. Procurement and construction authorized 4-6-61.</p> <p>The project proposal requesting extension of the completion date to 6-1-62 has been approved by the HOO-AEC. A new schedule based on this completion date has been requested by HOO-AEC. It is in preparation.</p> <p>Number of purchase orders required 12 Value \$200,000 Number of purchase orders placed 12 Value 200,000</p> <p>The Decladder, Flat Tensile Sample Former and two Manipulators are other items of equipment not delivered. They are all expected by 5-1-62. (Decladder 3-31-62, Flat Tensile Sample Former 5-1-62 and Manipulators 4-15-62).</p> <p>The cell castings have all been received and the cell is erected. Installation of equipment has begun.</p> <p>Modification No. 3 of Directive HW-499 extends project completion date to June 1, 1962.</p> <p>*Original authorization for design was October 1, 1959.</p>								

SEMI-MONTHLY PROJECT STATUS REPORT						DATE 3-31-62	
GENERAL ELECTRIC CO. - Hanford Laboratories							
PROJ. NO. CAH-867		TITLE Fuel Element Rupture Test Loop				FUNDING 58-e-15	
AUTHORIZED FUNDS \$ 1,500,000		DESIGN \$ 130,000	AEC \$ 820,000	COST & COMM. TO 3-18-62		\$ 533,936	
		CONST. \$ 1,370,000	GE \$ 680,000	ESTIMATED TOTAL COST		\$ 1,500,000	
STARTING DATES	DESIGN 8-1-60 CONST. 11-2-60	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN 3-15-61 CONST. 6-30-62	PERCENT COMPLETE		
					WT'D.	SCHED.	ACTUAL
ENGINEER					DESIGN	100	100
TR&AO-MEEO - PC Walkup					TITLE I		
MANPOWER FIXED PRICE 10 2220 COST PLUS FIXED FEE 5 1930 PLANT FORCES ARCHITECT-ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING					GE-TIT. II	91	100
					AE-TIT. II	9	100
					CONST.	100	92
					PF	2	100
					CPFF	57	100
					FP (1)	10	100
					(2)	31	75
SCOPE, PURPOSE, STATUS & PROGRESS (1) G. A. Grant Company (2) Lewis Hopkins Construction Company This facility is to be used for fuel rupture behavior studies with respect to physical distortion and rate of fission product release. Project is behind official schedule because of delays in delivery of material.							

PROJ. NO.		TITLE				FUNDING	
AUTHORIZED FUNDS		DESIGN \$	AEC \$	COST & COMM. TO		\$	
\$		CONST. \$	GE \$	ESTIMATED TOTAL COST		\$	
STARTING DATES	DESIGN	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN	PERCENT COMPLETE		
	CONST.	DIR. COMP. DATE		CONST.	WT'D.	SCHED.	ACTUAL
ENGINEER					DESIGN	100	
					TITLE I		
MANPOWER FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT - ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING					GE-TIT. II		
					AE-TIT. II		
					CONST.	100	
					PF		
					CPFF		
					FP		
SCOPE, PURPOSE, STATUS & PROGRESS							

SEMI-MONTHLY PROJECT STATUS REPORT

Project No. 100 - Hanford Laboratories

PROJECT NO.	100		TITLE	Biology Laboratory Improvements		PROJECT NO.	100-h-1	
ESTIMATED FUNDS	DESIGN \$	100,000	SEC \$	159,500	COST + COMM TO	3-10-60	\$	59,519 (GE)
	CONST. \$	375,000	GE \$	60,500	ESTIMATED TOTAL COST			425,000
DESIGN	8-3-60	DATE AUTHORIZED	4-18-61*	EST'D. COMPL. DATES	DESIGN 3-31-61	PERCENT COMPLETED		
CONST.	7-10-61	DIR. COMP. DATE	3-31-62		CONST 4-30-62		WT'D.	SCHED.
MANPOWER						DESIGN	100	NS
MANPOWER						TITLE I		
MANPOWER						GE-TIT. II	17	NS
MANPOWER						AE-TIT. II	83	NS
MANPOWER						CONST.	100	100
MANPOWER						PF	1	85
MANPOWER						CPFF	10	NS
MANPOWER						FP	89	100
MANPOWER								

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides additional space for biological research supporting services, and involves an addition to the 108-F Building.

The construction contract has been extended to April 18, 1962.

Installation of partitions is complete. Painting is 80% complete. All lighting fixtures are installed on first floor and well along on second floor. Vinyl floor covering is being laid in animal rooms. Insulation is being placed on piping and ductwork. A.T.P.'s are being run on some items. Allied Engineering has given us a new date of April 9, 1962 for start of Radiation Handling Equipment test at their plant.

The AEC has revised the work authority to General Electric Company to include installation of the Radiation Handling Equipment.

*Original authorization for design was May 3, 1960.

SEMI-MONTHLY PROJECT STATUS REPORT						DATE 3-31-62		
GENERAL ELECTRIC CO. — Hanford Laboratories								
PRQJ. NO. CAH-916		TITLE Fuels Recycle Pilot Plant				FUNDING 4-62-d-3		
AUTHORIZED FUNDS \$ 385,000		DESIGN \$ 385,000	AEC \$	COST & COMM TO 3-18-62	\$ 321,952			
		CONST. \$ -0-	GE \$ 385,000	ESTIMATED TOTAL COST		\$ 5,000,000		
STARTING DAYS	DESIGN 3-15-61	DATE AUTHORIZED 10-27-61**	EST'D. COMPL. DATES	DESIGN 8-15-62	PERCENT COMPLETE			
	CONST. 6-1-62*	DIR. COMP. DATE --		CONST. 11-1-64	WT'D.	SCHED.	ACTUAL	
ENGINEER FEO - RW Dascenzo					DESIGN	100	58	50
					TITLE I	11	100	100
<u>MANPOWER</u> FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT-ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING					GE-TIT. II	89	53	53
					AE-TIT. II	0		
					CONST.	100	0	0
					PF			
					CPFF			
					FP			
AVERAGE					ACCUM MANDAYS			
					30			
SCOPE, PURPOSE, STATUS & PROGRESS This project is to provide a facility to perform a full scope of engineering tests and pilot plant studies associated with fuel reprocessing concepts. The revised design schedule has been approved by AEC. A revised project proposal is being prepared to request total project funds. Design is continuing on all phases as scheduled. Satisfactory design progress was made during the month. A total of 165 drawings have been issued for comment and 16 for approval. *Estimated construction starting date for removal of burial ground fill. **Original authorization for initiation of design was February 9, 1961.								

SEMI-MONTHLY PROJECT STATUS REPORT						HW 73102	
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 3-31-62	
PROJ. NO.		TITLE				FUNDING	
CAH-917		Field Service Center-Atmospheric Physics				61-j	
AUTHORIZED FUNDS		DESIGN \$	AEC \$	COST & COMM. TO		\$	
\$		CONST. \$	GE \$	ESTIMATED TOTAL COST		\$ 154,000	
STARTING DESIGN DATES	5-15-62*	DATE AUTHORIZED		EST'D. COMPL. DATES	DESIGN 9-15-62*	PERCENT COMPLETE	
CONST. 11-1-62*		DIR. COMP. DATE			CONST. 7-15-63*	WT'D. SCHED. ACTUAL	
ENGINEER				FEO - JT Lloyd		DESIGN 100	
MANPOWER				AVERAGE		TIT. II	
FIXED PRICE				ACCU MANDAYS		AE-TIT. II	
COST PLUS FIXED FEE						CONST. 100	
PLANT FORCES						PF	
ARCHITECT-ENGINEER						CPFF	
DESIGN ENGINEERING OPERATION						FP	
GE FIELD ENGINEERING							
SCOPE, PURPOSE, STATUS & PROGRESS							
This project will provide facilities necessary to conduct atmospheric physics research and development programs.							
The project proposal was submitted to the AEC on January 23, 1961.							
There has been no change in the status of this project for some time.							
*Based on AEC authorization by 4-15-63.							

PROJ. NO.		TITLE				FUNDING	
CAH-922		Burst Test Facility for Irradiated Zirconium Tubes				62-k	
AUTHORIZED FUNDS		DESIGN \$	AEC \$	COST & COMM. TO		\$	
\$ 29,600		CONST. \$	GE \$ 29,600	ESTIMATED TOTAL COST		\$ 298,000	
STARTING DESIGN DATES	11-7-61	DATE AUTHORIZED	10-23-61	EST'D. COMPL. DATES	DESIGN 5-31-62	PERCENT COMPLETE	
CONST. 6-15-62		DIR. COMP. DATE	To be estab-		CONST. 12-1-62	WT'D. SCHED. ACTUAL	
ENGINEER				lished at a later date		DESIGN 100	
FEO - KA Clark						TIT. II	
MANPOWER				AVERAGE		ACCU MANDAYS	
FIXED PRICE				3		100	
COST PLUS FIXED FEE				1		60	
PLANT FORCES						CONST. 100	
ARCHITECT-ENGINEER						PF	
DESIGN ENGINEERING OPERATION						CPFF	
GE FIELD ENGINEERING						FP	
SCOPE, PURPOSE, STATUS & PROGRESS							
This project will provide facilities to permit deliberate destructive testing of irradiated zirconium tubing. This will provide operating and tube life data not now available because of the limited operating history of Zircaloy-2 pressure tubing in reactors.							
The revised design schedule is awaiting approval.							
Design decisions are being made with particular regard to high construction costs of recent projects and test data from prototype experimental runs.							

SEMI-MONTHLY PROJECT STATUS REPORT						DATE 3-31-62		
GENERAL ELECTRIC CO. — Hanford Laboratories								
PROJ. NO. CAH-924		TITLE 200-KW Induction Heating System - 306 Building				FUNDING 0290		
AUTHORIZED FUNDS \$ 31,000		DESIGN \$ 3,750	AEC \$ 24,650	COST & COMM. TO 3-18-62		\$ 24,587		
		CONST. \$ 27,250	GE \$ 6,350	ESTIMATED TOTAL COST		\$ 26,000		
STARTING DATES	DESIGN 5-1-61 CONST. 12-4-61	DATE AUTHORIZED 3-27-61 DIR. COMP. DATE 2-28-62	EST'D. COMPL. DATES	DESIGN 10-15-61 CONST. 3-20-62	PERCENT COMPLETE			
ENGINEER FEO - LF Higginson <u>MANPOWER</u> FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT-ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING					DESIGN	100	100	100
					TITLE I			
					GE-TIT. II		100	100
					AE-TIT. II			
					CONST.	100	100	100
					PF			
					CPFF			
					FP			
					AVERAGE	ACCUM MANDAYS		
						120		
						38		
SCOPE, PURPOSE, STATUS & PROGRESS This project will provide a source of power for induction heating for R&D work in the 306 Building. The physical completion notice is being processed. All work was completed with exceptions on March 20, 1962. The exceptions include installation of a larger cooling water lines and functional check of remote work stations. This project will no longer be reported.								

PROJ. NO.		TITLE				FUNDING	
AUTHORIZED FUNDS		DESIGN \$	AEC \$	COST & COMM. TO		\$	
\$		CONST. \$	GE \$	ESTIMATED TOTAL COST		\$	
STARTING DATES	DESIGN CONST.	DATE AUTHORIZED DIR. COMP. DATE	EST'D. COMPL. DATES	DESIGN CONST.	PERCENT COMPLETE		
ENGINEER <u>MANPOWER</u> FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT - ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING					DESIGN	100	
					TITLE I		
					GE-TIT. II		
					AE-TIT. II		
					CONST.	100	
					PF		
					CPFF		
					FP		
					AVERAGE	ACCUM MANDAYS	
SCOPE, PURPOSE, STATUS & PROGRESS							

SEMI-MONTHLY PROJECT STATUS REPORT						HW-73202	
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 3-31-62	
PROJ. NO. CAH-927		TITLE Additions to the 271-CR Building-Waste Treatment Demonstration Facility				FUNDING 61-j	
AUTHORIZED FUNDS \$ 80,000		DESIGN \$ 11,000 CONST. \$ 69,000		AEC \$ 64,300 GE \$ 15,700		COST & COMM TO 3-18-62 \$ 14,896 ESTIMATED TOTAL COST \$ 93,000	
STARTING DATES	DESIGN 6-15-61 CONST. 2-15-62	DATE AUTHORIZED 5-16-61 DIR. COMP. DATE 6-30-62	EST'D. COMPL. DATES	DESIGN 2-5-62 CONST. 8-1-62	PERCENT COMPLETE		
					WT'D.	SCHED.	ACTUAL
ENGINEER FEO - KA Clark					DESIGN	100	100
					TITLE I		
MANPOWER FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT-ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING					AVERAGE	ACCLM MANDAYS	
					2	80	
						150	
					GE-TIT. I	100	100
					AE-TIT. I		
					CONST. *	100	8 10
					PF		
					CPFF		
					FP		
SCOPE, PURPOSE, STATUS & PROGRESS This project provides facilities for pilot plant development of decontamination processes for intermediate level chemical processing plant waste for safe discharge to the plant environs. Design was accomplished by the Bovay Engineers. A revised project proposal has been prepared by HOO-AEC requesting total project funds of \$93,000 and completion date extension to 8-1-62. Walls have been poured for the cell and storage tank room to 2 ft. above grade. Drain line installation is complete outside building. *Project planning schedule is used pending approval of the revised project proposal. The official revised schedule will be submitted upon receipt of the modified directive.							

SEMI-MONTHLY PROJECT STATUS REPORT									
GENERAL ELECTRIC CO. - Hanford Laboratories								DATE 3-31-62	
PROJ. NO. CAH-936		TITLE Coolant Systems Development Laboratory 1706-KE Building Addition						FUNDING 62-k	
AUTHORIZED FUNDS \$ 93,000		DESIGN \$ 9,000 CONST. \$ 84,000		AEC \$ 78,000 GE \$ 15,000		COST & COMM. TO 3-18-62 \$ 14,687		ESTIMATED TOTAL COST \$ 130,000	
STARTING DATES DESIGN 9-8-61 CONST. 4-20-62		DATE AUTHORIZED 10-27-61* DIR. COMP. DATE 10-31-62		EST'D. COMPL. DATES DESIGN 1-1-62 CONST. 10-31-62		PERCENT COMPLETE			
ENGINEER FEO - KA Clark						WT'D.			
						SCHED.			
						ACTUAL			
						DESIGN 100 100 100			
						TITLE I			
						GE-TIT. II 100 100 100			
						AE-TIT. II			
MANPOWER						AVERAGE			
FIXED PRICE						ACCU MANDAYS			
COST PLUS FIXED FEE						166			
PLANT FORCES									
ARCHITECT-ENGINEER									
DESIGN ENGINEERING OPERATION									
GE FIELD ENGINEERING									
SCOPE, PURPOSE, STATUS & PROGRESS									
<p>This project provides facilities for the conduct of corrosion and decontamination studies for nuclear reactor coolant systems, by the addition of 2,700 sq. ft. laboratory facility on the west side of the 1706-KE Building. Design was accomplished by the Bovay Engineers.</p> <p>Bids were opened 3-14-62. The low bid was \$105,300. The fair cost estimate was \$82,000.</p> <p>A review of the fair cost estimate is being made. Further action will depend on the result of this review.</p> <p>Current estimate of Title I & II costs - \$11,000.</p> <p>*Original authorization for design was August 9, 1961.</p>									

SEMI-MONTHLY PROJECT STATUS REPORT									
GENERAL ELECTRIC CO. - Hanford Laboratories								DATE 3-31-62	
PROJ. NO. CGH-937		TITLE Safety Improvements to 231-Z Building						FUNDING 61-j	
AUTHORIZED FUNDS \$ 45,000		DESIGN \$ 5,500 CONST. \$ 39,500		AEC \$ GE \$ 45,000		COST & COMM. TO 3-18-62 \$ 44,226		ESTIMATED TOTAL COST \$ 45,000	
STARTING DATES DESIGN 7-12-61 CONST. 7-10-61		DATE AUTHORIZED 6-29-61 DIR. COMP. DATE 5-15-62		EST'D. COMPL. DATES DESIGN 9-29-61 CONST. 2-20-62		PERCENT COMPLETE			
ENGINEER FEO - JT Lloyd						WT'D.			
						SCHED.			
						ACTUAL			
						DESIGN 100 NS 100			
						TITLE I			
						GE-TIT. II NS 100			
						AE-TIT. II			
MANPOWER						AVERAGE			
FIXED PRICE						ACCU MANDAYS			
COST PLUS FIXED FEE						4 460			
PLANT FORCES									
ARCHITECT-ENGINEER									
DESIGN ENGINEERING OPERATION									
GE FIELD ENGINEERING									
SCOPE, PURPOSE, STATUS & PROGRESS									
<p>This project provides supplemental ventilation and installation of a fire detection system in the 231-Z Building.</p> <p>Fire detection system is complete.</p> <p>Final inspection was made February 16, 1962. The building was accepted with minor exceptions. All exceptions are completed except the wiring change to motors to cut off air supply. This wiring change will not be done therefore, the project is complete. A physical completion notice is being prepared and this project will no longer appear in this report.</p>									

SEMI-MONTHLY PROJECT STATUS REPORT						HW-73202		
GENERAL ELECTRIC CO. — Hanford Laboratories						DATE 3-31-62		
PROJ. NO. CGH-951		TITLE A-C Column Facility - 321 Building				FUNDING 0290		
AUTHORIZED FUNDS \$ 55,000		DESIGN \$ 5,000	AEC \$ -0-	COST & COMM. TO 3-18-62		\$ 2,684		
		CONST. \$ 50,000	GE \$ 55,000	ESTIMATED TOTAL COST		\$ 55,000		
STARTING DATES	DESIGN 1-30-62	DATE AUTHORIZED 1-12-62	EST'D. COMPL. DATES	DESIGN 4-1-62	PERCENT COMPLETE			
	CONST. 3-15-62	DIR. COMP. DATE 10-31-62		CONST. 10-31-62	WT'D.	SCHED.	ACTUAL	
ENGINEER FEO - OM Lyso MANPOWER FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT-ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING					DESIGN	100	95	95
					TITLE I			
					GE-TIT. II	100	95	95
					AE-TIT. II	0		
					CONST.	100		
					PF			
					CPFF			
					FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide a closely integrated "A" column in series with the relocated "C" column to permit the development of a mathematical model for the mass transfer of uranium, as well as the exploration of the possibilities of computer optimization of a combined "A-C" extraction battery.

Specifications for procured items are being prepared and when available, purchase requisitions will be prepared.

PROJ. NO. CGH-955		TITLE Reactivation of the H-1 Loop - 105-H Building				FUNDING 0490		
AUTHORIZED FUNDS \$ 10,000		DESIGN \$ 10,000	AEC \$	COST & COMM. TO		\$		
		CONST. \$	GE \$ 10,000	ESTIMATED TOTAL COST		\$ 105,000		
STARTING DATES	DESIGN 4-15-62	DATE AUTHORIZED 3-29-61	EST'D. COMPL. DATES	DESIGN 8-30-62	PERCENT COMPLETE			
	CONST. 7-15-62	DIR. COMP. DATE - - -		CONST. 12-15-62	WT'D.	SCHED.	ACTUAL	
ENGINEER FEO - OM Lyso MANPOWER FIXED PRICE COST PLUS FIXED FEE PLANT FORCES ARCHITECT - ENGINEER DESIGN ENGINEERING OPERATION GE FIELD ENGINEERING					DESIGN	100		
					TITLE I			
					GE-TIT. II			
					AE-TIT. II			
					CONST.	100		
					PF			
					CPFF			
					FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide the primary test facility for determination of the feasibility of using aluminum-clad fuel elements in high temperature water by studying improved alloys and corrosion inhibitors.

The project proposal was submitted to the AEC February 21, 1962.

Title I & II design funds have been requested from the AEC. This will provide a better basis for a construction estimate. The AEC questioned the construction and contingency figure which was based on scope only and no actual design. Design funds in the amount of \$10,000 have been authorized for initiation of design and preparation of a review of estimated costs.

SEMI - MONTHLY PROJECT STATUS REPORT						HW - 73202	
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 3-31-62	
PROJ. NO. CGH-957	TITLE Small Particle Technology Laboratory - 325 Building					FUNDING 62-k	
AUTHORIZED FUNDS \$ 40,000	DESIGN \$ 2,000	AEC \$ --	COST & COMM. TO \$ -0-	ESTIMATED TOTAL COST \$ 40,000			
STARTING DATES	DESIGN 4-9-62	DATE AUTHORIZED 3-21-62	EST'D. COMPL. DATES	DESIGN 6-1-62	PERCENT COMPLETE		
	CONST. 7-1-62	DIR. COMP. DATE 11-1-62		CONST. 11-1-62	WT'D.	SCHED.	ACTUAL
ENGINEER FEO - DS Jackson					DESIGN	100	NS
<u>MANPOWER</u>					TITLE I		
FIXED PRICE					GE-TIT. II		
COST PLUS FIXED FEE					AE-TIT. II		
PLANT FORCES					CONST.	100	
ARCHITECT - ENGINEER					PF		
DESIGN ENGINEERING OPERATION					CPFF		
GE FIELD ENGINEERING					FP		
SCOPE, PURPOSE, STATUS & PROGRESS							
<p>This project provides laboratory space for research and development in small particle technology related to the generation, control, and disposal of radioactive wastes.</p> <p>The project proposal was submitted to the Commission for approval on March 2, 1962.</p> <p>Directive HW-535, dated March 21, 1962, authorized total project funds in the amount of \$40,000.</p> <p>Initiation of design is being scheduled.</p>							

PROJ. NO. CGH-958	TITLE Plutonium Fuels Testing and Evaluation Laboratory-308 Bldg.					FUNDING 62-k	
AUTHORIZED FUNDS \$	DESIGN \$	AEC \$	COST & COMM. TO \$	ESTIMATED TOTAL COST \$ 132,000			
	CONST. \$	GE \$					
STARTING DATES	DESIGN 6-1-62*	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN 9-1-62*	PERCENT COMPLETE		
	CONST. 9-1-62*	DIR. COMP. DATE		CONST. 12-15-62*	WT'D.	SCHED.	ACTUAL
ENGINEER FEO - OM Lyso					DESIGN	100	
<u>MANPOWER</u>					TITLE I		
FIXED PRICE					GE-TIT. II		
COST PLUS FIXED FEE					AE-TIT. II		
PLANT FORCES					CONST.	100	
ARCHITECT - ENGINEER					PF		
DESIGN ENGINEERING OPERATION					CPFF		
GE FIELD ENGINEERING					FP		
SCOPE, PURPOSE, STATUS & PROGRESS							
<p>This project provides for the extension of plutonium research laboratories on the second floor of 308 Building by erection of plastered ceilings and walls to provide contamination control barriers. It also includes laboratory service extension and fabrication of a metallography hood.</p> <p>The project proposal was submitted to the Commission for approval March 7, 1962.</p> <p>Additional justification for subject work is being prepared for submission to the AEC.</p> <p>*Based on AEC authorization by May 1, 1962.</p>							

SEMI-MONTHLY PROJECT STATUS REPORT						HW- 73202	
GENERAL ELECTRIC CO. - Hanford Laboratories						DATE 3-31-62	
PROJ. NO. CAH-959		TITLE Graphite Machining Shop - 300 Area				FUNDING 62-k	
AUTHORIZED FUNDS		DESIGN \$		AEC \$		COST & COMM. TO \$	
\$		CONST. \$		GE \$		ESTIMATED TOTAL COST \$ 105,000	
STARTING DATES	DESIGN 6-1-62*	DATE AUTHORIZED	EST'D-COMPL. DATES	DESIGN 9-15-62*	PERCENT COMPLETE		
	CONST. 11-15-62*	DIR. COMP. DATE		CONST. 6-1-63*	WT'D.	SCHED.	ACTUAL
ENGINEER					DESIGN	100	
FEO - M Lyso					TITLE I		
MANPOWER					GE-TIT. II		
FIXED PRICE					AE-TIT. II		
COST PLUS FIXED FEE							
PLANT FORCES					CONST.	100	
ARCHITECT-ENGINEER					PF		
DESIGN ENGINEERING OPERATION					CPFF		
GE FIELD ENGINEERING					FP		
AVERAGE					ACCUM MANDAYS		
<p>SCOPE, PURPOSE, STATUS & PROGRESS</p> <p>This project provides for a new graphite machining facility near the graphite storage building. The facility will permit greater flexibility in the handling and machining of graphite shapes as well as providing additional space for non-metallic materials testing in the area presently used for graphite machining.</p> <p>The project proposal was submitted to the Commission for approval March 2, 1962.</p> <p>Additional justification has been requested for the proposed work.</p> <p>*Based on AEC approval by May 1, 1962.</p>							

PROJ. NO.		TITLE				FUNDING	
AUTHORIZED FUNDS		DESIGN \$		AEC \$		COST & COMM. TO \$	
\$		CONST. \$		GE \$		ESTIMATED TOTAL COST \$	
STARTING DATES	DESIGN	DATE AUTHORIZED	EST'D-COMPL. DATES	DESIGN	PERCENT COMPLETE		
	CONST.	DIR. COMP. DATE		CONST.	WT'D.	SCHED.	ACTUAL
ENGINEER					DESIGN	100	
					TITLE I		
MANPOWER					GE-TIT. II		
FIXED PRICE					AE-TIT. II		
COST PLUS FIXED FEE							
PLANT FORCES					CONST.	100	
ARCHITECT - ENGINEER					PF		
DESIGN ENGINEERING OPERATION					CPFF		
GE FIELD ENGINEERING					FP		
AVERAGE					ACCUM MANDAYS		
<p>SCOPE, PURPOSE, STATUS & PROGRESS</p>							

TEST REACTOR AND AUXILIARIES
MONTHLY REPORT - MARCH 1962REACTOR DEVELOPMENT - O4 PROGRAMPLUTONIUM RECYCLE PROGRAMPlutonium Recycle Test ReactorOperation

Reactor output was 605 MWD for a plant efficiency of 26.8% and a total experimental time efficiency of 33%. Accumulated exposure to-date is 6,217 MWD. Additional exposure to-date information is as follows:

Maximum UO ₂ exposure/element	1930 MWD/TU
Average UO ₂ exposure/element	1288 MWD/TU
Maximum Pu-Al exposure/element	79.0 MWD
Average Pu-Al exposure/element	39.4 MWD

The eighth refueling was accomplished on March 6, 1962. Four new LX Pu-Al elements were charged around the outside of the Pu-Al zone. The loading configuration of 39 Pu-Al and 46 UO₂ elements remained as before. The shim-free critical level was 65.8". The status of the various test fuel elements at the end of March is as follows:

PRTR Test Number	Tube Location	Element Number	Description	Date Charged	Date Discharged	Accumulated MWD
5	1653	1501	UO ₂ -Tubular	11-3-61	--	45.1
10	1051	1082	UO ₂ -Hot Swaged	11-3-61	--	25.6
10	1857	1067	UO ₂ -Vipac	11-3-61	--	28.2
13	1253	5092	Pu-Al Instrumented	12-3-61	--	40.8
13	1544	5093	Pu-Al Instrumented	12-3-61	--	35.8
13	1556	5095	Pu-Al Instrumented	1-17-62	--	32.0
13	1247	5051	Pu-Al Instrumented	1-17-62	2-7-62	13.1
13	1847	5052	Pu-Al Instrumented	1-17-62	--	29.5

D₂O and Helium losses were 650 pounds and 140,500 scf, respectively.

There were seven reactor scrams during the month, all occurring at low power level. Four were spurious flux monitor trips during shim rod calibration tests, one was a flow monitor trip, one was a spurious shutdown valve trip, and one was a pressurizer level trip during raise to full power.

There was one major outage during the month. Significant work included replacement of primary pump #2 seal, replacement of bearings on the moderator and reflector pumps, adjustment of gas gap on process tube #1857, replacement of 3U flux chamber, replacement of helium compressor diaphragms, and rework of the shim control system. The outage was extended approximately one week, primarily because of pressurizer level instrumentation problems and pressurizer over-pressure control valve leakthrough problems.

Equipment Experience

Bearings on moderator pump #3 failed, resulting in motor damage, and has not been returned to service. Bearings were replaced on moderator pumps #1 and #2.

Primary pump #2 was replaced because of seal failure. The fission chamber in position 3U was replaced.

The gas gap on process tube 1857 was found to be reduced to 0.035" minimum compared to 0.270" for concentric tubes. The gap was increased to 0.090" by rotating the tube 180°. Tube rotation was hampered by high radiation fields, inaccessible flange bolts, and sheared bolts resulting from galled threads. The discharge hook of the fueling vehicle was damaged following a fall when the cable shackle broke.

The pressurizer overpressure control valve failed, requiring in place machining of the valve body and installation of a gasket between the modified body and the stellite seat.

Preventive maintenance required 560 man-hours or 11.4% of total craft effort.

Improvement Work Status (Significant items)

Work completed:

- Interlock between biological shield door and bypass for charge-discharge hoist.
- Primary flow low flow instrumentation (permanent position).
- Primary pump lubrication control change.
- Deactivate relay from startup channel to eliminate possibility of unsafe bypassing.
- Interlock between charge-discharge machine shroud seat and discharge hoist.

Work partially completed:

- New helium unloading station valves.
- Safety circuit ground and low voltage detector.
- Outlet nozzle cap modification - increase of 7 for a total of 26.
- Automatic condensate drain from HX-7.
- Relocate pressurizer level transmitters.
- Primary pump bowl vent line.
- Gas bottle storage facility.

Design work completed:

- Rerouting of primary system D₂O injection.
- Primary oxygen analyzer modification.
- Install dual valves on helium stack lines.
- Enlarge chemical feed system (secondary system)
- Modification of primary pump power cables.
- Shim rod readout modification.

Design work partially completed:

- Decontamination facility.

Process Engineering and Reactor Physics

Critical Test Results. The preliminary issue of the critical test results document has been issued as HW-61900-BA.

Two test proposals were received during the month and are being evaluated for performance. These proposals covered the irradiation of six instrumented UO_2 fuel elements and a series of graphite oxidation experiments under shutdown gamma flux.

The following work was done on PRTR tests:

PRTR Test Number 26 (Shim Rod Recalibration) was performed. Three shim assemblies, one from each ring, were calibrated in terms of change in moderator level.

PRTR Test Number 27 (Moderator Level Coefficient) was performed. The level coefficient was measured at full height (approximately 104") using the method of partial water height. Seven measurements of the reactor rising period were made, at different increments of moderator height.

PRTR Test 32 (Shim Worth Measurement) was performed. The length of operating period, estimated from these measurements of shim worth, has been in fair agreement.

PRTR Test Number 28 (Degasser Efficiency During Reactor Operation) Performance was completed.

PRTR Test Number 29 (Flow Monitor Changes During Reactor Operation) Supplement Number 1 was issued authorizing the taking of five more sets of data. The first set was obtained.

PRTR Test Number 31 (Rate of Temperature Buildup in Thermal and Biological Shields and Lower Access Room Wall). Performance was completed and the final report issued.

PRTR Test Number 33 (Functional Test of S-80 Valve). Performance was completed.

PRTR Test Number 36 (Total Power Failure Test). The test was outlined and the test sections assigned to TR&AO components for preparation.

Procedures

Revised Operating Procedures issued	36
Revised Operating Standards issued	16
Temporary Deviations to Operating Standards issued	6
Revised Process Specifications accepted for use	2
Maintenance manuals issued	1
Maintenance procedures issued	2

Significant program maintenance checksheets:

All motor control centers		
All flux monitor equipment		
Specialized helium PRV units		
Drawing As-built status	<u>March</u>	<u>Total</u>
Approved for as-built	97	307
Ready for approval		58
In drafting		170
Voided		16

Personnel training was as follows:

Qualification subjects	151 Man-Hours
Specifications, Standards, Procedures	222
Fueling Vehicle	26
Unit Motion	8
Critical Facility	17
	<u>424 Man-Hours</u>

Status of Qualified Personnel at Month-end:

Qualified Reactor Engineers	9
Provisionally Qualified Reactor Engineers	1
Qualified Technicians	6
Qualified Technologists	18

Project Status

Second Communication System - work started and is 20% complete.

Fuel Element Examination Facility - received replacement gears and replacement is under way.

PRTR Shipping Cask - Chemical Processing Department requested changes were completed.

PRTR Electrical Power Supply - Study was initiated to determine need for improvements in the 13.8 KV supply to PRTR.

Plutonium Recycle Critical FacilityProject Status (Project CAH-842)

Work was completed with exceptions. Cadmium plating thickness of safety rods were under test. Replacement sections are on order. Installation of temperature detectors was under way. Design test status is as follows:

Electrical	95%
Instrumentation	80%
Fuel Handling	20%
Moderator System	20%
Thimble Coolant	20%

Operation

A new component, Plutonium Recycle Critical Facility Operation, was established, reporting to the Manager, Test Reactor and Auxiliaries, to operate the Critical Facility.

The Operating Procedures and a Training Guide were issued. Startup Physics Test descriptions were given preliminary approval and issued. Work continued on other startup test preparations, qualification questions, data sheets, handbook inserts, and preparation of Design Change descriptions.

Fuel Element Rupture Test Facility

Project Status (Project CAH-867)

Overall construction is estimated at 88%. The waterplant contract is approximately 75% complete and the CPFF work is 97% complete. The installation of equipment and piping insulation in B Cell is one of the major items of work remaining.

Electrical design tests and acceptance test procedures continued during the month. Electrical design tests on the makeup pumps (RLP-4A, 4B) and the deaerator vacuum pump were completed. The sump pump ATP was completed.

The entire makeup system was transferred to General Electric Company. The makeup ion exchangers were successfully regenerated.

Detail design of the mechanical components required for cooling of the test section during discharge is essentially complete. Most components are in the fabrication phase. An initial design of shielding for the heat exchanger complex has been completed. Design of the storage basin cleanup system was completed; drawings and specifications were released to purchasing.

Concrete placement for the water treatment plant is essentially 100% complete. The wheeler filter bottoms were completed during the month. The outfall sewer at the river was completed.

Operation

Two PRTR0 personnel are employed full time on startup preparations.

Acid and caustic handling procedures, safety rules, and job hazard breakdowns were established. All rupture loop personnel participated in the initial regeneration of the makeup ion exchangers.

The valve book was completed and distributed. Handbook inserts await duplication. Work continued on the instrument manual and preparation of visual training aids.

GAS COOLED POWER REACTOR PROGRAMGas Cooled LoopProject Status (CAH-822)

The project continues at 91% complete based on the new schedule. Construction forces work will not be resumed until receipt of the new heater scheduled for May.

The design-fabricate contract (DDR-143) for the new heater was signed by the vendor. Material was forwarded to the vendor.

Expediting effort with Bristol-Siddeley for the blowers has been increased. However, the earliest shipment date for the blowers is June 30.

Performance of design tests continued.

Operation

A valve list and diagram, reflecting the updated flow diagram, was distributed. The instrument manual has been completed and is being reproduced. A listing of information for the handbook has been completed and awaits reproduction. The visual training aid for the heater was completed. Chapter 52 of the Technical Manual, "Pressurized Gas-Cooled Loop" has been prepared in handwritten form and is being circulated for comments. Each shift received copies of four basic gas loop documents for initial training purposes. Work has begun on an outline of subjects to be covered during formal training sessions. Availability of the necessary operating supplies for loop operation was confirmed. Carbon dioxide, carbon monoxide, and oxygen measuring and detecting devices were ordered. These devices will be used to monitor the air in B Cell including the sump area.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 22,418 hours. This includes 18,167 hours performed in the Technical Shops, 3,589 hours assigned to Minor Construction, 269 hours assigned to off-site vendors, and 393 to other project shops. Total shop backlog is 20,899 hours, of which 60% is required in the current month with the remainder distributed over a three-month period. Overtime hours worked during the month was 3.5% (836 hours) of the total available hours.

Distribution of time was as follows:

	<u>Man-Hours</u>	<u>% of Total</u>
Fuels Preparation Department	3,340	14.90%
Irradiation Processing Department	3,462	15.44%
Chemical Processing Department	271	1.21%
Hanford Laboratories	15,308	68.28%
Construction Engineering and Utilities	37	.17

Requests for emergency service decreased, requiring 3.5% overtime ratio compared to 4.2% ratio for the previous period.

WD Richmond:bk
April 12, 1962

WD Richmond

UNCLASSIFIED

INVENTIONS OR DISCOVERIES

All persons engaged in work that might reasonably be expected to result in inventions or discoveries advise that, to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during the period covered by this report except as listed below. Such persons further advise that, for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

INVENTORTITLE OF INVENTION OR DISCOVERY

R. K. Koler

Pneumatic Injection Casting of Aluminum-Plutonium Fuel Elements (PRTR)
HW-68964 dated March 24, 1961.H. E. Hanthorn and
H. Harty

Direct Energy Conversion

H. Harty

High Temperature, Molten Fueled Reactor

D. R. Green (with M. C. Lambert,
Chemical Research and Develop-
ment)Indicator for Stress Relief in Metals
and Maximum Temperatures Attained
During Heat Treatment (HWIR-1498)

L. C. Amos

A Composition of Matter

L. C. Amos

Centrifuge

James Dunn

Solution Clarifying Agitator-Pump

R. W. Perkins and
D. E. RobertsonThe Reduction of Radioisotope Con-
centrations in Reactor Effluent Water
by Addition of Silicates (HW-73014)G. E. Benedict and
J. L. SwansonSeparation of Plutonium from Uranium
and Trivalent Rare Earths in Molten
Chloride Salt Solutions by Precipitation
of Plutonium Oxide (HW-73079)
for Manager, Hanford Laboratories

END

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