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HANFORD LABORATORIES OPERATION MONTHLY ACTIVITIES REPORT

MAY, 1962

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JUNE 15, 1962

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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HANFORD LABORATORIES OPERATION
MONTHLY ACTIVITIES REPORT
MAY, 1962

DECLASSIFIED

By Authority of CG-PR-2

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By JL Phillips 8-13-92

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Compiled by
Operation Managers

June 15, 1962

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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PRELIMINARY REPORT

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

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TABLE I - HLO FORCE REPORT

DATE: May 31, 1962

	At Beginning of Month		At Close of Month		Total
	Exempt	Salaried	Exempt	Salaried	
Chemical R & D	131	123	133	127	260
Reactor & Fuels R & D	172	156	171	156	327
Physics & Instrument R & D	93	59	95	60	155
Biology	35	54	34	58	92
Operations Res. & Syn.	17	4	18	4	22
Radiation Protection	40	90	39	93	132
Finance and Administration	100	99	95	97	192
Programming	17	3	16	2	18
General	3	4	3	4	7
Test Reactor & Auxiliaries	49	183	51	183	234
TOTAL	657	775	655	784	1,439

BUDGETS AND COSTS

May operating costs totaled $2, 084, 000, an increase of $27, 000 from the previous month; fiscal year-to-date costs are $24, 050, 000 or 86% of the $27, 888, 000 control budget. Hanford Laboratories' research and development costs for May, compared with last month and the control budget are shown below:

(Dollars in Thousands)	COST Current Month	Previous Month	FY To-Date	Budget	% Spent
HLO Programs					
02 Program	$ 46	$ 46	$ 488	$ 605	81
03 Program	52	6	85	175	49
04 Program	716	746	9 650	11 024	88
05 Program	76	78	870	1 065	82
06 Program	223	204	2 231	2 637	85
	1 113	1 080	13 324	15 506	86
FPD Sponsored	82	95	1 183	1 400	85
IPD Sponsored	106	83	1 207	1 348	90
CPD Sponsored	105	117	1 445	1 636	88
Total	$ 1 406	$ 1 375	$ 17 159	$ 19 890	86

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

Ball bearing supports currently in use on prototypic N Reactor "perfs" severely scratch Zircaloy-2 process tubes. Flexible low carbon steel supports are being evaluated as replacements.

The experimental single tube, dual enriched fuel element was examined after successfully completing one cycle of irradiation in the ETR. The element appeared to be in excellent condition with no visible indication of damage or corrosion.

An analysis of cladding strain data obtained from Zircaloy-2 clad uranium rods irradiated in NaK capsules has been made. At cladding temperatures below 325 - 350 C, the maximum strain without failure

[redacted]

is approximately 1.5 per cent, but at 400 C, the maximum observed cladding strain without failures increases to ca. 8%. A lesser amount of annealing of irradiation damage at the lower temperatures is considered to be a major factor limiting the uniform strain capabilities.

The prototype NPR tube-in-tube fuel element has been recharged into the GEH-M3 Loop in the ETR and is operating satisfactorily. At the present time, the average exposure is approximately 1000 MWD/T. The test will be terminated at the end of the current operating cycle, at which time the average exposure will be approximately 1200 MWD/T and the maximum exposure will approach 2000 MWD/T.

Experimental extrusions do not confirm the hypothesis that the best quality N Reactor fuel coextrusion will result from matching the extrusion coefficient of the clad to that of the uranium. In the presence of the large grain size of the uranium, stiff cladding alloys (Zircaloy-2 or mild steel) with a small grain size form the smoother interfaces with less over-all variation in wall thickness.

Zircaloy discs irradiated in a simulated NPR gas atmosphere for 167 days at temperatures of 290, 350 and 400 C have been retrieved and weighed. Corrosion rates for irradiated etched samples were generally two to three times those of samples exposed to the same atmosphere out of the reactor. Contrary to expectations, samples pre-autoclaved in 400 C steam corroded in-reactor at about the same rate as the etched samples.

Average graphite burnout rates for KE Reactor have decreased by a factor of three during the latest monitoring period. The decrease apparently results from a decrease in reactor gas flow rate from 2000 cfm to about 800 cfm or less.

In conjunction with a measurement of the rate of gamma-induced oxidation of graphite in air, a series of 10 gamma dosimetry measurements were made in the PRTR in tube 1556 from which the fuel had been discharged. Oxidation data were consistent with expectations, but more tests must be completed to establish a rate equation.

Preliminary analysis of data from tests to determine fuel temperatures during the shutdown transients following tube plugging or a failure of an inlet connector at K Reactor indicates that fuel jacket melting will not occur following such an event at an initial tube power of 1250 KW if the rear header pressure is 40 psig or higher.

The NPR charging machine was tested using a full-size process tube and prototypical fuel elements. The machine performed satisfactorily, but it was found that the fuel elements scratched the tube to depths of 0.3 to 1.1 mils over a considerable portion of its length. A separate charge of dummy spacer elements did not scratch the tube. Causes and corrective measures are under study.

High energy compaction of UO_2 was accomplished with the new Dynapak machine, using ten-pound containers. Single pieces of UO_2 weighing a pound or more can be separated from the compacted materials, and after crushing, can be readily sintered to greater than 99 per cent of theoretical density.

Spacing members were successfully attached to the small diameter cladding for a nested tubular fuel element by high voltage electron beam welding, off-site. There was no appreciable distortion of the eight-foot long tube.

Accurate calibration of the transmission electron microscope high temperature stage reveals that temperatures of approximately 2900 C are now achieved.

Six hundred UO_2-PuO_2 fuel element rods have been fabricated and 18 clusters assembled and readied for the PRTR during the past two months.

A cold swaged MgO-PuO_2 cluster has been completed. A ZrO_2-PuO_2 spike element, a stainless steel clad UO_2-PuO_2 element, a segregated plutonium element, and a zirconium clad Pu-Zr alloy plate element are being fabricated for early PRTR testing.

Radiometallurgical examination of the 42-inch long UO_2-PuO_2 cosine enriched cluster irradiated to 1100 MWD/T in the ETR revealed no detrimental defects.

ZrO_2-PuO_2 and MgO-PuO_2 fuel capsules have successfully completed irradiation in the MTR.

The total gas release from swage compacted UO_2-PuO_2 rods irradiated in the ETR high temperature loop does not indicate any discernible difference between outgassed and unoutgassed fuel material under the test conditions.

Corrosion coupons were removed after 220 days of exposure in the primary heat exchanger of the PRTR. The 304 stainless steel suffered negligible corrosion. The A212B C/S surfaces were pitted especially in the stressed portion of the U-bend sample.

A room temperature burst test was performed on a second sample from the first pressure tube removed from the PRTR. Failure occurred at about 85 per cent of the anticipated ultimate strength when a short crack penetrated the wall. The crack did not propagate further, indicating that the bulk metal behaved in a ductile manner. The origin of the crack was a thin hydrided layer in the bottom of a wear corrosion mark caused by contact between the pressure tube and an outer bundle wire wrap of a UO_2 fuel element.

Evidence that wear corrosion is continuing at points of contact between the PRTR pressure tube and the fuel element was found in 16 tubes examined during the month. A 12-mil deep wear corrosion mark was measured in one tube.

INOR-8, a nickel base alloy developed by Oak Ridge as a container vessel for molten fluoride salts, is being considered as a candidate for inclusion in the Irradiation Effects on Reactor Metals Program. This alloy, manufactured by the Haynes-Stellite Company under the trade name Hastelloy N, has excellent oxidization resistance in air and considerable strength at temperatures to 1800 F. The alloy is also easily fabricated and might be used to advantage in gas cooled reactor applications.

During reactor operation a creep rate below 5×10^{-7} in/in/hr was observed for a 20 per cent cold worked Zircaloy-2 creep specimen stressed to 30, 000 psi at 250 C. In this in-reactor test total deformation appears to be greater than in an identical test being run ex-reactor. Most of the strain occurs during the reactor down periods. This behavior is similar to results of the test at 310 C on the same material.

Fifty-four Zircaloy-2 tensile specimens irradiated in the ETR at 280 C to about 3×10^{19} nvt were tested during the month. Fifteen of these tests were performed at 300 C. Irradiated properties at room temperature and at 300 C were markedly influenced by the anisotropy of this material.

Field ion microscopy as a means for detecting neutron damage in molybdenum is being evaluated in a joint HLO-Linfield Research Institute effort. Work has begun at Linfield to prepare single crystal emitters suitable for irradiation and post-irradiation study.

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The safeguards analysis for the PRTR Fuel Element Rupture Facility (HW-61236 SUP5) was published and transmitted to AEC.

Fifteen experimental boiling burnout points were determined in the laboratory with an electrically-heated model of a 19-rod bundle fuel element with 0.015 inch spacing between rods. The boiling burnout heat fluxes were found to be considerably less than those obtained with a similar test section with 0.074 inch spacing between rods when compared at the same bulk coolant conditions. There were indications that the coolant mixing was quite poor with the 0.015 inch spaced test section.

2. Physics and Instruments

NPR control rod strengths, as measured in a new exponential pile with correct spacing between rods, are about two per cent less than the theoretically predicted values.

Analyses of future exponential pile experiments for the N lattice will be simplified by development of a consistent set of extrapolation distances.

Total cross-section measurements with pyrolytic graphite showed a dependence upon direction of preferred orientation for neutron energies below 0.028 ev. This information is of particular help in the design of the proposed High Temperature Lattice Testing Reactor.

A new series of critical mass experiments was begun on effects of surrounding a spherical container of plutonium nitrate with a water reflector; these data are needed for correlation with data from early Hanford experiments.

A split-half machine for criticality measurements on plutonium oxide-plastic mixtures is being installed at the Critical Mass Laboratory. Experiments with it will begin after completion of a Hazards Summary report.

Satisfactory agreement has now been obtained between the calculated and measured values for the Pu-239 concentration in H_2O which gives $k_\infty = 1$; the new HISMC Monte Carlo code computation value is 8.0 gm/1, compared to the PCTR experimental value of 8.4 ± 1 gm/1.

The transport theory analysis of a plutonium metal-solution system simulating the dissolution of plutonium metal has been completed, and the data put in a form suitable for nuclear safety application.

Authorization was received from IPD to design and procure the necessary instrumentation for planned studies on fuel failure detection to be conducted at the PRTR Fuel Test Loop.

Progress on the development of instruments for measuring the displacement of installed reactor process tubes included (1) successful demonstration of the Mark IV optical device and an IPD decision to use it for measuring horizontal displacements of tubes in B, D, and F reactors, (2) design of a new model which will reduce indicated displacement errors by a factor of four, and (3) favorable response to a proposal to develop a model with electrical readout for NPR use.

Technical specifications and design criteria are being prepared for the NPR Simulator and the proposed new comprehensive HAPO Analog Simulation Laboratory of which it will be a major part.

An unusual, miniature probe, eddy current, nondestructive test was successfully developed and applied on a short time schedule to inspect small diameter Inconel tubes which are already installed through concrete for use as NPR fuel failure monitor sample lines.

Further theoretical analyses of PCTR studies on Pu-Al fuel in graphite lattices reproduced the experimental neutron flux traverses, but the derived k_∞ value did not agree with experimental results.

In the analysis of 5 w/o Pu-Al rods in H_2O experiments, the derived k_∞ values agreed within $\pm$ 2% with measured values for lattice spacings in the range 0.75 to 1.5 inches. The neutron spectrum will be analyzed by more sophisticated methods in an attempt to reduce these differences.

Techniques for making neutron spectrum measurements on PRTR fuel during irradiation were advanced by trial irradiations of lutetium foils in the PRTR and accompanying theoretical calculations.

Resonance integral data on Pu-240 have been analyzed for a range of Pu-240 concentrations which reduce the effective resonance integral value by a factor of five.

It was estimated that complete loss of coolant in the PRTR loaded with green moxtyl fuel would result in a reactivity loss of 16.9 mk.

In fuel cycle code development, a revision to the Monte Carlo collision routine modifying the treatment of collisions with bound scattering centers was inserted in the RBU code and partially debugged. Preliminary runs on CALX, the new point burnup code, indicates satisfactory performance. Additional work on data tape generation and debugging of parts of the code are required before it will be operational.

In other fuel cycle analysis effort, optimization work on the non-linear aspects of nuclear fuel-cycle analysis progressed through formulation, FORTRAN programming, and check-out of a versatile complex field subroutine for generating, differentiating, and integrating the entire family of doubly-periodic elliptic functions. These functions will be used in solving the isotope balance equation in fuel cycle analysis.

The recently developed liquid effluent gamma monitor satisfactorily passed all laboratory tests and is ready for installation in the PRTR containment trip circuit. A second generation prototype with electronic rather than contact-meter trip circuits is being built.

In the extended lifetime Neutron Flux Monitor program, it was established that further studies of U-234, U-238, and Pu-240 will encompass a sufficient range of cross-sections to give a good evaluation of use of fertile nuclides in regenerating type detectors.

In the development of ultrasonic tests for fuel sheath tubing, under the AEC/AECL cooperative program, further data were obtained to guide the preparation of equipment specifications, and basic measurements of ultrasonic velocities were made to aid comparison of experimental data with Lamb-wave theory.

A decision was reached to terminate the study on electron induced Radiation Effects in Graphite at the end of FY-1962. A terminal report is being prepared.

Testing of the miniature signaling dosimeters under development established that one model will operate reliably up to accumulated exposures of about 1000 r. An alternate readout method was devised and is being evaluated.

[redacted]

Portions of the instrumentation for a portable mast system for atmospheric physics studies were completed and initially tested. All major design modifications required to make the Atmospheric Physics radiotelemetry system operational were completed.

All laboratory test work was completed on the "CPD" Automatic Conveyor Laundry Monitor for alpha-beta-gamma detection. Plutonium contamination spots of 1000 d/m are consistently detected and spots as low as 200 d/m are detected when the geometry is favorable.

Analysis of atmospheric dispersion data will be facilitated by the development of a numerical method for spectral analysis of wind velocity fluctuations in the range of 1 to 16 cycles per hour. These are considered "meander" frequencies in dispersion analyses.

Preparations were completed to begin the measurement of body burdens of radioactivity in Alaskan eskimos in early June at the first location, Kotzebue.

Techniques were developed for use of the positive ion accelerator for neutron activation.

The discrepancy between the calculated power output of a piece of plutonium and the power output measured calorimetrically was confirmed by calorimetric measurements of the same piece at Mound Laboratory. Because of the confirmation of previous work provided by this result, a serious effort to find the cause of the discrepancies will be made.

3. Chemistry

The presence of radioiodine in reactor effluent water, presumed to originate from defective fuel elements, obviates the ready interpretation of data obtained by the AS-76 monitor. A carbon tetrachloride extraction treatment of the monitor sample appears to be a satisfactory method of eliminating the radioiodine interference.

The in-reactor study of the effect of addition of silicate on the production of radioisotopes in the effluent water was interrupted after five days of operation. A twofold reduction in the concentrations of P-32, Mn-56, As-76, Cr-51, Np-239 and Zn-65 was observed.

Laboratory studies show that degraded Purex solvent (Soltrol) is not the primary source of surface active agents which result in poor uranium transfer rates and, presumably, the mal-operation of the 2D column. Extracts from material collected on filters downstream from the Purex ion exchange water treatment beds did, however, cause an appreciable reduction in uranium transfer rates.

No fuel element jamming was observed during the charging of 16 buckets of simulated overbore production slugs into a mockup of the Redox multi-purpose dissolver.

Less than one-half of the fission product tritium present in irradiated Purex and Redox fuel elements could be accounted for in waste streams discharged to ground during the month of November, 1961.

The removal of I-131 from simulated plant gas streams by charcoal is substantially impaired in the presence of an oil mist.

Clinoptilolite continues to show promise for the selective removal of cesium from Purex formaldehyde treated waste. Using simulated solutions, fission product decontamination factors of 2,000, $>$ 1,000, $>$ 1,000 and 10 were observed for strontium, cerium, ruthenium and niobium, respectively.

Laboratory studies continue to show promise for the near quantitative recovery of neptunium from Purex formaldehyde treated waste by a single batch extraction with D2EHPA in Soltrol. The neptunium extraction is not significantly altered in the presence or absence of TBP or by the substitution of Soltrol by Shell Spray Base.

Laboratory studies show that the conventional four-step Purex strontium recovery flowsheet can be reduced to a two-step process with a substantial increase in processing capacity and a decrease in cumulative strontium loss.

Engineering studies were initiated to determine the hydraulic and thermal behavior of coating waste during in-tank solidification processes. Synthetic coating waste was successfully solidified in a four-foot diameter by 15-inch deep insulated tank by a simulated hot-air sparging technique.

A hot cell experiment with the supernatant from a Purex waste tank shows the nickel ferrocyanide precipitation technique to be capable of achieving the high (95-98 per cent) removal of cesium desired in the Waste Management Program. Phase separation in this experiment was through centrifugation at 700 G in a continuous overflow, solid-bowl centrifuge and should, therefore, realistically stimulate attainable plant performance.

Capability for processing of cesium by solvent extraction methods (using dipicrylamine in nitro-benzene) was strengthened by laboratory studies which show that use of a dilute scrub solution of an ammonium salt allows a low-sodium cesium product to be produced in a single solvent extraction cycle. The quality of the product from this test is more than adequate for achieving the desired high loadings of cesium on Decalso for shipment or on inorganic "resins" for interim storage.

Electrodes fabricated from pyrolytic graphite are found superior to those of reactor grade graphite for the electrolytic preparation of UO_2 from molten chloride salt systems.

Acidification of steam-stripped, unfiltered Purex tank farm condensate waste significantly improves the hydraulic characteristics of an Amberlite 200 ion exchange column. Over 6,000 column volumes of condensate waste were treated, and all radioisotopes except Ru-106 were effectively removed during pass age through the column.

Performance tests of the electrostatic bubble scrubber made during a radiant heat spray calciner run were excellent. The decontamination factor across the scrubber was in excess of 100 (analytical detection limit).

A mechanism of radiation induced hemolysis of human red blood cells was postulated, reduced to an equation, and tested experimentally with striking agreement.

4. Biology

On April 30 most of the fish population being held in river water at 146-FR was killed, evidently by chlorinated water backing down through pipes from the IPD water treatment plant.

Liver and kidney mitochondria were found to require ATP for SR-85 and Ca-45 absorption. Liver mitochondria also required carrier calcium. When calcium was present, magnesium seemed to be required by both.

Young pigs do not discriminate against strontium in favor of calcium as do older animals. This confirms earlier observations from research on rats, but this time included transfer studies of Sr and Ca via sow's milk.

DTPA can reduce body burden of Np by about 1/2.

By comparing effects of injecting Pu-238 and Pu-239, chemical toxicity of Pu is being shown to be more important than generally believed.

Seven days after intradermal injection into a pig of 1 μc of $Pu(NO_3)_4$ translocation of the Pu to the lymph nodes was 11 per cent, to the liver 7 per cent, and to bone 5 per cent.

Clearance of radionuclides from sheep plasma following a single intravenous administration was fastest for U-233, slowest for Np-237 and Pu-239, with Cm-244, Am-241, and Ca-45 intermediate. Uranium, Am, and Cm plasma concentration first decreased, then increased, presumably due to initial deposition in the liver and subsequent release. The milk to plasma ratio was highest for Ca-45 (10-40), followed by Cm-244 and Am-241 (2-5), and lowest for U-233, Pu-239, and Np-237 (0.02-0.15).

0.03 to 1.0 nc $Pu^{239}O_2$ originally deposited in lungs of mice did not alter life span or cause abnormal pathology.

Fecal Pu content relates less variably with Pu lung burden in dogs than urine. Analyses of data from this experiment should be of value in interpreting human body burdens following PuO_2 inhalation.

5. Programming

A month ago it was reported that the maximum lifetime for Phoenix fuel appeared to be on the order of 15 years because of neutron capture in americium-241 formed by decay of plutonium-241. It is now recognized that the situation is not this pessimistic because the Pu-239 and Pu-240 neutron absorption resonances coincide rather closely with the Am-241 absorption resonances, so that neutron absorption in the Am-241 resonances will be substantially reduced in the presence of high concentration of the two plutonium isotopes. Results are being recomputed with this refinement.

In conventional plutonium recycle depleted uranium may be mixed with purchased plutonium and fabricated into a fuel element. In an alternative process, entitled the Reuse cycle, fuel elements made from depleted uranium are enriched by being irradiated in the blanket of a fast reactor and are then inserted into a thermal reactor. After irradiation in the thermal reactor the fuel elements may be re-enriched by further irradiation in the fast reactor blanket.

An attempt is being made to evaluate the economic incentives for developing the Reuse cycle without delving into fast reactor economics. Preliminary results indicate that if the thermal reactor is located in the vicinity of a large fast reactor and if the thermal reactor has provision for charging fuel while the reactor is operating, the Reuse cycle may lower fuel costs on the order of one mil per kilowatt hour.

TECHNICAL AND OTHER SERVICES

No new cases of plutonium deposition were confirmed by bioassay analyses during May. The total number of plutonium deposition cases that have occurred at Hanford is 288, of which 207 are currently employed.

The sporadic increases in concentrations of air-borne fallout materials noted in April did not continue into May. The spring peak of materials generated in the 1961 USSR tests was somewhat lower in the Pacific Northwest than predicted. No positive evidence of fallout from the 1962 U. S. tests has been found to date at Hanford. The average fallout concentration in air was 3 $\mu\mu c/m^3$ during May in contrast to the averages of 5 to 6 $\mu\mu c/m^3$ for the first four months of this year.

Iodine-131 emitted from the Purex stack returned to normal ($<$ 0.2 curies per day) after last month's high values of 44 curies in 10 days.

Columbia River flow rates exceeded 15,000 cfs. No significant changes in river pollution or temperatures were observed. Seasonal increases in manganese-56 and sodium-24 transport rates persisted throughout the month.

In 100 per cent testing situations, the bases for setting specifications often take into account only the cost to the producer. That is, the specification is set to reject the worst α per cent, where α is chosen such that the cost is not "excessive." A document was issued, coauthored with FPD personnel, pointing out that in setting specifications at many fuel element test stations, it is also possible to estimate the cost to the consumer. This being the case, a logical basis for setting specifications is the minimization of producer's and consumer's cost.

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Progress reports on the IPD Operations simulation study were made during the month to reactor managers, maintenance managers, and reactor analysts.

Density and homogeneity tests on actual fuel elements whose particle sizes and size distributions had been predicted by theoretical considerations confirmed their exceptional quality. Additional calculations are now being made for annular and nested annular elements.

Document HW-73537, "A Theory of Control for δ - ω Rotary Lathes," has been issued. This document develops the appropriate mathematical relationships which exist between the polar coordinate design specifications of a desired surface of revolution and the controlling positioning and timing variables of a δ - ω lathe on which it is to be machined.

SUPPORTING FUNCTIONS

PRTR output was 654 MWD for a plant efficiency of 30% and a total experimental time efficiency of 49.9%. Accumulated exposure through May 31, is 7867 MWD. Additional exposure information is as follows:

Maximum UO_2 exposure/element	2440 MWD/TU
Average UO_2 exposure/element	1370 MWD/TU
Maximum Pu-Al exposure/element	76.3 MWD
Average Pu-Al exposure/element	47.0 MWD
Maximum Moxtyl exposure/element	22.8 MWD
Average Moxtyl exposure/element	10.5 MWD

The tenth refueling was performed during the extended outage that began on April 30, 1962. Six UO_2 elements that are instrumented with Cobalt-Zircaloy wire for flux monitoring were charged in ring 1 under PRTR Test 37. Seven new mixed oxide elements were charged in ring 9. The core was composed of 39 Pu-Al elements, 35 UO_2 elements and 11 mixed oxide elements at startup on May 20. Two additional fresh LX Pu-Al elements and one additional fresh mixed oxide element were charged during a short outage on May 25.

The purpose of the extended outage (April 30 to May 20) was to perform some of the modification and improvement work noted below. Operation was without incident. No scrams occurred. D_2O and helium losses were 2120 pounds (4-16 to 5-25 inventories) and 73,500 scf, respectively. A total of 20,900 pounds D_2O equivalent was shipped to SRP for reprocessing.

The pump occupying No. 1 position in the PRTR was replaced during the month when high leakage (over one gallon per minute) persisted. The seal exhibited minor damage in the form of a faint flow path across the face. The raised face had worn approximately .002 inch from an original 0.18 inch. No indications of eminent failure were noted although numerous chips were displaced from the seal inside edge. The pump bearings were rough, which is believed to account for high amperage experience. Bearings were replaced.

Work continued on design tests and items initiated as a result of startup tests on the Plutonium Recycle Critical Facility.

Over-all construction of the Fuel Element Rupture Test Facility is estimated at 95% complete. CPFF work is 97% complete and the water plant 92%. All construction work halted on May 16 because of the strike. Design of shielding for piping and heat exchangers is 90% complete.

The project for the Gas Cooled Loop is 91% complete. Approval drawings were received for review from the heater vendor. Bristol-Siddeley continued to experience difficulties with the gas lubricated bearings for the blowers and testing was delayed. Fifty per cent of the PRTR operating personnel have completed formal classroom training in the operation of the facility, and maintenance craftsmen received 166 man-hours of training.

Total productive time for the Technical Shops was 23,807 hours. This includes 18,581 hours performed in the shops, 3761 hours assigned to minor construction, 853 hours assigned to off-site vendors, and 612 hours to other project shops. Total shop backlog is 21,518 hours, of which 70 per cent is required in the current month with the remainder distributed over a three-month period. Overtime hours worked during the month were 7 per cent of the total available hours.

PRTR operating costs in May included heavy water charges of $30,600 to cover losses of $29,280 and scrap of $1,320. This month's loss represents an increase of $9,140 compared with last month's loss. Heavy water scrap returned to Savannah River Operations Office during May aggregated 20,910 pounds and was valued at $253,800.

Seventeen Ph. D. applicants visited HAPO for employment interviews. Eighteen offers were extended; six acceptances and three rejections were received. Two program and thirteen direct placement offers were made to BS/MS applicants. Program results included eighteen acceptances and sixty-one rejections. Direct placement offer responses produced three acceptances and ten rejections.

Hanford Laboratories will edit and publish on a quarterly basis the results of studies of Irradiation Effects on Reactor Structural Materials performed at Hanford and other sites.

Nineteen Hanford Laboratories' displays are being furnished for use in the Visitors' Center soon to be opened in the Richland Community House.

Carl A. Bennett
for Manager
Hanford Laboratories

HM Parker:CAB:st

[redacted]

REACTOR AND FUELS RESEARCH AND DEVELOPMENT OPERATION

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 2000 PROGRAM

1. METALLURGY PROGRAM

Corrosion Studies

Effect of Hydrogen in Excess of Solubility on Corrosion and Hydrogen Absorption. Long term experiments have been started to investigate the effect of precipitated hydride phase on corrosion and hydrogen absorption. A 0.030-inch sheet of Zircaloy-2 was charged with 400 ppm hydrogen and "soaked" at 750 C to homogenize the hydrogen. Metallography and hot extraction analysis confirmed the uniform distribution of hydride phase. This sheet was cut into coupons, which were etched and placed in 360 C and 400 C autoclaves along with control samples which had the same heat treatment, but no added hydrogen. Weight gains after three weeks exposure show both groups of samples corroding at the same rate agreeing with literature references, that this level of hydrogen does not affect corrosion. Hydrogen data are not yet available. Enough samples are included to continue this experiment for a year pulling samples every twenty days.

Metallurgy Studies

Zircaloy Components for Coextrusion. A composite N-outer fuel billet assembly with copper, brass, Zircaloy, and mild steel strips inserted into slots on the outer uranium surface was successfully coextruded. A preliminary analyses of the interfaces formed indicate that the relatively large grain size of the uranium will cause extreme interface roughness and clad variation when coextruded with metal that has approximately the same extrusion coefficient (copper and brass). Conversely, stiffer metals (Zircaloy or mild steel) with a small grain size will predominate in forming the contour of the interface with less roughness and variation being produced. Thus, Zircaloys with a lower alloy content and extrusion coefficient than Zircaloy-2 would apparently result in greater interface roughness and clad variation while stiffer Zircaloys may enhance the uniformity of the cladding.

[redacted] [redacted]

Metallic Fuel Development

Fuel Irradiations. Radiometallurgical examination is nearly complete on the production brazed irradiation test from the MTR, GEH-4-63 and 64. Extensive cracking was found in the uranium in tube 63 and in some cases the cracks extended from the center of the element out to the cladding. None of the cracks appeared to have started into the cladding. Cracking appeared to be most severe in the uranium in the braze-heat-affected zones, with some cracks extending into the braze material between the uranium and the Zircaloy end cap. No evidence of a shear failure mechanism has been found in this irradiation similar to that found in two earlier irradiations with the Fe-Be-Zr braze composition.

The variable braze thickness irradiation test, GEH-4-68, 69 and 70, has completed six cycles of irradiation at the MTR and is now awaiting shipment back to Hanford for radiometallurgical examination.

Irradiation of the prototype NPR tube-in-tube fuel element in the M3 Loop at the ETR will be completed early in June. The element will have received an average total exposure of approximately 1200 MWD/T and a maximum exposure of approximately 2000 MWD/T.

The experimental single-tube, dual enriched fuel element was examined after successfully completing one cycle of irradiation in the ETR. The element appeared to be in excellent condition with no visible indication of damage or corrosion. Dimensional measurements indicated that 100-mil warp had occurred as a result of the severe radial flux gradient which exists in the test facility. No appreciable changes in diameter had occurred as a result of the irradiation. This element is scheduled to be recharged into the reactor in July.

Examination of a coextruded Zircaloy-2 clad I&E^2 element which failed at 1700 MWD/T exposure in a KW Reactor production channel has continued. No evidence has been found to indicate the failure resulted from a shear-induced fracture of the clad at the fuel-cap juncture as a consequence of thermal incompatibility between the Zircaloy-2 cap and the uranium fuel. The cause of failure has not been determined. However, fuel cracking was observed, and it is possible that the failure was induced by core splitting. The highest exposure element from the same tube as the failure is being examined for evidence of incipient failure.

The end cap of the failed thermocouple probe discharged from a KER loop last month has been examined metallographically. The end cap was heavily hydrided and the braze layer in the stainless steel-to-Zircaloy-2 braze joint, used to attach the thermocouple to the probe, was cracked. Water entry probably occurred through the braze joint leading to corrosion and hydriding of the Zircaloy-2 and failure of the test element. Examination of the braze joint and of the test element will continue.

Fluted Fuel Development. Cylindrical fuel elements with fluted exterior surfaces have been proposed as a fuel element design in which the fuel is capable of undergoing large volume expansions without failure of the Zircaloy-2 cladding. Additional analyses, assuming the fuel behaves as a viscous fluid, have been made to estimate the mechanical conditions in the clad for various fuel geometries. The program has been extended to account for alterations in the stress distribution as the shape of the fuel element changes. For flutes formed by circular arcs, both an elastic criteria and a criteria of equal bending strains in the arcs established similar ratios of hill-to-valley arc radii. Calculations based on these criteria indicated that the approximate hill-to-valley radii ratios are 0.25 for elements with six flutes, 0.35 for eight flutes, and 0.4 for ten flutes. As the number of flutes increases, the stresses associated with one percent swelling increase. A single-tube element with eight flutes has been designed for irradiation testing. This design will provide for an 11 percent volume increase if the fluted surface is completely extended to the circular shape. The tangential arcs which form the flutes have a radii ratio of approximately 0.225. The nominal dimensions of the element are 2.327 inches OD and 1.286 inches ID. The Zircaloy-2 clad element will be fabricated by coextrusion using 0.95 percent enriched uranium as the fuel. Fabrication of the extrusion die has been started.

Cladding Deformation Studies. An analysis of the cladding strains and failures of Zircaloy-2 clad uranium fuel rods irradiated in NaK capsules to exposures of 2000 MWD/T has been made. Below an average Zircaloy-2 cladding temperature of 325-350 C, the maximum uniform strain observed without a failure by fracture or localized necking was 1.5 percent. Above this temperature the observed strain increases to approximately eight percent at 400 C. This transition temperature corresponds to the beginning of rapid recovery of mechanical properties in irradiated Zircaloy-2 indicating that irradiation damage is one of the factors limiting the uniform strain in the Zircaloy-2 cladding. The cladding thickness was non-uniform in these tests and cladding performance was probably also influenced by clad irregularities.

To obtain more information on the effect of cladding thickness variations on uniform strain limits of Zircaloy-2 cladding, a series of NaK capsule irradiations is being conducted. Preparations are being made to open four irradiated capsules in which defects were machined into the clad surface. Irradiation samples for a second series of cladding-study capsules have been cut and are now being machined.

Cladding Deformation Studies. A study of the effects of surface irregularities on the mechanical behavior of Zircaloy-2 cladding is being made. Test data have been obtained from various forms of notched tensile and tube burst specimens, as well as from fuel element cladding. The results indicate that hydrostatic pressure in a tubular specimen gives the most sensitive stress state for determining the effect of notches or thickness irregularities in Zircaloy-2 cladding. Hydrostatic testing has been nearly completed on a sufficient number of notched tubular burst specimens at room and elevated temperature to assess the effect of internal notches on uniform cladding strains.

Three capsule type experiments are being assembled for irradiation in the MTR to investigate the relative swelling performance of uniformly enriched and dual enriched uranium fuel rods operating at equivalent power generation and surface temperatures. Density of the Zircaloy-2 clad fuel rods has been measured, and a fine grid has been etched on the cladding surface to measure non-uniform straining of the cladding which may occur during irradiation. Assembly of the capsules prior to filling with NaK is about fifty percent complete.

Uranium Billet Heat Treatment. Thirty-five NOE, and thirty-three NIE 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.

A 19-inch long NON high Fe, Al ingot was single beta heat treated and quenched prior to primary extrusion. The average composition of the metal was 643 Al, 400 Fe. Visual examination disclosed no evidence of quench cracking due to the heat treatment.

Cerium-Modified Zr-2. A series of five zirconium buttons have been melted, containing 0.1, 0.5, 0.75, 1.0 and 2.0 wt/% cerium. The purpose of this work is to determine if there is any deoxidation or scavenging of the zirconium from the addition of cerium. Zirconium of 1335 ppm oxygen and 163 BHN was used. Hardness readings and chemical analysis will be used to determine the effects of the cerium additions, and high pressure steam corrosion tests will be made on rolled sheet samples.

N-Fuel Support Development. Tests are being developed to measure quality of Zircaloy-2 strip intended for use as fuel element supports. High fatigue resistance and good formability are desired. An air-powered fatigue tester that vibrates strip samples at natural frequencies was made and is being tested. Formability of the Zircaloy-2 strip is being measured by making increasingly severe bends in the strip and determining the radius of bend at which cracking of the strip first appears.

Shaker-table fatigue testing of supports on assembled fuel elements, dry and in a water environment, was continued. In these tests the natural resonant frequencies of the fuel system in and out of water was measured for the inner fuel tubes. It was found that water in the annulus between the inner and outer tube lowered the resonant frequency slightly but did not significantly affect the amplitude amplification factor at resonance.

K-Reactor Fuel Supports. Proposed self support for the KVNS fuel element is of the "suitcase handle" configuration. The "bridge" section of the support is a channel cross section to give added strength. Collapsibility specifications should be met by splaying the legs of the channel outward upon crushing of the support. Die design is complete and fabrication will begin immediately.

Supports - Dummies, KER Loops 3 and 4. "Suitcase handle" supports for the KER dummies were fabricated from 16 gage (0.061 inch) sheet steel containing 0.084 percent carbon, 0.01 to 1.0 percent Cu, Mn, and Ni, and traces of Ag, Cr, Mg, Pb, Si and Sn. Load deflection curves for this support show a spring rate of approximately 24 lb/0.001 inch for the first 0.010-inch deflection. Spring rate drops to about 20 lb/0.001 inch for the balance of the range to 0.070-inch deflection where the support collapses under an 1150-pound load. Some difficulty in controlling support height was encountered during welding but can be eliminated by modifying the electrode design. Eight dummies are now being charge-discharge and flow tested in a full loop mockup to determine the support wear properties.

Fission Gas Yields in Uranium. Values given in the literature for the volume of fission gases generated in irradiated uranium, expressed as cm^3 gas (STP) per cm^3 uranium per atom percent burnup, range from 3 to 4.95. An analysis of the fission gas yield for uranium irradiated in a thermal flux was completed. The yield of xenon and krypton in irradiated uranium is found to vary with neutron flux from 4.8 cm^3 (STP)/cm^3 uranium-atom percent burnup at 10^{12} neut/cm^2-sec to 5.9 cm^3 (STP)/cm^3 uranium-atom percent burnup at 10^{15} neut/cm^2-sec. The flux dependency is a result of neutron capture by the Xe-133 and Xe-135 isotopes. In the analysis consideration was made of the effects of fission gas contributions from plutonium fissioning and from the small percentage of fast fissioning of uranium and plutonium in a thermal reactor. A document, HW-73394, describing the calculations and giving the complete results, was issued.

Hot Headed Closure Studies. The objective of Phase I is to determine the conditions for projection welding a cap to a "hot-headed" fuel element and obtain a minimum void area between the cap and the fuel element.

In an effort to improve upon the projection welding of end caps to fuel elements a modification of the welding projection was made. A series of welds were made on end caps using a modified projection design. Evaluation of this series of weld tests is not complete.

The objective of Phase II is to investigate and develop methods to obtain a continuous bond between the cap and the end of the fuel element in an area circumscribed by projection welds.

Several series of welds were made on assemblies consisting of two Zircaloy caps which had the same size and shape as the fuel element end cap. One disc contained welding projections while the other cap had flat surfaces. Sn and Cu foil preforms were cut from shim stock and inserted between the end caps.

The principal objectives of these series of tests were to determine if the preplaced interface materials could be encapsulated without voids and without expulsion. The results indicated that the above objectives can be obtained. The use of interface materials in shim or washer form seems to be an undesirable method for preplacement. Slight misalignment of the shim results in contamination of the projection weld. More emphasis and effort will be applied to vacuum metallizing as a method for preplacement.

Self-Brazed Closure Process. About two dozen self-brazed closures were processed on the Sciaky spot welder in an effort to find the best processing conditions. The trial specimens included some with copper plated interfaces and Cu-Ni plated interfaces, which had been previously pressed by the closed-die 400 T press process but were poorly bonded; some with Cu plated bands on the sidewalls, previously pressed in the closed die but poorly bonded; some with Cu plated bands previously tried in the Sciaky machine, but poorly bonded; and some with Cu plated bands, in the unpressed condition. Two types of specimen holders were used: (a) one in which the specimen, fitted with a stainless steel interior plug, is tightly gripped in a multiple split disc collet, and the upper copper electrode is applied against the upper closure only, and (b) an arrangement whereby the specimen is clamped axially between the two electrodes, being partially or wholly supported by a close-fitting surrounding sleeve. In the first case, one closure only is processed; in the second, both closures are processed simultaneously. Obviously, the power, timing, and pressures must be different for each case.

The results of the experimental work may be summarized as follows:

(1) Pre-pressed specimens with plated U/Zr interfaces developed sound bonds when reprocessed in the Sciaky machine, both in case (a) and case (b).

(2) Unpressed specimens with unplated U/Zr interfaces did not produce sound bonds on single processing either in case (a) or case (b).

(3) Previously processed specimens with unplated U/Zr interfaces produced partially sound bonds on reprocessing and in one instance of case (a), a completely sound bond.

From these results it appears that the most propitious conditions include the use of caps with plated interfaces, cathodic etching of the uranium interface, case (a) type processing in the spot welder, and a welding program that increases the pressure during the welding cycle. One problem not yet overcome is that of preventing the stainless steel bore plugs from arcing to the bore walls during the welding cycle.

<u>Resistance Welded N Fuel Closure</u>. Attempts to produce a closure on N-inner fuel by resistance welding a Zircaloy-2 ring to the inside of the cladding walls have been encouraging in that the resistance welds have retained the uranium during the resistance brazing cycle. This closure is similar to the projection welded resistance brazed closure except that the uranium is recessed to a greater depth and

the closure cap is projection welded to the inside of the cladding walls through small Zircaloy-2 rings around the inside of the OD and ID cladding. A conventional fusion weld between the top of the cap and the cladding is made as a final closure.

The resistance weld on this closure was improved over initial work with KER inner fuel elements by changing the design of the welding rings to produce more pressure against the cladding walls and by machining projections on the caps to aid in fusing the cap to the intermediate weld rings.

Copper Brazed Closure. Copper has been found to form a good bonding agent between zirconium and uranium in the brazing process. Induction heating to 1050 C and holding for 30 seconds has been sufficient to form the eutectic and completely bond the cap. The addition of a small amount of tin to the copper appears to have beneficial results in the bonding obtained and the temperature necessary for brazing. Several attempts to form the primary closure first by electron beam welding and then brazing have proven unsuccessful to date. Investigations are continuing in this area.

Copper Welding. The increased penetration possible when welding copper with nitrogen as a shield gas may make this a desirable process for closing extrusion billets. Five welds were made using silicon copper billet can stock and silicon copper end plates. These were helium leak checked. No leaks were found. The simulated billet cans were then furnace heated to 600 C and quenched in water. A subsequent leak check disclosed that one had developed a leak.

Development of Drawing Lubricant. Cobalt containing Zr-2 wire for PRTR elements has been fabricated. The final batch of 182 wires, 8 to 10 feet long, were sent to Plutonium Metallurgy on May 21, 1962. Prior to finishing this job, it was demonstrated that one coat of the new drawing lubricant would last through five successive reductions and in so doing actually produce a better finish than if the wire were recoated after each draw. Drawing speeds up to 18 feet per minute were used successfully.

2. REACTOR PROGRAM

Corrosion and Coolant Systems Development

Electrical Resistance of ZrO_2 Films. The electrical resistance of ZrO_2 is being investigated as a measure of its protective qualities in preventing hydriding of zirconium. Difficulties have been reported previously in obtaining reproducible resistance values of the ZrO_2 film on Zircaloy-2 while in various gas atmospheres. A new assembly was fabricated with three graphite contacts on the Zircaloy surface to permit three simultaneous resistance measurements on the same sample. First measurements of the rate of resistance increase of the ZrO_2 film in oxidizing atmospheres and the rate of fall of resistance in vacuum have been much more reproducible than previous assemblies. After an oxidized sample is placed in a vacuum, a variable induction time is observed before the resistance begins to fall. However, after the resistance drop begins, the rate of fall is fairly consistent from point to point taking about three days to drop from 100,000 to 100 ohms at 450 C. An unexplained increase in the rate of resistance drop occurred after twenty hours for all three contacts.

Hydriding Corrosion Capsules. Zircaloy samples exposed to a simulated NPR gas atmosphere in the reactor for 167 days at temperatures of 290, 350, and 400 C have been retrieved and weighed. Samples irradiated in the "as etched" condition showed weight gains generally up to three times more than comparable samples exposed to the same conditions out of the reactor. Zircaloy-4 samples autoclaved (400 C, 1500 psi) for 56 days prior to capsule exposure showed signs of spalling at exposure temperatures of 350 and 400 C. "As etched" samples exposed at 400 C showed between two and four times more weight gain than samples exposed at 350 C, and samples exposed at 350 C generally showed two to three times more weight gain than samples exposed at 290 C. The protective effect of pre-autoclaving at 400 C was substantially reduced under irradiation; weight gains (in-reactor) of pre-autoclaved samples generally exceeded those of "as etched" samples. This is in contrast to out of reactor exposure where weight gains of pre-autoclaved samples were generally small at temperatures below 400 C.

Small punched samples are being prepared for hydrogen analysis.

Dynamic Corrosion Tests. Exposure to neutral pH water at 900 F, 5000 psi, flowing at 30 fps for 964 hours produced the following penetrations: 304 stainless steel, 0.030 mils; 316 stainless steel, 0.010 mils; Hastelloy-X, 0.006 mil; Zircaloy-2, 1.66 mils.

An 1800-hour exposure to pH 10 (LiOH) water at NPR primary system inlet water temperature produced a corrosion rate of 0.016 mil/year for 304 stainless steel and 0.01 mil penetration of Zircaloy-2. After 1200 hours, A212 C/S was corroding at a rate of 0.2 mil/year.

Decontamination Studies. Samples of Stellite 6, Haynes Alloy 25 and chrome plated steel were exposed to alkaline permanganate at 105 C for varying lengths of time. There was negligible corrosion of the Haynes Alloy 25. Penetration of the Stellite 6 samples varied from 0.2 to 0.4 mil. The weight loss of the chrome plated steel was attributed to undercutting and flaking at imperfections in the chrome plate.

A production test was completed at H Reactor in which Wyandotte 5061 was used at 6 oz/gal and 75 C to decontaminate reactor rear face piping. This treatment was successful in significantly reducing radiation levels for subsequent maintenance work involving this piping. A coupon holder, containing prefilmed samples of aluminum, Zircaloy-2 Inconel, and carbon and stainless steels, was in place in the rear of tube 3372 H during this decontamination. Subsequent to the decontamination, this sample holder was recharged into an out-of-reactor test facility for further exposure to water at 122 C.

Crevice Corrosion of Stainless Steel. A stainless steel patch is being developed by Plant Engineering Operation, IPD, to repair reactor crossheaders when welded fittings are removed. This patch must be welded in place with water flowing through the header, and consequently, a gasket is used under the patch as a water seal during welding. The gasket may cause crevice corrosion of the stainless steel at the patch-to-header contact points. A severe test of gasketed assemblies (in a 45 gm/l solution of Turco 4306-C at 60 C for 168 hours) revealed that severe crevice corrosion of stainless steel can occur at gasketed joints. A sample of an actual patch used in the test is being sectioned for metallographic examination. Testing is continuing to compare the crevice corrosion in non-gasketed joints vs gasketed joints.

Structural Materials Development

Brittle Fracture Studies. Experiments to determine the effect of local areas of high hydrogen concentration were continued. A section of annealed, Zircaloy-2, KER pressure tube with a 0.3-inch square area containing about 5000 ppm of hydrogen was pressurized to 90 percent of ultimate burst strength at 150 C. A projectile was fired into the tube wall at a spot 180 degrees from the hydrided area to determine the relative crack propagating characteristics of the base metal prior to introducing a crack in the hydrided area. A crack formed on the inside surface but did not propagate. The plan was to weld this area, repressurize and test the effect of the hydrided area; however, the impact of the projectile initiated a crack in the hydrided area that propagated about 50 percent of the specimen length. This unexpected behavior of the tube will require further investigation and experimentation.

Burst Testing of Irradiated Pressure Tubes. (Project CAH 922) Design of the facility for burst testing irradiated pressure tubes will be ready for detailed study and approval during June. Cost estimates of the present design and re-evaluation of construction and equipment costs have necessitated an increase from $228,000 to $289,000 for the total project.

Nonmetallic Materials

Graphite Burnout Monitoring. Burnout rates calculated from data on small monitoring samples charged in channel 1880 at KW Reactor on March 14, 1961, and removed on May 5, 1962, are tabulated below:

Distance from front of graphite stack, in.	Burnout rate, percent per 1000 operating days
20	1.42
24	1.80
28	1.52
32	1.57
36	1.76
135	5.51
139	2.70
143	1.82
147	1.80
151	1.37
253	3.06(?)
257	0.55
261	0.48
265	0.38
269	0.37

The samples were charged into the reactor at a time when the location of the maximum burnout was not established with certainty. Since the date of charging it has been found that under current operating conditions the maximum burnout for the Hanford reactors occurs at 80 to 100 inches from the front of the graphite stack.

The peak height and location cannot be determined from these results; therefore, a new set of samples has been charged to obtain burnout rates at locations from 36 to 135 inches in from the front of the graphite stack. During the first scheduled shutdown they will be removed and measured.

The average burnout rate in the sixth charge of burnout monitors at KE was 0.52 percent per 1000 operating days. This represents a <u>decrease</u> of more than a factor of three from rates obtained during the fifth charge on similarly exposed samples. During the fifth charge the reactor gas-flow rates were generally greater than 2000 cfm, whereas, during the sixth charge flows ranged from 300 to 800 cfm. A consideration of the relative rates of reaction of water, carbon dioxide, and oxygen, and the location and intensity of burnout peaks at the several reactors, suggests that the species causing oxidation at the peak burnout location is primarily oxygen (with a possible contribution from water) and that more oxygen is carried into the reactor when higher flow rates are employed.

<u>Graphite Burnout in the NPR</u>. A graphite burnout program based on equilibration of reactive gases with graphite (HW-71737) has been used to calculate the burnout profile for the NPR from the estimated temperature distribution and proposed gas-system specifications (0.04 percent oxygen and a -20 F inlet dew point). The profile indicates that $>$ 95 percent of the oxidation will occur in the first 9 feet of core graphite with a peak 4 feet into the core (13.5 ft in front of graphite center line). The calculated average burnout for this region is 3.7 percent per 1000 operating days for the gas composition given above and a flow rate of 400 cfm assuming the reactions:

$$C + O_2 \longrightarrow CO_2$$

$$C + H_2O \longrightarrow CO + H_2$$

The utility of the program has been demonstrated by calculating a burnout profile for D Reactor. The shape of the curve of burnout <u>vs.</u> distance along the channel agrees well with the experimental results. An estimate of the ratio of the peak-to-average burnout rate for small monitoring samples in D Reactor is approximately 9. If this is assumed to apply in the NPR, the peak rate would be approximately 33 percent per 1000 operating days. It should be remembered that these rates apply to the small burnout monitoring samples.

NPR Graphite Irradiations. Fabrication of the initial second-generation capsule, H-4-2, in the series of long-term irradiations of NPR graphite has been completed. The capsule contains 13 previously irradiated samples from the H-4-1 capsule and 11 new samples. It will be installed in the General Electric Test Reactor upon removal of the second capsule in the series, H-5-1, at the end of the current reactor cycle. No problems were encountered in the capsule construction even though about half of the samples were radioactive due to the prior irradiation. The maximum body dose rate during handling of the samples was only 30 mr/hr.

Verification of the data from the samples in the H-4-1 capsule is continuing. The two capsules presently in the GETR, H-5-1 and H-6-1, are operating satisfactorily with all thermocouples functioning properly.

Thermal Hydraulic Studies

Thermal Hydraulic Characteristics of I&E Fuel Elements in a K-Reactor Process Tube. Laboratory experiments were completed which allow prediction of the fuel temperatures during the shutdown transient following a rapid plugging or a complete failure of an inlet connector to a process tube at K Reactor. These experiments completed a program to upgrade such information for all of the present production reactors at Hanford.

The test section used was a full-scale mockup of a 38-piece charge of K-IV-N fuel elements in a K process tube. It consisted of a 1.447-inch OD by 0.385-inch ID electrically heated rod in a 28.5-foot long, 1.658-inch ID process tube. Metals of different electrical resistivities were used in fabricating the rod to produce cosine heat generation along the length of the test section.

In the experiments simulating rapid plugging of an inlet line, the experimental procedures consisted of initially establishing a rear header pressure of 60 psig and an equilibrium flow rate which resulted in an outlet temperature of 130 C at a given tube power. A remotely operated valve, located in the supply line, was then closed forcing the liquid flow through a by-pass line around the experimental test section. The valving in the by-pass line was adjusted prior to the transient such that a desired reduced flow rate was obtained through the test section upon closing the remotely operated valve at a front header pressure of 450 psig. Constant front and rear header pressures were maintained during the run.

Heater rod surface temperatures were then observed and a 1400 in-hour power decay was initiated when the temperatures either "leveled off" or continued to increase to a point which jeopardized the test section. The time lapse between plugging and initiation of the power decay varied from 7 seconds to 23 seconds depending upon the severity of the plug.

In the experiments simulating rupture of an inlet connector, the experimental procedures consisted of initially establishing an equilibrium flow rate which resulted in an outlet temperature of 130 C at a selected tube power and rear header pressure. A remote operated valve located in the supply line was then closed. Simultaneous with this a remote valve located near the front nozzle and front connector assembly was opened, venting the connector to atmosphere. Rear header pressure was maintained constant during the transient by means of a small centrifugal pump. Three seconds after the remote operated valves were actuated a 1400 in-hour power decay was initiated. Test section surface temperatures were observed during the decay period. This procedure was followed for runs at initial tube powers of 750, 1250, 1750, and 2250 KW. In addition, since the ability to cool the test section adequately in such experiments depends upon the resistance to flow back through the tube from the rear header, experiments were performed with and without a 0.425-inch venturi placed in the front nozzles.

Preliminary analysis of the data indicates that rear header pressures of approximately 56 psig with the venturi and 40 psig without the venturi afford adequate protection against fuel jacket melting for initial tube powers of 1250 KW and less.

Excessive temperatures were not reached during the 750 KW run when the rear header was held at 15 psig. This corresponds to a quality of about 1.5% at the rear header when the coolant temperature is 130 C.

Revised Front Nozzle for K Reactor. Flow tests were conducted to determine the pressure drop characteristics of a K Reactor front face nozzle designed for use with ribless zirconium tubes. In this nozzle a perforated plate insert was used to cover the nozzle inlet water flow passage to prevent lodging of the support feet of self-supported fuel in this opening. In addition, the 90° turn in the flow passage at the nozzle was eliminated and a 45° turn was incorporated into a new pigtail-nozzle adapter similar to the CG-558 fittings on the old reactors. The nozzle is described on drawing SK-23101.

The pressure drop characteristics of this nozzle assembly were compared with those of the present standard K front nozzle. It was found that there was no detectable difference in pressure drops between the front header and tube inlet for these two assemblies. Evidently, any increased losses associated with the perforated plate insert were compensated by the new pigtail-nozzle adapter.

Uniform Wall BDF Tube with O-III-N Fuel Elements. Pressure drop and flow split experiments were performed with a charge of 32 O-III-N fuel elements in a new BDF size process tube which has a uniform wall thickness (drawing H-1-24174). From the data, the ratio of the flow rate in the annulus channel to that in the hole channel was found to be 2.63 and 2.80 for hot and cold flow, respectively, at normal central zone flow rates. For fringe tube flow rates, the flow ratios drop to 2.60 and 2.74 for hot and cold flow, respectively. These ratios are about equal to those found previously using the old style tubes.

It was found that the pressure drop for O-III-N fuel in the uniform tube was slightly higher than for the same fuel in the old style tube. This pressure drop difference (8 psi at 48 gpm) would result in a decrease of about 0.35 gpm or 0.7 percent in operating flow rate of a central zone tube.

Critical Flow of Steam-Water Mixtures. The first series of experiments to investigate the high pressure critical discharge of steam-water mixtures were performed in the heat transfer laboratory. The apparatus consisted of a length of straight pipe, 0.508" ID, conveying the flashing fluid from a 1000-gallon pressurizer to atmosphere. The pressure within the pressurizer was maintained constant at about 1000 psig throughout the blowdown by injecting high pressure air into the pressurizer as the liquid level dropped. The critical pressure was defined as that pressure existing at the outlet of the pipe during the presence of the "choked flow" conditions and was regulated within limits by means of a valve placed in the pipe. The axial pressure profile along the pipe was defined by means of wall pressure taps, and the exact value of the critical pressure was determined by extrapolating this profile to the end of the pipe (see HW-68934 for details of the method). The flow rate was measured by means of a Gentile flow tube, and the temperature of the subcooled water in the tank was determined by thermocouples. All pressures and differential pressures were measured with transducers and all data were automatically recorded on the THO data logging system.

Critical pressures up to 580 psia were attained during experimentation with initial water temperatures as high as 575 F. Qualities, generally, were less than 15 percent. Mass velocities ranged between 2×10^5 to 7×10^5 lb/min/ft^2. Complete analysis of the data and comparison with theoretical predictions has not yet been performed.

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.

C. REACTOR DEVELOPMENT - 4000 PROGRAM

1. PLUTONIUM RECYCLE PROGRAM

Thermal Hydraulic Studies

An analysis was made of flow data obtained at the PRTR during six startups when the temperature of the primary coolant was increasing. The following observations were made:

(1) The decrease in density of the D_2O upon heating from 130 F to 460 F causes a decrease in the bulk and average tube readings of about eight percent. This is the same as calculated from data in February, 1962.

(2) The decrease in viscosity of the D_2O upon heating causes an increase of three percent in the volumetric flow rate. This corresponds to a value of two percent calculated from data taken during a startup in February, 1962.

(3) Based on latest data the indicated bulk flow reading should be multiplied by 1.18 rather than 1.16 as is now done. This accounts for a 20 percent scale correction (in the instrument) and a two percent difference between the bulk flow and the sum of the individual tubes. (It is believed that the summation of the tube flows is more accurate than the bulk flow reading.)

(4) There is a maximum of about 15 percent difference between flow rates in the various tubes.

Component Testing and Equipment Development

Mechanical Shim Rod. Design of the second generation shim rod for PRTR has been delayed pending vending information on position indicating equipment and available design drafting manpower.

The second generation shim rod as presently proposed has the drive motors and position indicating transmitters located above the top primary shield (in the upper access space) as opposed to within the shield for the existing shim rods. Advantages of this approach are (1) additional space is gained for component arrangement, and (2) the control rod assembly may be divided into two subassemblies (drive and position indicating components group, and lead screw and shim rod component group) which can be separated to facilitate maintenance and repair. Relocating the drive and position indicating components groups in the upper access space subjects them to high temperatures (400-500 F) and, hence, will necessitate cooling. Alternative methods of cooling are being studied.

Environment Control Test Facility. Off-site procured materials for the shim rod environment control test facility have been received. On-site fabrication of the water (moderator) and gas (top access space and top primary shield) housing units is in progress.

EDEL-1 Modification. The engineering flow diagram was completed for the EDEL-1 modifications described in last month's report. Work has been initiated on equipment arrangement drawings.

Shroud Tube Replacement Mockup. Construction of the shroud tube replacement mockup pit was started but is presently suspended as a result of the construction strike. The design of the calandria and shield mockup has been completed. Fabrication of the mockup is also being delayed by the strike.

Process Tube Gas Seals. A major source of helium leakage from the PRTR is thought to be the gas seals at the ends of the pressure tubes. Development work has been initiated on alternate gas seals for this application.

Seals which are to be tested for the inlet bellows gas seal include zirconium O-rings, stainless steel O-rings, and a zirconium "ferrule" type seal. The present seal is a solid copper O-ring. The zirconium O-ring is presently available for testing, and fabrication has started on components for the "ferrule" type seal. All necessary testing equipment is essentially completed.

For the outlet nozzle to top primary shield gas seal, two means of improving the seal are being developed. First, nozzle hold-down is being redesigned to provide greater gasket pressure (and possibly better nozzle alignment). Second, the use of a softer gasket material, perhaps a highly compressible metal gasket, is being explored.

Hazards Analysis

One revision of PRTR Process Specifications was approved during the month.

The components of the PRTR primary coolant system were checked for materials that might be susceptible to brittle failure at temperatures below the normal operating temperature. The only material in the system which might exhibit a nil-ductility temperature (NDT) above the ambient cell temperature is A-212 grade B which was used in the pressurizer. Since the NDT of A-212 grade B steel could be raised above 40 F only by strain-aging under stresses far greater than experienced in operation of the reactor, it was concluded that brittle fracture of components of the PRTR primary coolant system would not occur.

In addition to the prompt lifetime calculations for mixed crystal loadings in the PRTR, reported last month, values were obtained for the effective delayed neutron fractions and decay constants in seven groups, the fuel temperature coefficient, and an estimate of the coolant void effect. Although the fuel temperature coefficient will be more strongly negative than at present, the coolant coefficient appears to be weakly negative or zero with the present degree of coolant degradation ($\sim$ 5% H_2O). Additional calculations are being completed to better define the reactivity change on coolant loss in the mixed oxide loading.

Design Studies

A study of the feasibility of increasing PRTR power level was initiated. The objective of the study is to estimate the power level which can be achieved without major alteration of equipment and with a minor investment in cost and effort. Primary system temperature would be held constant and secondary steam pressure and temperature lowered in order to remove the additional heat.

Plutonium Recycle Critical Facility

Hazards Analysis. Analyses were completed to provide supplementary information on PRCF nuclear safety. Comparative calculations were completed which show the difference between excursions with a core loading of UO_2 and Pu-Al and a loading of Pu-Al only. Assuming the maximum reactivity insertion rate (0.275 mk/sec), results show the excursion for the all-plutonium loading to be only slightly more severe than for the combined loading. This is due to the fact that the reactor is still well below prompt critical when the trip point is reached for either case. Thus, in the case of the all-plutonium core, although the power rises more rapidly, the trip point is reached sooner, terminating the excursion at a power level about six percent higher than the mixed loading case.

Drafts of two Plutonium Recycle Critical Facility (PRCF) process specifications were issued for comment. Writing of the comment drafts of the PRCF process specifications is 60 percent completed.

PRTR Rupture Loop

Component Testing and Equipment Development. Eleven thermal cycles of the in-reactor test section, with fuel charge, have been completed at a maximum operating pressure of 1800 psig and with temperature cycles from 200 F to 575 F. Data on flow versus pressure drop, water leak rates, and helium leak rates were taken. Specific leakage rates will be reported as soon as the data are processed. Pressure drops appear acceptable although the data have not been accurately analyzed as yet.

EDEL-2 is being readied for thermal cycle tests on the rupture loop Grayloc connector to determine (a) seal leakage rates and (b) connector bolt stresses. Piping, electrical, and instrument checkout of the loop is in progress.

Fabrication of the rupture loop fuel discharge equipment is approximately 80% complete. An acceptable quotation was received on the hose assemblies, and delivery is expected in a month.

Design of special tools to perform the fuel discharge operation is complete except for three or four tools which will be tailored to specific requirements established during testing operations. Fabrication of the special tools is approximately 20% complete. Fabrication of the shielding cart (by construction forces) has been interrupted by the strike of construction crafts.

A draft issue of the discharge procedure (covering those items which are the responsibility of EDO) was issued for comment.

Hazards Analysis. The safeguards analysis for the PRTR Fuel Element Rupture Facility (HW-61236 SUP5) was published and transmitted to AEC. The analysis shows that the facility cannot cause a reactor excursion and that a reactor excursion would not cause an accident in the facility. The facility would have a negligible effect on a reactor maximum credible accident. The maximum credible accident in the rupture facility, rupture of the pressure tube, could damage the reactor but would not affect its nuclear safety nor increase its radiological hazards.

Plutonium Fuels Development

Plutonium-Bearing Fuel Elements for PRTR. Fifty plutonium fuel elements have been operating in the Plutonium Recycle Test Reactor during the month of May. The present charge is made up of 33 Pu-Al clusters, three Al-Pu instrumented with flux monitoring pins and wire, three Hx Al-Pu clusters (16.5 w/o Pu-240), seven Vi-Pac UO_2-PuO_2, and four swaged UO_2-PuO_2 clusters. All of the fuel elements are the Mark-I, 19-rod cluster geometry. The present program calls for adding eight UO_2-PuO_2 clusters in the PRTR each month for the remainder of 1962. Except for a few process tubes containing special test elements, the reactor will by that time contain a core loading of uniformly enriched UO_2-PuO_2 fuel elements. These will be fabricated by both cold swage compaction and vibratory compaction in about equal numbers.

The Al-Pu spike fuel element fabrication has been phased out with a total of 70 Mark I-H, 19-rod clusters now irradiated in the PRTR. Of this number, twelve have reached goal exposure of 80 megawatt days, 17 between 60 and 70 MWD, eight between 50 and 60 MWD, and the remaining something less than 50 MWD.

The concentrated fabrication effort in April resulted in the output of over 600 UO_2-PuO_2 fuel element rods. This is sufficient to make up ten Vi-Pac 19-rod clusters and twenty cold swaged 19-rod clusters. The final steps of etching, autoclaving, wire wrapping, and cluster assembly have been continuing through May. Eight Vi-Pac UO_2-PuO_2 clusters and two swaged UO_2-PuO_2 clusters were delivered to the PRTR at the first of the month, and six swaged UO_2-PuO_2 and two Vi-Pac UO_2-PuO_2 clusters will be ready for delivery on June 1.

Experimental Fuel Elements for PRTR Irradiation. A 19-rod MgO-PuO_2 fuel element was fabricated by cold swaging. Feed material consisted of coarse MgO (-6+325 mesh) and a mixture of fine (-325 mesh) MgO and PuO_2. The master blend of fines contained 29.7 w/o PuO_2. The final composition of the fuel rods was 2.1 w/o PuO_2. The rods were incrementally loaded, with 160 increments per rod. The fuel rods are currently being etched prior to autoclaving and final assembly.

Several fuel rods containing ZrO_2 are being swaged to determine the swaging characteristics of the material. Feed material is being prepared for the fabrication, by cold swaging, of a 19-rod cluster containing ZrO_2-PuO_2.

Three thin-wall stainless steel UO_2-PuO_2 19-rod clusters are being fabricated for irradiation testing in the PRTR. Work is currently in progress on an element clad in 0.008-inch thick 304-L seamless tubing. The other two elements will be clad in 0.010-inch thick AISI 406 type stainless steel tubing when it becomes available. These eight-foot long elements are being fabricated by vibratory compaction of UO_2-PuO_2 using the incremental loading technique. The fuel composition is UO_2 - 0.48 w/o PuO_2 using plutonium which contains 11.61 percent Pu-240. Since the operating pressures in the PRTR would collapse any unsupported areas of the cladding, the fuel material completely fills the fuel rods and provides the necessary cladding support. Preliminary autoclave testing of 88-inch long thin-wall stainless tubes containing UO_2 has shown that it is necessary to obtain a uniform density of at least 87% of theoretical density for the fuel to support 0.008-inch wall cladding at autoclave pressures of 1300 psi. Normal tamping procedures used with 0.030-inch wall Zircaloy tubing to compact the oxide near the top of the rods causes deformation of the thin wall tubing. Apparently coarse particles of oxide are indented into the tubing. For this reason it was necessary to explore other means of obtaining uniform fuel density without using the tamping rod. Powder variations and compactor operating procedures are being investigated in an effort to produce thin wall fuel rods which have adequate internal support. PuO_2-bearing fuel rods will be subjected to hydrostatic testing at room temperature before autoclaving in order that the autoclaves will not be unnecessarily jeopardized.

One fuel rod for the segregated plutonium fuel element was completed. Data for the rod are shown below. Unfortunately, a ruptured glove was encountered during the final closure step, and some atmospheric contamination of the closure zone probably occurred.

<u>Segregated Plutonium Fuel Rod</u>

Zr-2 (low nickel) cladding	0.565-inch OD, 0.030-inch wall
Al-7.5 w/o Pu-2.0 w/o Ni alloy tube	0.500-inch OD, 0.443-inch ID, 88-inch long
Pu-240 content	9.70%
UO_2 (natural)	0.433-inch OD
UO_2 density	84% of theo.
UO_2 weight	2047 gm
Pu (total) weight	13.5 gm

A second Al-Pu-Ni alloy tube was successfully extruded. Considerable difficulty has been encountered with the tooling for this extrusion work. A number of mandrels have been necked down or fractured. The length/diameter and reduction ratios are on the borderline of being workable but were used since, if successful, it would mean expediting the tube fabrication and keeping the equipment expenditures to a minimum.

A zirconium clad, plutonium-zirconium alloy core fuel plate was made by roll-cladding; the aluminum foil wrapping previously used around the core for contamination control was eliminated. It had been hypothesized that the aluminum foil reacted with the Pu-Zr alloy core to form an Al-Pu intermetallic which caused localized thickening of the core in the finished plates. With the elimination of the aluminum foil, no intermetallics were detected radiographically, and the fuel density per unit area appeared uniform with some thinning at the edges and end of the core. Some localized alpha contamination occurred on the plate after rolling and trimming, but the contamination could be readily removed. A one-inch wide section through the core of a clad plate was tested in an autoclave to determine the corrosion and undercutting of the core. After 72 hours at 350 C, the maximum depth of penetration into the core was one-fourth inch. The bond could not be broken in the area which was not penetrated. Two one-fourth inch thick plates of Pu - 15 w/o Zr alloy were cast to provide core material for additional roll-cladding experiments; the plates will be rolled to 0.025-inch thick sheets. No inhomogeneities were detected in radiographs of the plate castings. The density of the cast plate was 14.9 g/cc.

A summary of the roll-cladding process is as follows: plutonium is alloyed with 15 percent zirconium at 1350 C. The alloy is cast into one-fourth inch thick plates in cold graphite molds at 1250 C. The plates are cold-rolled to 0.025-inch sheets. Cores are sheared from the sheets and placed in a zirconium picture-frame assembly. The picture-frame assembly is, in turn, placed in a mild steel sandwich. The sandwich is welded shut and hot-rolled at 840 C to a 12:1 reduction (14-17 passes). After rolling the steel sandwich is stripped from the fuel plate. The fuel plate is decontaminated (if necessary) sheared to size, and formed into the desired fuel assembly.

<u>Phoenix Experiment</u>. The irradiation and reactivity measurements on the high-exposure aluminum-plutonium samples is continuing in the MTR and the ARMF. The current status of the irradiations and

reactivity measurements is as follows: the sample which initially contained plutonium with 6.25 percent Pu-240 (GEH-21-1) has received four cycles of irradiation and ARMF measurements have been made; the sample containing plutonium with 16.33 w/o Pu-240 (GEH-21-3) has completed its fourth cycle of irradiation and is now cooling prior to ARMF measurements; and the sample containing plutonium with 27.17 percent Pu-240 (GEH-21-19) is now being irradiated for its third cycle. No reactivity data have been received since recalibration of the ARMF. Approval has been obtained to irradiate the aluminum-plutonium samples for seven MTR cycles each.

An underwater work platform and underwater handling tools are being designed and fabricated to permit easier manipulation of the irradiated samples in the MTR canal.

An analysis of the aluminum-plutonium-boron ARMF poison standards has been started in the PCTR. PCTR reactivity measurements are being made on these samples to determine the boron content and homogeneity.

Irradiation Testing. Radiometallurgical examination of the 42-inch long swage compacted uniformly enriched UO_2-PuO_2 seven-rod cluster element (GEH-11-7) is continuing. Autoradiographs made by exposing the irradiated fuel rods to glass revealed areas of localized fission product concentration which resulted from segregation of the plutonium enrichment during fabrication. Each fuel rod was sampled for fission gas and the results are given below:

Rod No.	Fuel Material Type	Total Gas Collected (ml) at STP
Center	Outgassed*	50.1
16	Unoutgassed	40.4
17	Outgassed	42.9
12	Unoutgassed	39.9
19	Outgassed	51.6
15	Unoutgassed	58.7
20	Outgassed	44.9

*Vacuum outgassed at 1000 C for two hours.

This element was irradiated for 23 full power days in the ETR high temperature loop which is an estimated maximum exposure of 800 MWD/T. The total gas release data do not indicate any discernible difference between outgassed and unoutgassed fuel material for these exposures and operating conditions. There is, however, a difference in the amount of gas released from rods in the element which were closest to the center of the reactor and those which were furthest from the reactor center. Those fuel rods closest to the center of the reactor released less gas than those which were furthest from the center. It is estimated that the horizontal peak to average flux ratio in the loop facility is about 1.2 which would result in approximately a 20 percent difference in the specific power generation of the fuel rods from one side of the element to the other. Those rods closest to the reactor center will therefore operate at a higher temperature and hence, more in-reactor sintering and higher fuel densities occur. Apparently the higher density material retains the fission gasses better than the less fully sintered material which operated at lower temperatures. If the fuel temperatures were further increased, however, so that center core melting or central voids and columnar grains were formed, the gas release from the higher operating temperature core material would probably be greater than from the lower temperature material due to the sweeping action which occurs during recrystallization.

One of the high operating temperature fuel rods from this element was sectioned both transversely and longitudinally in areas of high and low plutonium concentration. The appearance of the UO_2-PuO_2 fuel material clearly showed areas of different operating temperatures characterized by different structures. Even in the areas which operated at the highest temperatures, however, there was no central void formation or columnar grain growth; only large equiaxed grains had formed. Autoradiographs were made from polished specimens which clearly illustrated where the plutonium enrichment is located. In the same fuel rod there were areas of unsintered fuel material indicating that in-reactor sintering is incomplete for swage compacted arc-fused material operating under PRTR conditions.

The 42-inch long UO_2-PuO_2 cosine enriched seven-rod cluster is being examined in Radiometallurgy. The PuO_2 enrichment was varied along the length of this element in an attempt to achieve longitudinal power flattening. It operated for 27.3 full power days in the ETR high temperature loop which gave a maximum exposure of about 1100 MWD/T. The appearance of the element is good following its irradiation; i.e., there are no loose bands or wires, no harmful corrosion effects are visible, and no warpage of the element has occurred. The color of the Zircaloy cladding in the fuel regions is not uniform, and there are darker spots located on the surface of the rods. The element will not be disassembled, and autoradiographs of individual fuel rods will be made. It will be attempted to relate the discoloration on the surface of the fuel rods with light or dark spots on the autoradiographs.

Four of the eight ZrO_2-PuO_2 fuel capsules have completed their irradiation in the ETR, and six of the eight MgO-PuO_2 capsules have successfully completed their irradiation period. Four of the MgO-PuO_2 capsules have been sampled for fission gas, and the data are being analyzed. One of the MgO-PuO_2 capsules (GEH-14-350) gave off zero gas even though there was no indication of a malfunction of the sampling apparatus and no obvious indication of a cladding failure in the sample. The condition of the capsule will be scrutinized more closely in an effort to explain this apparent anomaly. The samples are being sectioned transversely.

One UO_2-PuO_2 capsule (GEH-14-85) which contains high density UO_2 - 2.57 a/o PuO_2 pellets was recharged into the MTR for additional irradiation. The second capsule (GEH-14-86) containing high density UO_2-4.13 a/o PuO_2 is tentatively scheduled for insertion during the next cycle.

The four specimens with UO_2-0.154 a/o PuO_2 and UO_2 (1.00 a/o U-235) pellets have not been irradiated in the MTR VH-4 Facility. Reactor personnel state that as a result of their recent experiments the series of tests on the specimens (GEH-21-13 thru 16) are just not compatible with the existing facility. The major problem is that the operating personnel cannot control the coolant flow closely enough to permit discharge of the samples and yet not scram the reactor. It has been decided to modify the facility at our expense so that the tests may proceed.

The bellows assemblies for the ICARUS Experiment were received, and a full-scale flow model of the fuel assembly and transport vehicle was prepared. Flow tests and calibration runs are being initiated.

Uranium Fuels Development

PRTR Fuel Elements. Vibrationally compacted (771 MWD/T_U), hot swaged (681 MWD/T_U), and cold swaged (2000 MWD/T_U) UO_2, 19-rod cluster PRTR fuel elements were visually examined in the PRTR storage basin during a reactor outage. All reflected continued, expected good performance, and no indication of corrosion or mechanical damage. Following the examination, the hot swaged and vibrationally compacted elements were recharged into the reactor. The cold swaged element, replaced in reactor by a plutonium-containing element, will be destructively examined to provide the first data on PRTR UO_2 performance and to provide exposure data for physics analyses of the PRTR core.

High Energy Impaction. High energy impaction is a method for preparing ultra-high density fuel particles from UO_2 and other refractory nuclear materials. Under investigation are (1) impaction to ultra-high density and (2) impaction followed by heat treating.

The effects of impact pressure, in the range 100,000 - 300,000 psi, on compacted and sintered densities for the impaction-heat treating process were determined. Although the compacted UO_2 density varied from 8.4 to 10.6 g/cc over this pressure range, the density of each sample after sintering was greater than 10.8 g/cc. The high pressures used in this series of experiments were attained with a 12 percent cobalt - 88 percent tungsten carbide punch. Impactions were performed at successively greater pressures, until the punch failed at 390,000 psi.

Four-inch diameter, six-inch long, stainless steel cans containing ten pounds of micronized UO_2 powder were heated to 1100 C and compacted by high energy impact at 150,000 psi in the vertical Model 1220B Dynapak machine, using an energy release of 145,000 foot-pounds. Material with a density of > 99 percent TD was obtained after crushing it to minus-four mesh screen size and heat treating 12 hours at 1700 C. It was necessary to crush the compacted UO_2 before heat treating, because air trapped in the powder during impaction, which was sealed in by surface sintering of large pieces of UO_2 during heat treating, caused bloating of the UO_2. Compacted UO_2 which sintered to 10.87 g/cc when crushed to minus-four mesh screen size, sintered to 10.48 g/cc in approximately one-inch diameter pieces.

An air atmosphere furnace which will heat sixteen capsules, four inches in diameter by six inches long, is being installed. A 40-minute heating time will provide one capsule for compaction every two and one-half minutes (a rate which would be capable of providing approximately 2000 pounds of UO_2 per eight-hour day). A transfer mechanism to deliver the capsules from the furnace to the Dynapak machine is being designed.

A tungsten carbide punch and die liner are being installed for compacting UO_2 at pressures of 350,000 psi (compared to the presently usable pressures of 150,000 psi on tool steel dies). It will be used in attempts to develop the impaction process in which ultra-high densities are obtained by the initial impact, without subsequent heat treatment.

Remote Fabrication Studies. Simultaneous vibrational compaction of UO_2 into all three components of a preassembled Mark II-C nested tubular fuel element appears feasible.

Disassembly and inspection of a complete fuel element, after extensive vibration tests revealed essentially no damage to cladding ribs. Some scuff marks, 0.001-inch or less deep, were found on the inner surface of the large tube. In-reactor tests on 19-rod cluster elements have shown that similar cladding marks soon obtain an "in-reactor-autoclave film," and do not constitute a hazard.

A UO_2 dispenser-loader tool was built for loading fuel into the inner tube of a Mark II-C fuel element by remote operation. Functional tests are under way to develop remote compaction techniques. The inner tube will be used alone for all preliminary remote operation tests because its particular geometry makes it the most difficult component in which to obtain high fuel compaction efficiencies.

The remotely operated welding turntable and remote TV equipment are scheduled for shipment June 4 and 25, respectively.

Physics Test Elements. Six UO_2, 19-rod cluster PRTR fuel elements were assembled with cobalt-Zircaloy-2 alloy wire spacing members and charged into the PRTR. Each wire was numbered and marked at critical locations for future reference. A special pin, which will facilitate bundle disassembly, was used to fasten the bottom hanger fitting to the central rod.

Cladding Studies. An off-site vendor, using high voltage electron beam welding equipment, attached the spacing members to five sets of the smallest diameter cladding for nested tubular, PRTR Mark-II, fuel elements. A weld pass was made on each side of each member to assure complete penetration, a virtually crevice-free weld, a smooth weld contour, and correct positioning of the spacer, i.e., the member will be perpendicular to the tangent of the cladding at the point of contact. The depth of the heat-affected zone does not exceed one-half the cladding thickness. Distortion of the tubes over the eight-foot length was not appreciable.

Four types of electron beam welds were used to assemble a prototypic length of an internally-only cooled fuel element. The advantages of good fit-up of components are clearly demonstrated. Unexplained phenomena of the electron beam were apparent in microscopic examination of the weld cross sections. An appreciable widening of the weld area occurred after the narrow beam had entered the metal.

The capabilities of the welding equipment are revealed to a great extent by these test sections. The high voltage electron beam welding equipment will perform well under a wide variety of conditions.

Prototype PRTR Fuel Element. A simple, potentially useful, PRTR UO_2 fuel element comprises a single rod containing at least as much UO_2 as is now contained in a 19-rod cluster. A three-foot long section of such a fuel element was fabricated by vibrationally compacting high energy impact formed UO_2 into a 2.328-inch OD by 0.060-inch wall Zircaloy-2 tube to a bulk density of 89 percent of theoretical. TIG closures were made between the heavy wall tubing and massive end caps. Calculations indicate that much of the fuel will be molten during reactor operation. Further detailed calculations are being made to predict the mechanical stability and thermal hydraulic performance during operation.

Materials Development

Water Quality Studies. The relationship between measured pH and lithium concentrations in solutions of LiOD in D_2O was calculated and compared with analytical data obtained from samples of the PRTR primary coolant. In general, the calculated values are in good agreement with the analytical results. The relationship of solution conductivity to lithium concentration and pH has not been calculated; however, analytical results are in good agreement with values obtained at the Savannah River Project (SRP). SRP reports that they obtain pH values by measuring the pH with standard laboratory equipment and then adding 0.4 to this value.

Discussions of the procedure for fluoride analysis in D_2O solutions with SRP personnel revealed the following differences between their procedure and that used at PRTR:

(a) Blanks, reagents, and standards are prepared using high purity D_2O for dilution. At PRTR, H_2O is used.

(b) Fluoride ion is measured by determining the concentration of the reaction product of fluoride ion and the chelate formed by mixing lanthanum (or cerium) and alizarin complexone. At PRTR zirconium is used in place of lanthanum.

The present PRTR procedure has been checked by comparing standards prepared using H_2O and D_2O for dilution. These tests indicate that the measured fluoride concentration is about 0.9 parts per million higher with H_2O than with D_2O. A procedure based on that used at SRP has been written and forwarded to PRTR for use in place of the present procedure. Using this revised procedure it should be possible to lower the fluoride detection limit to about 0.1 parts per million.

<u>Capacitance Measurements on Vapor Blasted and Vacuum Annealed Zircaloy-2 Surfaces.</u> Polarization capacitance measurements are being used to compare the relative roughnesses of Zircaloy-2 surfaces prepared by different methods. The range of the available bridge capacitance limits the geometric area of the electrode to about 1-2 cm^2. Two Zircaloy-2 wires (70-mil diameter) were vapor blasted and etched, respectively, for use as electrodes. The ratio of capacitances (C vapor blasted/C etched) was 5.48. Weight gains during autoclaving of vapor blasted and etched Zircaloy-2 specimens were also compared, using platinum crucibles to preclude spalling losses from the vapor blasted specimens. In the early stages of the weight gain versus time curves (~ 3-5 days) the ratio Δ W vapor blasted/Δ W etched was 4.6 which correlates relatively well with capacitances measurements as an indication of the relative surface roughnesses of etched and vapor blasted surfaces.

<u>Corrosion of Carbon and Stainless Steel in 300 Area Boiler Water.</u> Corrosion coupons exposed for 220 days in the primary heat exchanger of PRTR were removed for examination during May. The coupons were weighed, cleaned, weighed, examined, and some metallurgical sections made. The type 304 stainless steel coupons showed no localized corrosion and only minute weight losses (7 to 10 mg/dm^2). The type A212B carbon steel coupons lost 6 to 7 gr/dm^2 and displayed pitted surfaces especially in the stressed region of the U-bend. Evaluation and corrosion product analyses are currently in progress.

<u>Corrosion of 17-7 pH Spring.</u> A spring made of 17-7 pH stainless steel which had been exposed to a hot moist atmosphere for about one year displayed several cracks. This atmosphere might have been contaminated with chlorides at one time. Metallurgical sections of

this spring revealed the cracks to be similar to stress corrosion cracks which have been reported in other similar alloys.

PRTR Process Tube Monitoring. Sixteen PRTR process tubes were inspected during the May PRTR outage. In general, visual examinations disclosed new wear corrosion marks on most of the tubes examined. The depth of the wear marks ranged mostly from 1 to 7 mils. However, the pressure tube in channel 1946 had a 12-mil deep wear corrosion mark caused by the lower end bracket of a UO_2 fuel element. Also, a mark 12 mils deep by $1\frac{1}{2}$ inches long of unknown origin was observed 18 inches above the location of the lower fuel element end bracket in the process tube in channel 1643. This tube was removed for destructive testing and evaluation. The measured gas gap of channels 1857 and 1348 were essentially unchanged from the previous inspection of two months ago, being about 85 and 40 mils, respectively. Inside diameter and gap measurements for the remaining tubes are still being analyzed. A document describing the results of the PRTR tube monitoring program through April of 1962 has been issued.

Post-Irradiation Evaluation of PRTR Pressure Tubes. Studies have continued on PRTR pressure tubes removed from the reactor. In the room temperature burst test of an annealed section exposed to 5×10^{19} nvt ($E > 1$ Mev), failure was initiated by a wear corrosion mark at about 85 percent of the predicted ultimate tube strength. The crack did not propagate after penetrating through the wall indicating that the metal was still behaving in a ductile manner. Metallographic examination indicated a layer of massive hydrides about two mils deep located directly below the corrosion mark which measured 1/2 inch long, 1/8 inch wide, and a maximum of five mils deep. Many small cracks traversed the hydride layer throughout its length.

Another burst test specimen with an exposure of 10^7 nvt ($E > 1$ Mev) with two wear corrosion marks failed at about the predicted ultimate tube strength. The fracture did not propagate the entire specimen length nor was the failure through the corrosion marks. Both of the marks had a maximum depth of five mils and had the appearance of a contoured relief map with several plateaus. Metallographic examination of these marks showed a massive layer of hydride beneath the one, and only a slight increase of hydrides beneath the other mark. A document (HW-73698) describing the post-irradiation evaluation of Zircaloy-2 PRTR pressure tubes through April 1962 has been issued.

Mark III Tube Monitoring Equipment Development. The eyepiece section of the Mark III tube monitoring Ominiscope has been corrected to give the required image size and returned to Hanford.

A body for the Mark III gas gap instrument has been successfully machined from phenolic. The two previous bodies of silicone resin with Fiberglas and mica fillers split transversely and longitudinally, respectively, during machining. Tests to determine the effect of water absorption and radiation on the phenolic are in progress. The eddy current coils for the instrument are presently being wound with final assembly scheduled for early June.

2. PLUTONIUM UTILIZATION STUDIES

Plutonium Dioxide

The dilatometer used in obtaining bulk expansion data on PuO_2 to 900 C has been calibrated using a high purity copper standard of length such that the total expansion should be identical with the total expansion of the unknown. After these data are analyzed, the PuO_2 expansion will be corrected and a least squares fit to the data will allow an equation to be written for the thermal expansion of PuO_2.

Additional points on the lattice expansion of PuO_2 were obtained on the high temperature diffractometer. At 1126 and 1275 C, $\Delta a/a_o$ values of 11.472 and 12.936 were obtained. After the high temperature run the recirculating coolant water was found to be heating considerably and the experiment was halted. Installation of a heat exchanger to the recirculating pump is nearing completion.

Another anneal of FCC $PuO_{1.62}$ was made in an effort to transform this structure into the BCC phase. The sample was heated to 1300 C for one-half hour in vacuum and slowly cooled. The O/Pu after this treatment was 1.61 and the phase relations were similar to previous experiments. A strong PuO_{2-x} pattern with a_o = 5.409A was found in addition to a moderate amount of BCC alpha-Pu_2O_3. Apparently the high temperature FCC allotrope of Pu_2O_3 cannot be wholly transformed to the low temperature BCC phase, but rather undergoes a partial decomposition reaction. High temperature x-ray diffraction would be most helpful in studying this reaction. In addition, a small amount of a third phase was detected which gave a cubic structure similar to PuO, though the lattice constant of 4.964A is higher than anticipated.

A set of PuO_{2-x} specimens has been made by sintering at various temperatures in dry hydrogen. These samples will be used for resistivity and thermal emf measurements which are planned for this summer.

Plutonium Carbide

The change in lattice parameter of PuC samples stored at room temperature with time which was reported last month has been noted in additional samples. Powder samples stored under vacuum and stored in 0.5 mm diameter capillaries have both shown this change. The following table shows the change in cell size of a set of three samples, two of which were located on the previously reported lattice constant horizontals in arc-melted material. Samples were run in both a back reflection integrating camera and a 114 mm Debye camera. Lattice constant data from the Debye patterns were obtained by extrapolation against a Nelson-Riley plot.

Sample No.	a/o C	August 15, 1961 Back Reflection	May 21, 1962 Back Reflection	Debye
E-141-1	29.25	4.9659 ± 0.0002A	4.9724 ± 0.0006A	4.0724 ± 0.0005A
E-141-2	33.02	4.968_2 ± 0.001	4.9734 ± 0.0007	4.9742 ± 0.0004
E-141-3	35.95	4.9684 ± 0.0006	4.9754 ± 0.0003	4.9756 ± 0.0007

These samples all showed extremely weak lines due to a second phase and were quite close to reflections expected from alpha-plutonium.

Another specimen containing 48.24 a/o C has exhibited lattice constant changes in both the arc-melted and annealed (400-550 C for 91 hours) condition.

Sample	Date	Lattice Constant
L-2-9 arc melted	12/11/61	4.9709 ± 0.003A
L-2-9 arc melted	4/23/62	4.9728 ± 0.0004
L-2-9A annealed	1/15/62	4.9747 ± 0.0002
L-2-9A annealed	4/18/62	4.9771 ± 0.0006
L-2-9A annealed	5/23/62	4.9780 ± 0.0008

Some scatter is apparent on these lattice data; nevertheless, the trend appears toward increased lattice parameters. These samples showed PuC plus a small amount of Pu_2C_3. There is no obvious explanation for these observed changes. If a precipitation from the defect PuC phase were occurring, one might anticipate a decreased cell size rather than an increase. Another possibility which comes to mind is self-induced radiation damage though at present this is pure speculation.

Some thirty coatings on graphite were tested for their ability to resist attack by PuC at 1550 C. Of these the most successful were: a commercial BeO coating which includes a binder and is applied in two coats with an eight-hour bake; a Plasmatron-sprayed alumina coating; and a Plasmatron-sprayed five-mil TaC coating. The last of these showed a good resistance to attack. It was tightly enough bonded to the graphite and was thin enough so that it successfully withstood three sintering cycles (one hour at 1550 C) without any signs of attack or spalling. The oxide coatings were satisfactory in resisting attack but tended to spall after one cycle.

Plutonium Nitride

Several rods of alpha plutonium weighing 10-20 grams each were arc melted under one atmosphere of gettered nitrogen. Examination of the products by x-ray diffraction showed plutonium nitride, PuN, plus residual alpha plutonium. Photomicrographs of these arc-melted buttons show a PuN dendritic structure in a plutonium matrix. To date, a single-phase structure has not been formed by this method. The most propitious means of PuN synthesis appears to be to react plutonium hydride with either anhydrous ammonia or nitrogen. Plutonium hydride can be obtained easily by heating massive plutonium to 150 C in high purity flowing hydrogen; the plutonium rapidly disintegrates into small particles of hydride. On completion of the hydriding process, nitrogen gas is substituted for hydrogen, and the temperature is raised to 600 C. An alpha plutonium rod or button is thusly converted in situ to plutonium nitride. This same technique may possibly be used to convert the plutonium matrix in the arc-melted products to the nitride. Attempts to accomplish this matrix conversion will be made during the next reporting period.

3. CERAMIC (URANIUM) FUELS RESEARCH

Specific Volume of Molten UO_2

The specific volume of UO_2 at temperatures between 3000 K and 3200 K was measured. The change in volume during melting was 7.2 percent. At the melting point (3063 K), the densities of the solid and the liquid were 9.88 g/cc and 9.33 g/cc, respectively. The density of the liquid at 3200 K was 9.03 g/cc.

Measurements were made by radiographing sealed tungsten capsules partially filled with UO_2 and self-resistance heated to temperature. Each capsule contained slightly more than one gram of UO_2 in a cavity one-eighth inch in diameter by one inch long. Radiographs were made at 3000 K, 3063 K, 3100 K, and 3200 K during both heating and cooling.

The radiographs required a two-minute exposure to a 100 curie Co-60 radiation source. The dimensions of the mass of UO_2 were precisely determined from each radiograph, using a measuring microscope. From these dimensions and the accurately known weight of UO_2 in the capsule, the specific volume was calculated.

Each capsule was designed so that the UO_2 solidified directionally from bottom to top. This prevented entrapment of shrinkage pores in the body of the oxide and caused the shrinkage to occur as a single, easily measurable pipe at the top of the UO_2 column.

Single Crystal UO_2 Irradiations

The UO_2 single crystal thermal conductivity specimen studied in the joint HAPO-BMI program was irradiated in the Hanford "snout" facility to an approximate exposure of 10^{15} nvt. Flux monitoring materials included in the capsule will be used to determine exact exposure. Thermocouple and centering guide wires imbedded in the crystal during previous measurements at BMI were not removed for fear of damaging the crystal; they will be replaced, using the ultrasonic drill at BMI, before further thermal measurements are made. The crystal was returned to BMI, and the next series of measurements is expected to be completed within two weeks.

A second precharacterized single crystal of UO_2 was irradiated in the Hanford "snout" facility to an exposure of approximately 10^{16} nvt. This irradiation test is the first in a series of short time irradiation experiments. The low exposure irradiations permit post-irradiation studies of radiation damage to be conducted with minimum shielding in the laboratory.

UO_2 Relocation Experiment

A fuel capsule containing tungsten marker wires in sintered UO_2 pellets was irradiated in the MTR GEH-4 facility. Two of the pellets contained 1/4-inch long, 0.014-inch diameter tungsten wires in a spiral array. During irradiation, relocation of fuel was expected to change the relative positions of the tungsten wires. Comparison

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of the locations of the wires before and after irradiation will provide quantitative relocation data. The test was terminated when high gamma activity was detected in the coolant shortly after the test element reached full power. The element is being returned to Hanford for destructive radiometallurgical examination.

Electron Microscopy

Comparison of mosaics of reflection electron micrographs of an irradiated UO_2 crystal revealed that recognizable changes occurred during heating to 980 C. Most prominent was the development of a broad, fluted micro-crack. The shape and structure of the line suggest it might have resulted from an accumulation of defects in a sub-grain boundary during heating. Experimental conditions did not allow separate identification of effects due to irradiation plus heating, and to heating alone. Further work is being planned with more detailed examinations before and after each treatment.

Sequential Ceramographic Examination of Irradiated UO_2

A series of photographs were made of the cross-sectioned surface of a UO_2 fuel rod to show the changes in the appearance of the surface as it is ground away, while moving through the fuel from a low heat flux zone to a high heat flux zone. Photographs were taken at approximately 0.002-inch intervals through a one-half inch long section of a rod from the "Princess" (heterogeneously enriched) test element. The photographs were made into a motion picture which gives the illusion of a three-dimensional view of an irradiated UO_2 fuel element.

Irradiation Effects

A paper, "Irradiation Alteration of Uranium Dioxide," (HW-73072), was written and presented at the IAEA Symposium on Radiation Damage in Solids and Reactor Materials, May 7-11, 1962. The paper reviews irradiation behavior studies by members of Reactor and Fuels Research and Development Operation.

4. BASIC SWELLING PROGRAM

Irradiation Program

Two general swelling capsules were discharged after attaining their goal exposure. These capsules are cooling in the reactor discharge basin prior to shipment to Radiometallurgy for disassembly. Two additional capsules are still under irradiation in tandem in a single test tube. These capsules are being operated at constant temperatures of 575 C and 625 C, respectively, regardless of reactor operating conditions. One additional capsule is complete and has been shipped to the reactor for charging. Another capsule is approximately 75 percent complete. Assembly of additional general swelling capsules will be delayed until heaters which are on order are received. In the interim, construction of metallographic swelling capsules will proceed. The assembly of a prototype metallographic swelling capsule is now in progress.

Post-Irradiation Examination

Capsules 7 and 8, each containing three pairs of half, hollow, uranium cylinders, have been opened and the inner spacer containing the specimens retrieved. Capsules 7 and 8 were irradiated to a burnup level in the specimens of about 0.4 a/o at respective control temperatures of 525 C and 575 C. The specimens are in very poor condition. It is extremely difficult to remove them from the holder. Each sample was originally held loosely in position with 0.003-inch thick, perforated tantalum foil. After irradiation the samples are severely warped and bulging against the foil. In many places, portions of the specimens protrude through the perforations in the foil by as much as one-sixteenth of an inch. The specimens in Capsule 7 (525 C control temperature) are much worse in this respect than are the specimens in Capsule 8 (575 C control temperature). If these effects are due to the nucleation and growth of fission gas pores in the normal sense, the relative magnitude of the damage to the specimens appears to be reversed, i.e., the specimens in Capsule 7 should have been less severely affected. The fact that they were not suggests that something other than swelling has occurred. A temperature excursion that occurred late in the irradiation due to an operator error may have had considerable influence on the specimens. The specimens in both capsules momentarily were heated into the beta phase when a burnup of about 0.3 a/o had been achieved. The effect of a high temperature pulse (probably less than two minutes) on the behavior of uranium irradiated to high burnup levels is not known.

One of the pairs of specimens in Capsule 8 was removed from the specimen holder and the tantalum removed from the specimens. One of the half, hollow cylinders was mounted in boat resin, mechanically polished and examined metallographically. The sample appeared to be metallic but extremely porous allowing large portions to be impregnated with boat resin. This is not considered to be typical behavior of uranium irradiated to this burnup at this temperature. Additional study is required to understand the damage accumulated by this specimen. Efforts are being made to develop the microstructure of the sample with the use of chemical etchants. Microhardness and x-ray diffraction measurements are also planned. The other half of the specimen pair is being mounted in a mechanical mount to attempt polishing, cathodic vacuum etching and replication. Attempts will be made to metallographically examine the other specimens that are still on the holders.

Duplicate samples given the same thermal treatment in NaK filled laboratory capsules as the in-reactor Capsules 7 and 8 show no distortion. The specimen surfaces exhibited only minor staining where contact was made with the tantalum foil wrapped around the samples.

Zircaloy-2 clad coaxial U-U diffusion couples irradiated to two burnup levels have been annealed for 100 hours at 650 C and also at 700 C. Prior to the annealing treatment, the irradiated specimens showed no evidence of cracking in the uranium, or cracking at the bond between the clad and the uranium. The outer layer of enriched uranium had sustained a burnup of 0.2 a/o and 0.4 a/o during the two exposures; burnups in the depleted central core are lower by a factor of 20.

After a 650 C anneal, some cracking in the uranium (both enriched and depleted) had occurred, but the bond between clad and uranium was sound. In the 0.4 a/o burnup uranium shell immediately adjacent to the clad, a band of small, equiaxed grains with relatively large pores at their boundaries developed. Within the enriched layer only fine pores smaller than 0.5μ in diameter had developed. The core of depleted uranium shows only very small pores, $\sim$ 400 A in diameter, whereas in the specimen with lower burnup, no pores whatsoever were observable in the core containing the depleted uranium. No evidence of pores in the intermetallic zone between the clad and uranium was found.

In the case of similar specimens annealed 100 hours at 700 C, marked changes have occurred. The intermetallic zone has widened and a circumferential crack between this zone and the enriched uranium layer developed. The enriched uranium from optical microscope examination has developed a wide band due possibly to diffusion of clad metal into the uranium. Cracking in the enriched uranium shell and depleted uranium core has also occurred. Extensive metallography, both by optical and electron microscopy, will be conducted. Pore formation at boundaries of depleted uranium grains adjacent to the enriched uranium may have occurred during the 650 C anneal in high burnup specimens, as some evidence of this has been found. Additional evidence, however, is needed.

Restrained Irradiations

In order to gain insight into the influence of Zr-2 cladding on the swelling of uranium, Zr-2 clad uranium rods with selected uranium temperatures, cladding thicknesses, and exposure are being irradiated in NaK filled temperature monitored capsules.

Density and burnup were measured on five fuel rods removed from capsules irradiated in the MTR. A sixth fuel rod from these capsules was so severely oxidized that a sample could not be obtained. The results from the fuel rods are tabulated below:

Sample	Density Change g/cm^3	Burnup a/o	Volume Avg. Uranium Temp., C
94-2	- 1.91	0.48	567
95-1	- 4.96	0.33	482
95-2	- 5.01	0.30	462
99-1	- 1.01	0.49	430
99-2	- 6.95	0.42	480

The large density decreases of samples 95-1, 95-2, and 99-2 are attributed to badly cracked and oxidized samples. The swelling of samples 94-2 and 99-1 agree very well with previous swelling data reported from these capsule irradiations.

5. IRRADIATION DAMAGE TO REACTOR METALS

Alloy Selection

Two square feet of 0.062-inch sheet of experimental precipitation hardening Alloy R-27 has been obtained. This material is a nickel, iron, chromium alloy of the following composition: C, 0.076; Mn, 0.24; Si, 0.03; Cr, 15.18; Ni, 44.78; Fe, 32.54; Ti, 2.07; Al, 0.35; Cb, 4.56; Ta, 0.15; B, 0.011; P, 0.008; S, 0.004. The exceptional strength properties of this alloy make it suitable for many reactor applications. Its 0.2 percent yield strength in the double aged condition is about 210,000 psi.

A three-inch strip of the R-27 alloy is being cold rolled to 0.005 inch. This will be used to develop thinning techniques for electron microscopy transmission examination. Additional material is being prepared for Erichsen Cup tests to obtain data on the alloy's formability. Tensile specimens will be prepared and irradiated to obtain data which will indicate the effect of irradiation on its mechanical properties.

INOR-8 is a nickel base alloy which was developed by ORNL as a container vessel for molten fluoride salts to be used in the ORNL molten salt reactor. This alloy is presently being produced by the Haynes-Stellite Company under the trade name "Hastelloy N" and has the following nominal composition: Ni, 67.0; Mo, 16.5; Cr, 7.0; Fe, 5.00; Si, 1.0; Mn, 0.8; C, 0.06. Available data indicate that this alloy has excellent oxidation resistance to air as well as excellent strength and stress to rupture properties at temperatures to 1800 F. These properties together with good fabrication properties indicate the possibility of using the alloy in high temperature gas cooled reactor applications. Samples of this alloy are being obtained for cursory corrosion, oxidation, and irradiation tests.

Data Collection and Exchange

The first meeting of the Data Collection and Exchange Committee, whose members are T. T. Claudson (HLO - chairman); D. S. Billington (ORNL); T. A. Trozera (GA); and F. R. Shober (BMI), was held at Hanford Laboratories this past month. The results of this meeting included recommendations that first a quarterly progress report made up of contributions from each participating site be compiled and edited at HAPO and, second, that a semi-annual meeting of all participating sites be held to exchange data and information. The first such semi-annual meeting is scheduled to be held at HAPO on October 16, 17, and 18 of this year. Subsequent meetings will be held at other participating sites.

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.

During the month, 54 Zircaloy-2 tensile specimens irradiated in the G-7 ETR hot water loop were tested. Of these specimens, 39 were tested at room temperature and 15 were tested at 300 C. The specimens contained 0, 20, and 40 percent cold work prior to irradiation and represented both the longitudinal and transverse directions with respect to rolling. In the unirradiated condition, Zircaloy-2 exhibited marked directional effects in tensile properties both at room temperature and at 300 C. At room temperature the transverse specimens had a higher yield strength and lower uniform elongation in the annealed condition, but lower yield strength and about the same uniform elongation in several cold worked conditions. Furthermore, the tensile fractures of the transverse specimens became shear-like in appearance as the cold work level increased. Raising the testing temperature to 300 C reduced the tensile strength for the 0, 20, and 40 percent cold worked specimens to about 50, 57, and 60 percent, respectively, of their room temperature values. However, both uniform and total elongation underwent little change, and in some cases decreased at the higher temperature. Shear fractures for cold worked specimens representing the transverse direction were more perfectly developed at 300 C than at room temperature. Many of the observed directional effects both at room temperature and at 300 C are due to differences in the mode of deformation resulting from the crystallographic texture induced by cold work. Whereas thinning is prevalent in transverse specimens in which slip is restricted by texture, thinning is virtually absent in longitudinal specimens in which the texture favors slip.

Neutron irradiation at 280 C to about 3×10^{19} nvt (fast) had important effects on both the strength and plastic strain of Zircaloy-2. In general, many directional features observed in the stress-strain curves for the unirradiated specimens were also observed in curves for the irradiated specimens. The transverse specimens showed lower uniform strain than the longitudinal specimens and exhibited well-developed shear fractures for the cold worked conditions both at

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room temperature and at 300 C. However, in contrast to the unirradiated state, the yield strength for the transverse direction was higher for all cold work levels at both testing temperatures. The effect of temperature in reducing the tensile strength of the irradiated specimens was about the same as for the unirradiated specimens; however, the effect of temperature on plastic strain was inconsistent. For the transverse direction, increased testing temperature (to 300 C) increased the total strain of the cold worked specimens, but reduced the total strain of the annealed specimens. For the longitudinal direction, increased testing temperature increased the total strain of the annealed specimens, but had little effect on the total strain of the cold worked specimens.

At 3×10^{19} nvt exposure, marked reductions in uniform strain occurred for the annealed specimens both at room temperature and 300 C. These reductions for the longitudinal direction were from about 13 to 4 percent at room temperature and from 16 to 2 percent at 300 C. Corresponding reductions for the transverse direction were from 9 to 0.7 percent at room temperature and from 11 to less than 0.2 percent at 300 C. Little change due to irradiation occurred for cold worked specimens tested at room temperature. However, a marked change occurred for transverse, cold worked specimens tested at 300 C. These specimens displayed a sharp yield point, beyond which plastic strain occurred with decreasing load to fracture. Hence, the advent of plastic strain also marked the start of plastic instability. It is of importance to note that of the various conditions tested to date, transverse specimens, both annealed and cold worked, irradiated near 300 C and tested at 300 C have exhibited the greatest instability in tension with respect to uniaxial stress.

Notch tensile investigations of both annealed and cold worked Zircaloy-2 are continuing. Data representing cold work levels to 40 percent reveal that Zircaloy-2 is notch-insensitive in the unirradiated state and is not prone to brittle fracture at room temperature or above. Attempts to measure the stress intensity for both the plane strain (K_{IC}) and plane stress (K_C) criteria have been unsuccessful to date due to extensive plastic strain that precedes crack propagation. Several methods, including specimen and notch geometry optimization, are being tried to correct for plastic strain in calculating K_C. Although the proportional elastic limit is reached before crack "pop-in" in the unirradiated state, it is expected that this order will be reversed after a critical low neutron exposure. Beyond this exposure, the relationship between stress intensity and yield strength for irradiated Zircaloy-2 will be determined.

In-Reactor Measurement of Mechanical Properties

Two creep capsules have been charged in the reactor during the month. Each contain a 20 percent cold worked Zircaloy-2 specimen. A creep test is in progress in one capsule at 30,000 psi stress and at a temperature of 250 C (482 F). The other capsule is being used to conduct an in-reactor creep test at 350 C and 30,000 psi stress on a 20 percent cold worked specimen.

The 250 C - 30,000 psi in-reactor test has been running nearly 400 hours. A parallel ex-reactor test in an instron creep machine has accumulated 144 hours. After 144 hours, the in-reactor creep strain is greater than the ex-reactor creep strain. The creep rates in-reactor, however, are lower at 144 hours than ex-reactor. These are about 5.0×10^{-7} in/in/hr and 1.4×10^{-6} in/in/hr, respectively. This behavior is consistent with the test conducted at 310 C and 30,000 psi in that in-reactor rates were lower than ex-reactor rates, but the total in-reactor creep strain is greater than the ex-reactor creep strain.

Precise measurements of creep rates during reactor outages have not been obtained due to the temperature instability of the micro-positioner probe zero point during this period. Temperature fluctuations are produced in the probe by changes in the internal water pressure of the cooling coils. These pressure fluctuations, not present during reactor operating conditions, markedly affect the amount of heat transmitted to the micropositioner over short periods of time. This difficulty is not anticipated in the next shutdown as the water can be completely removed from the internal cooling coils without producing a temperature in excess of 250 C from gamma heating.

Damage Mechanisms

Three pounds of high purity (99.996 percent) iron have been received. This material is in the form of bars with a one-fourth inch diameter and 10 to 15 inches long. The impurities consist of 15 ppm metallics and 25 ppm nonmetallics. A portion of the iron has been cold swaged and the cross-sectional area reduced by 37.6 percent and 45.6 percent in four and five passes, respectively. The nominal 0.250 inch diameter bar was reduced to diameters of 0.234 inch, 0.218 inch, 0.203 inch, 0.187 inch, and 0.171 inch. During the swaging operation it was noted that the iron was extremely ductile.

The microstructures of both the as-received and the cold worked material have been characterized. The grain size of the as-received material was very large, averaging 1.25 mm in diameter. After cold work, a pronounced wrought structure prevailed and a veining type substructure was detectable. Annealing treatments in high vacuum to relieve stresses and produce the desired fine grain size are in progress. Complete characterization of this material is required so that comparisons between this material, high purity iron currently on order from other suppliers and iron to be refined on-site can be made.

A set of grips for tensile testing one-eighth inch diameter specimens has been designed and is currently being fabricated. These grips are to be used in conjunction with both the instron testing equipment and x-ray diffraction equipment.

6. GAS-GRAPHITE STUDIES

EGCR Graphite Irradiation

The fourth capsule in the series of irradiations of EGCR graphite, H-3-4, is operating satisfactorily in the third cycle in the GETR. Sample temperatures have returned to normal (450 to 825 C) from the high values reported last month. The reason for the temperature peaking has not been resolved.

Radioactivity from Irradiated Graphite

Gamma-emission spectrometric analyses were made on several samples of irradiated nuclear-grade graphite to determine which isotopes cause the radioactivity observed in the samples. Materials analyzed were CSF, TSX, TSGBF, NC7 and NC8. The period from the termination of irradiation to the date of analysis varied from 10 days to 850 days. The principal gamma emitters, which presumably originate from impurities in the graphite, were found to be scandium-46, cobalt-60, zinc-65, and antimony-124.

To obtain a more quantitative comparison of the activities of these four isotopes, calculations were made for samples irradiated for approximately 100 days at a fast flux of 2×10^{14} nv ($E > 0.18$ Mev) and a thermal flux of 1.35×10^{14} ($E > 0.17$ ev). The calculations indicate that for the first 100 days after irradiation nearly 85 percent of the gamma activity is caused by the scandium-46; after 200 days this has decreased to 75 percent; and after 800 days over 85 percent of the activity is due to the cobalt-60. These results after being converted to microcuries, were compared with "Cutie-Pie" measurements made on each sample. A conversion of 0.5 mr/hr per microcurie (at 1 meter) was used based on an average gamma energy of approximately 1.0 Mev for the four isotopes. The two results agreed within approximately 10 percent.

Calculations were also made for the isotopes causing beta activity. The principal beta emitters (beta only) were found to be phosphorus-32 and sulfur-35. The two tables summarize the activities for gamma emitters and gamma-plus-beta emitters.

Calculated Activity of Impurities in Graphite - Gamma Emitters

	Activity, %				
Days After Irradiation	0	20	100	200	800
Scandium-46	88	87	84	75	4
Cobalt-60	4	5	8	16	87
Zinc-65	3	3	5	7	9
Antimony-124	5	5	3	2	-

Calculated Activity of Impurities in Graphite - Gamma-Plus-Beta Emitters

	Activity, %				
Days After Irradiation	0	20	100	200	800
Scandium-46	62	73	84	75	4
Cobalt-60	3	4	8	16	87
Zinc-65	2	3	5	7	9
Antimony-124	4	4	1	2	-
Phosphorus-32	28	15	1	-	-
Sulfur-35	1	1	1	-	-

Boronated Graphite Irradiations

A heat transfer test was run in the Snout II facility confirming calculations which indicated relatively high heating rates of samples of graphite which contain boron. This experiment and calculations of burnup and heating rates are necessary for further planning of irradiations of the Fermi reactor shield graphite. As an example, a small sphere (0.6 cm diameter) will generate 4 watts/g in a thermal flux of 1×10^{13} nv. This heating load is excessively high for many experimental techniques used for graphite irradiations.

Gamma Irradiation Facility

An additional 51,000 curies of cobalt-60 were added to the gamma irradiation facility in the 3730 Building, bringing the total amount in the source to 61,000 curies. Dose rate measurements of the combined 61,000-curie source indicated a maximum of 3.45×10^6 r/hr in the usual configuration in which four irradiation experiments are conducted simultaneously in 2-inch diameter tubes positioned within the source. For higher dose rates, a source holder was fabricated for use with a 1-inch diameter irradiation tube. When 50,000 curies of Co-60 were placed around the 2-inch diameter tube, the dose rate was 6.6×10^6 r/hr as measured by ceric sulfate dosimetry. Glass loops are being assembled to continue experiments on the effects of gamma radiation on gas-graphite reactions.

Graphite Oxidation Tests in the PRTR

A study is in progress to determine the effect of gamma intensity on the rate of oxidation of graphite in air. The highest gamma intensity available until recently was the Co-60 facility in the 3730 Building in which experiments have been conducted at 1×10^6 r/hr. In an effort to measure the oxidation rate at an intensity closer to that expected in the EGCR after shutdown ($\sim 10^7$ r/hr), tests were conducted during shutdown in PRTR tube 1556 from which the fuel had been discharged. A series of 10 ceric sulfate dosimetry measurements indicated that the gamma intensity nine hours after shutdown was 5.6×10^6 r/hr. The intensity decayed with a 13.9-hour half-life as indicated by the equation:

$$\gamma\ (\text{r/hr}) = 8.70 \times 10^6\ e^{-0.05t} \qquad \pm 5\%$$

where t = hr after shutdown for the period $8.8 \leq t \leq 20.8$

The oxidation test was conducted at 610 C. During the period of the test the gamma intensity decreased from 5×10^6 to 3×10^6 r/hr. The oxidation rate of 0.02 g/g/hr lies within the scatter band of the tests conducted at 1×10^6 r/hr. More tests are required to establish a valid rate equation. Future tests will be conducted in the Co-60 facility in the 3730 Building in which a dose rate of 6.6×10^6 r/hr can now be achieved.

Graphite Compression Test

The latest results obtained from the AGOT-LS (NPR reflector) graphite samples in the latest 150 psi compression boat tend to substantiate the results observed in the previous test. The results suggest that dimensions may affect the rate of contraction of irradiated graphite. The contraction rate of the 1/4 inch diameter AGOT-LS transverse samples was on the order of -0.01 percent per 1000 Mwd/At. The total amount of contraction after approximately 6500 Mwd/At appears to be less than one-half the contraction of standard size samples (0.43-inch diameter) cut from the same graphite bar. A difference in the contraction of full-size bars and standard-size samples was reported in the December 1961 Monthly Report.

The differences between the length changes of loaded and unloaded samples apparently decreases with increasing exposure. Both the loaded and unloaded samples now appear to be deforming at approximately the same rate as initially observed for the loaded samples.

7. ALUMINUM CORROSION AND ALLOY DEVELOPMENT

Dynamic Corrosion Tests

The corrosion test to determine the corrosion rates of A212 carbon steel, RH 1031 carbon steel, RH 1051 carbon steel, 304 stainless steel, Hunter-Douglas C-1 alloy aluminum, and Zircaloy-2 in pH 6-7, 290 C, deionized water has been completed after 2500 hours of exposure. The aluminum samples were all removed after 1800 hours of exposure and were corroding at a rate of 70 mils/year. The stainless steel corrosion rate fell from 0.09 mil/year during the first half of the test to < 0.01 mil/year after 2500 hours. The carbon steel rate was also decreasing from the 0.26 mil/year obtained during the first 1800 hours of the test, but an exact determination of the rate could not be obtained from the limited amount of samples available during the remaining 700 hours of exposure. The Zr-2 rate was < 0.02 mil/year after 2500 hours.

The test of corrosion and crud deposition of coupons in reactor H-1 loop at pH 7 continues. Since the test was charged on April 13, it has accumulated only about one week of high temperature operation (260-290 C). The temperature was held down for the first few days of the test because of high pH and conductivity. A few days after reaching high temperature a leak developed in a sample line. Since this could not be valved off, it was necessary to reduce the temperature and pressure until this could be repaired. The reactor was down for a retubing outage May 13 to 28. Although the first few days of the test were characterized by poor water quality, the clean-up resin was replaced, and high water purity has been maintained since then (~10^6 ohm).

Aluminum Alloy Development

Several melts made to determine the reproducibility of the corrosion resistance of three high purity base alloys have now received an exposure of 15 months in 360 C water. The three alloys are (1) 0.64% Ni, 2.1% Fe; (2) 1.2% Ni, 1.8% Fe; and (3) 1.5% Ni, 1.5% Fe, all cast with 99.995% base aluminum. At 15 months all melts still show low penetrations, the highest penetration being 0.64 mil.

After nine months of exposure to 360 C water melts of the 1.2% Ni, 1.8% Fe in 99.995% aluminum fabricated to determine the effect of casting variables still show that the only change in casting procedure which affected the corrosion resistance of the alloy was a slow cooling rate. Different holding times, holding temperatures and pouring temperatures have not changed the corrosion characteristics of this alloy.

8. AEC-AECL PROGRAM

The feasibility of using drilled holes as primary calibration standards for the ultrasonic defect test on Zircaloy tubing is being determined. Two sets of small holes were fabricated, one set with varying diameter, the other with varying depths. Ultrasonic response measurements of these holes are in progress.

Raised portions of metal, such as a burr, sometimes give inordinately large ultrasonic responses for their size. Two plateaus of metal were formed on an outer surface by a chemical milling technique. In a transverse test with a flat rectangular crystal the plateaus did not give as large a response as notches or holes with comparable reflecting surfaces. Other types of raised pieces of metal, such as burrs, will be studied.

A curve of ultrasonic response as a function of entry angle for a spot-focused 10 Mc Li_2SO_4 crystal has been plotted from 14 degrees through 30 degrees. Unlike the curves taken with the line focused and the flat, rectangular crystals, this curve does not show distinct peaks at certain angles. However, there are small peaks which rise from 5 to 15 percent above the general curve at entry angles of 20°, 23°, and 26°; the 20° and 26° entry angles correspond to the sharp peaks obtained with the other crystals. Thus, the spot-focused crystal was not as sensitive to changes in angle as the other two crystals.

Fifteen experimental boiling burnout points were determined in the laboratory with an electrically heated model of a 19-rod bundle fuel element. The test section was made of 19 Inconel tubes, 0.629 inches OD, 19½ long and was installed in a 3.25-inch ID horizontal coolant tube. Twelve of the rods were wrapped with wire to maintain a 0.015-inch spacing between rods and to promote coolant mixing. All of the tubes had thermocouples placed to measure the inside surface temperature near the downstream end.

The tests were performed by setting a flow and inlet water temperature and then increasing the power in steps until boiling burnout was reached. The onset of boiling burnout was defined by a temperature excursion of one or more of the surface thermocouples. The tests were performed at a pressure of 1200 psig and at flow rates between 500,000 and 4,000,000 lb/hr-sq ft.

In this test section the radial heat generation (from rod to rod) was uniform. Therefore, since the spacing between the outer rods and the coolant tube was greater than the spacing between rods, the burnout generally took place on the inner rods. A total of seven and possibly eight different rods indicated burnout at one time or another, all occurring on the inner rods except for one burnout indication on the outer rods. On one occasion four different rods showed burnout while several burnout excursions came on two rods at once. During the runs at the lower flow rates the rods in the upper portion of the bundle indicated burnout. This would be expected because of the stratification influence at low flows, though the effect was not as pronounced at mass velocities of 3 and 4×10^6 lb/hr-sq ft and the lower enthalpies.

The concentration of the burnouts on the seven inner rods was to be expected. The hydraulic diameter of the channel between the process tube and the outer 12 rods as compared to the hydraulic diameter of the inner seven rods does not match the heat generating surfaces seen by these two flow paths. Since the inner seven rods receive less than their "share" of flow and since there is presumably little or no mixing between channels, these rods experience the larger enthalpy rise. On the other hand, if this disparity was overly severe, one would expect that the center rod would see a large portion of the burnouts, which was not the case.

Analysis of the data has been started, but no quantitative conclusions can be made yet. However, the raw data do permit several qualitative conclusions. The boiling burnout heat fluxes obtained with this test section are much reduced at the same bulk coolant conditions from those obtained with a similar test section which had a 0.074-inch spacing between rods. The 0.074-inch spaced test section had boiling burnout heat fluxes comparable to test sections of tubular or annular geometry. The reduction in burnout heat fluxes with the 0.015-inch spaced rods is more pronounced at low coolant flows. At a coolant flow rate of 500,000 lb/hr-sq ft, the boiling burnout heat fluxes were about 150,000 Btu/hr-sq ft, and generally independent of exit enthalpy from about 150 F subcooled to 15% quality by weight. These fluxes are reductions on the order of a factor of five from those obtained with the 0.074-inch spaced bundle. At a coolant flow rate of 3,000,000 lb/hr-sq ft, the boiling burnout heat flux was more dependent upon the exit enthalpy and ranged from about 600,000 to 950,000 Btu/hr-sq ft at exit enthalpies corresponding to about 5% quality by weight to 150 F subcooling, respectively. These fluxes are reductions on the order of a factor of about two from those of the 0.074-inch spaced bundle.

The heat fluxes quoted above show a strong coolant flow rate effect upon the boiling burnout heat fluxes. The fluxes for other mass flow rates substantiate this. Boiling burnout heat fluxes between 250,000 to 380,000 and between 480,000 to 500,000 Btu/hr-sq ft were obtained at coolant flow rates of 1,000,000 and 2,000,000 lb/hr sq ft, respectively. Boiling burnout heat fluxes obtained at a coolant flow rate of 4,000,000 lb/hr-sq ft were not much higher than those obtained at 3,000,000 lb/hr-sq ft, indicating that the strong effect of coolant flow rate may become less pronounced at the higher rates.

The behavior of the thermocouples used to measure rod surface temperatures gave some indication of the start of boiling. It may be deduced from this behavior that boiling started within the bundle at conditions where the bulk coolant was highly subcooled. This indicates poor mixing between the water within the bundle and the considerable amount of water between the bundle and tube wall, with the water within the bundle at a much higher enthalpy than the bulk average. This phenomenon may be largely responsible for the reduction in boiling burnout heat fluxes found with this test section.

9. REACTOR STUDIES PROGRAM

Advanced Reactor Concepts Studies

Preliminary estimates of core size for a 30 MWt fast reactor cooled by boiling rubidium show that the size will be governed by thermal hydraulic considerations rather than by fuel inventory requirements. For a cylindrical core of equal length and diameter and with 50 v/o coolant, a typical exit coolant velocity would be 300 fps. This value is insensitive to exit quality. Average heat flux with 3/16 inch diameter fuel pins would be 1.3×10^6 Btu/hr ft^2. Available published data on the boiling alkali metals are insufficient to support estimates of heat transfer coefficients, points of departure from nucleate boiling, or a reasonable range of coolant qualities and velocities. For the initial layout of this reactor, it will be necessary to make some arbitrary assumptions as to coolant capabilities.

Initial reactivity calculations for rubidium cooled, plutonium fueled fast reactor cores have been completed by Applied Physics Operation. Calculations to narrow down the critical reactor sizes to those of interest are now under way.

A review of the current status of thermionic converter technology and research was gained by attendance at the Thermionic Power Conversion Symposium. Thermionic power conversion remains a very promising scheme for reactors with space missions, but formidable materials problems remain to be solved before useful efficiencies and endurance can be demonstrated in the laboratory.

D. RADIATION EFFECTS ON METALS - 5000 PROGRAM

Single crystal and polycrystalline molybdenum containing interstitial and excess carbon as impurity are being studied to establish the combined effect of neutron irradiation and carbon impurity level on the properties of the metal. This program includes the following tests, measurements and studies: (1) tensile tests, (2) electrical resistivity measurements, deformation studies, electron microscope studies, x-ray diffraction analyses, microhardness tests, length change measurements, and some pre- and post-irradiation damage recovery studies.

Four single crystal bend test specimens recharged into the irradiation facility after an irradiation exposure of 5.7×10^{18} nvt (fast) have now attained the goal exposure of 1×10^{19} nvt (fast)(as determined from titanium flux monitors) and have been discharged. Two of these specimens contain 10-20 ppm C, and the other two contain 400-500 ppm C.

Polycrystalline foils, 0.003 inch thick, prepared from Johnson-Mathey high purity molybdenum have been thinned after an exposure of 1×10^{19} nvt (fast) and have been examined by transmission electron microscopy. No change in the pre-irradiation microstructure was detected. (Prior to irradiation these as-rolled foils had been stress relieved at 700 C.) They are currently being analyzed by x-ray diffraction techniques and will be subjected to various post-irradiation annealing treatments. Foils of polycrystalline molybdenum containing three levels of carbon as impurity (10-20 ppm, 100-200 ppm, and 400-500 ppm) have been irradiated to 1×10^{18} and 1×10^{19} nvt (fast); they will be processed by x-ray diffraction and electron microscope techniques.

Appreciable recovery of damage sustained by a specimen may occur during irradiation if gamma heating of capsule components is not dissipated. Tests on tensile specimens with 0.180 inch gage diameters have been conducted. The temperature at the center of the gage section of a specimen contained in a prototype capsule during irradiation was determined. The maximum temperature in the molybdenum tensile specimen in an environment of helium at a pressure of one atmosphere was 57 C, at a pressure of two atmospheres it was 55 C, and in a vacuum environment it was 160 C. Helium as a heat transfer medium must be added to capsules which do not have adequate internal metal-metal thermal conductance. New capsules with better thermal conductivity have therefore been designed and are being fabricated. They will be used in future irradiations of single crystal and polycrystalline specimens.

Resistivity measurements are being made on control non-irradiated specimens to evaluate the utility of mercury contacts on specimens. Preliminary tests are encouraging. Accurate length measurements on 112 single crystal tensile specimens require equipment which was not available for use during the past month. Irradiation of these crystals was therefore postponed. Pre-irradiation single crystal x-ray diffraction data from 24 specimens have been obtained, and include x-ray intensity, line shape, and lattice parameter data. A unique characteristic of the specimens containing high carbon (400-500 ppm) is that diffraction lines show fine structure. The high carbon content is presumably responsible for a polygonized structure within the matrix.

Field ion microscopy utilizing emitters prepared from specimens containing carbon as interstitial impurity will be evaluated under a joint Hanford-Linfield Research Institute program. Work has begun on preparing emitters suitable for irradiation and study both before and after irradiation.

E. CUSTOMER WORK

1. RADIOMETALLURGY EXAMINATIONS

The rotary core drill assembly for sludge samples of Redox waste storage tank was disassembled and contained liquid rather than a solid sample (RM 367).

Examination of an overbore and a standard production element revealed both elements failed due to groove corrosion penetrating the aluminum cladding (RM 443 and 448). Two low exposure I&E elements were received with separations at the male end caps. No defects were found in the weld closures (RM 449).

2. EQUIPMENT PROJECTS

CGH-857 Physical & Mechanical Properties Testing Cell

All portions of the Instron Tester with the exception of the in-cell leads are now scheduled for delivery in "early June 1962." The approval drawings for the Impact Tester were returned to the vendor. Delivery is scheduled for 7/27/62.

The B & L Precision Gage is scheduled for shipment on 6/15/62.

CGH-858 High Level Utility Cell

Minor construction forces have completed their work on the project. An engineer from AMF has installed the two extended reach manipulators and instructed the Radiometallurgy maintenance group in the maintenance and repair of the manipulators. The engineer also instructed the operating personnel on the operation of the manipulators.

Vacuum Annealing Furnace (E-9)

A furnace has been designed and fabricated for vacuum annealing of preformed samples. The furnace to be installed in the north end of "E" cell, has a maximum working temperature of 1850 F.

Micro Sampling Equipment

The fabrication of the micro sampling equipment has been completed. A wooden, cell mockup is being fabricated for equipment set-up and testing.

3. METALLOGRAPHY LABORATORIES

Three flanging tools, used to expand the ends of the NPR Zircaloy process tubes, failed in service during the month. The parts which are similar to a tapered roller bearing had been hardened to Rockwell C-65. Failure was a type of fatigue induced cracking which originated just below the surface in a circumferential ring at the location of the highest stresses.

A variation in polishing technique for certain samples, especially those containing uranium, has resulted in a much longer service life for polishing cloths in the vibratory polishing units. A solution of two percent sodium dichromate is substituted for the previously used two percent chromic acid. (A small amount of chromic acid must still be added to obtain proper results on the uranium.) The new solution produces excellent polishing results and does not attack or weaken the fibers of the polishing cloths.

An Elgeet Olympus metallurgical microscope, with camera attachment was added to the laboratory this month. This unit has an inverted stage, and photographs can be taken on either 35 mm film or with a Polaroid attachment. The microscope has four types of illumination; bright field, polarized light, sensitive tint, and phase contrast. The phase contrast can be varied through four steps or degrees of contrast. Very reasonable exposure time on the camera results even when using fully polarized light at high magnification.

Replicas have been examined from a polished, cathodically vacuum etched, irradiated, NPR inner tube cross-section surface. The material was irradiated to approximately 1000 MWD/T. No swelling pores were observed in the replicas examined. A type of artifact was observed which has been noticed previously. The artifact can be identified as 0.2 micron circular bumps on the specimen (depressions in the replica) that are homogeneously distributed over uranium and zirconium carbide particles. The source of the artifact is not known. The presence of condensed water vapor on the specimen at the time the softened plastic is added or particles deposited on the specimen surface during cathodic etching are two possible explanations.

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

4. N-REACTOR CHARGING MACHINE

Modifications

The modifications to the hydraulic connector control valves which were mentioned in the April monthly report have been completed and tested. The modifications corrected the shock condition.

About 130 hours of electrical craft time was expended in additions and modifications to the control circuitry.

Additional modifications to the "C" elevator magazine support trolley have been completed. These consisted of installing cam rollers in place of the trolley wheels and the fabrication and installation of a new brake assembly. Initial testing of these components indicate that performance has been improved.

A magazine drive roller, of new design, was fabricated, installed, and tested. This drive roller has completed 110 cycles without incurring excessive damage. Previous drive rollers were severely damaged after 65 cycles.

The dummy plugs on the plug conveyors have been removed and modified to incorporate a chain tensioning device. The earlier models did not include this feature and when the plug conveyors were operated forward, slack chain gathered immediately ahead of the drive sprocket. On one occasion the chain caught in the drive sprocket and broke. The modified system has not been tested.

Testing

Acceleration, deceleration, velocity and optimum setting of the pressure reducing valve and the flow control valve for the unloaded transfer arm drive system was investigated. The investigation indicates that there is a velocity, above which the unloaded transfer arm drive system will oscillate. This velocity is significantly lower than the design velocity. The scope of the modifications will be determined after the loaded transfer arm tests are performed.

During the month three fuel element charges were performed using an NPR process tube with nozzles. Prior to the first charging, the process tube was completely inspected using a boroscope. No serious defects were found. The first charge, consisting of nineteen 24-inch fuel elements, was made with a flow of 14 gpm flow of sanitary water through the tube. Discharging of the tube was accomplished using an aluminum push pole with teflon feet, pushed from a magazine into the process tube using the charging machine. Immediately after discharging the push pole was removed manually and the tube was dried. The tube was reinspected using a boroscope, and scratches were noted starting about 16 feet from the front of the tube, becoming more numerous toward the discharge end.

A second charge was made using new fuel (eighteen 24-inch pieces). The same charging, discharging, and drying techniques were used. About the same amount of scratching was observed; however, the scratches started at about four feet from the front of the tube. Photographs of the scratches were taken through the boroscope. An impression of a portion of the scratches near the rear of the tube was also made using room-temperature-hardening synthetic rubber. From this impression, the depth of the scratches was measured at between 0.3 mils and 1.1 mils with the majority ranging between 0.5 and 0.7 mils.

Subsequently, a charge was made using twelve long and eight short dummies (spacers). The tube was rotated prior to this test to present a fresh surface to the dummy feet. No scratching was observed as a result of this test.

5. SPECIAL PLUTONIUM FABRICATIONS

Fission Product Transient Samples for Phillips

Fabrication of the fission product transient samples for Phillips Petroleum Company is in progress. Substitution of Al - 2 w/o Si alloy cladding for the originally specified high purity aluminum has resulted in better quality extrusions. The better compatibility of the silicon alloy with the plutonium-aluminum core alloys has minimized the "dogboning" at the trailing end of the core.

Several of the elements require additions of small amounts of lithium for calibration purposes. Some difficulty has been encountered in controlling the oxidation of the lithium during alloying. Melting of the material in a controlled atmosphere of nitrogen appears to have corrected this problem.

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Manager, Reactor and Fuels Research and Development

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PHYSICS AND INSTRUMENT RESEARCH AND DEVELOPMENT OPERATION

MONTHLY REPORT

MAY 1962

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

Exponential Pile Experiments for NPR

The strength of the NPR control system has been measured with the correct rod spacing. The change in buckling of the wet lattice is 179 μB upon insertion of six rods. The previous value was 222 μB. The change is in approximate agreement with the change expected from a simple flux squared weighting of the rod worth. The range of rod strength calculated (HW-68407) is 182-220 μB, where this range arises from two methods of calculating the average transport mean free path.

A previous study (March, 1962) showed the need to determine more precise values of extrapolation distances from exponential experiments. Since then flux traverses from various NPR mockup experiments, which included variations in fuel loadings and flooding conditions, have been re-analyzed. The results were compared to see if one consistent set of extrapolation distances could be used to describe all experiments. The conclusions are:

(1) One can determine a consistent set of extrapolation distances that will apply to many experiments provided that the spectra are reasonably similar.

(2) The horizontal extrapolation lengths are more uniform than the vertical extrapolation lengths, which have large variations in a few cases.

(3) The consistent set of extrapolation distances which should apply to any NPR mockup experiment with fuel is 2.02, 1.60, and 2.29 inches for side-to-side, front-to-rear, and vertical extrapolation distances, respectively.

These experiments were analyzed with a new BVTθCθ program which will be described later.

The derivation of effective cross section and neutron temperatures from copper traverses has been completed and an article describing the work has been started.

Exponential pile flux traverses deviate from a pure exponential near the bottom of the fueled region because of transients at the interface with the base. A two group calculation (reported in March) reproduced the deviation for the first case tested. The same method was used for a second case, but this time the calculation accounted for only half of the deviation. No explanation for the remaining discrepancy has been found.

The cadmium covers for the BF_3 neutron counter tubes are open at the back to provide space for the cable. The leakage through this opening was measured to determine the error in the cadmium ratio. The epicadmium count rate with the hole open is 2% high for the half-inch tube and 7% high for the quarter-inch tube in a position where the cadmium ratio is about 40. The experiment was not affected by flux depression caused by the cadmium since symmetric geometries were chosen. An experiment was also performed to determine the epithermal flux depression at a cadmium covered BF_3 tube caused by the cadmium cover on the tube itself. The results show a few percent flux depression for the half-inch BF_3 cadmium sleeve but a negligible depression for the quarter-inch cadmium sleeve.

Slowing Down Theory and the Modified Gas Model

The first order approximation to the scattering law for a material will necessarily involve some sort of first moment. Slowing down or Fermi age theory employs the first moment in lethargy. On the other hand, as indicated by the modified gas model, the first moment in energy gives better results in the thermal region. This creates a problem in the joining region and suggests using the equivalent of slowing down theory with the scattering law approximated by the first moment in energy rather than in lethargy. The only change in the theory is that wherever the logarithmic decrement ξ appeared there now appears $2A/(A+1)^2$. Whether one should use the first moment in lethargy or first moment in energy has been a controversy extending back to the mid-thirties, although the usually accepted view is to use the first moment in lethargy. Consequently, spectrum comparisons between the two slowing down models and an exact calculation are planned before deciding on a final method.

Computational Programming Services

BVTθCθ, the program combining the two exponential data processing codes VTθCL and CθFIT2, is in preliminary production status, and a number of cases have been processed satisfactorily.

Instrumentation

Further testing continued with the prototype unit of twenty-four transistorized gamma spectrometers to be built by GE-APED for the NPR Fuel Failure Monitor. Although most of the spectrometer functions satisfactorily in tests, the high voltage supply and one discriminator still show excessive sensitivity to temperature and line voltage changes and to line noise. These sections have been returned to the vendor for further design changes while testing continues on the remainder of the prototype. Instrument Design, CE&UO, also was provided assistance in preparation of acceptance tests for the spectrometers.

A comment issue specification was prepared at the request of Advanced Engineering, FPD, concerning procurement of a source range neutron monitor system for the Hanford Test Reactor. Specifications similar to those for NPR were used so that valuable performance data might be obtained before NPR startup. In addition, the Hanford Test Reactor system will then be able to benefit from the NPR spare parts inventory.

A GE-APED current pulse amplifier was obtained and is being tested as a part of the instrument procurement program related to NPR.

Discussions were held with Tracerlab, Inc., representatives regarding their G.M. Tube detector count-rate and current logarithmic response area monitors purchased for use in N-Reactor buildings.

An estimate of instrumentation needs was given to Irradiation Testing, IPD, for their rupture monitor under the HAPO-264 Program.

Advice and assistance was given Research and Engineering, IPD, concerning the titanium wire to be used in the NPR traveling wire neutron flux monitor. The irradiated wire storage drum is shielded by 1.38 inches of lead as based on the low energy (0.32 Mev) gammas from titanium. Analysis of the wire indicates the presence of about 0.005% manganese and this will require several inches of lead to obtain adequate shielding. Further calculations will be done to establish shielding requirements as soon as refined analysis data become available.

Authorization was received to start detailed design and procurement of experimental fuel failure detection instrumentation for the fuel element testing loop at PRTR. A meeting was held with various PRTR, Instrumentation Design-CE&UO, and Reactor Physics-IPD members to review the project scope, the practicality of certain design and layout features, and to establish the proper lines of effort and liaison.

The 6" x 6" Process Tube Distortion Traversing Mechanism (Mark IV) was tested at 185-F in an out-of-reactor process tube. The target and image was clear and easy to read through the borescope even in a severely bent tube at full borescope length. Complete runs through the full length (40 feet) of process tube were made in as short a time as 20 minutes. One run yields data on both horizontal and vertical displacements. Analysis of the data shows that the general shape of the tube is faithfully indicated. The results were presented to IPD people at two meetings. At the last meeting, it was decided to use the 6" x 6" Traverse Mechanism for horizontal displacement measurements in B, D, and F Areas. A spare 6" x 6" unit is being completed in the Optical Shop. A new 12" x 12" unit also being fabricated is expected to reduce errors in the indicated displacement by a factor of four and decrease the time needed to make a run through the process tube. Because of its increased sensitivity, it will not have sufficient range to give on-scale readings in the most severely bent process tubes.

A proposal submitted to IPD to develop an electrical readout traversing mechanism for NPR was favorably received. Shop sketches detailing the construction of a demonstration model have been prepared and fabrication has begun.

Systems Studies

Technical consultations on NPR instrumentation continued in support of the NPR Project Section. Meetings of the NPR Failure Sequence Analysis Group were attended. Assistance was given in analyzing the possible failure modes of equipment in the primary loop pressure and inventory systems. Several sessions of the Burns and Rowe presentation on NPR systems were attended. A meeting was held in Los Angeles to determine the signal information needed for programming the Central Data Logger, and to discuss methods of flow monitor isolation recently proposed by Information Systems, Inc. Our recommendations are being withheld until the vendor discloses specific configurations and component values of the proposed isolation circuits.

Work continued on the preparation of preliminary specifications for the NPR simulator analog computer. The equipment list has been completed and the operational specifications are now being prepared. Contacts have been made with the major manufacturers of analog computers to discuss the types of analog computing equipment now available. Assistance was given to IPD and CE&UO personnel in the preparation of information to be included in the budget data sheet and design proposal for the simulator. A review of the equipment and space requirements was completed and estimates of design and procurement schedules and costs were submitted to Project Engineering Operation, IPD.

The reactor instrumentation problem using an eleven-node kinetics simulation is progressing as expected. The simulation is operating satisfactorily except for operating difficulties with the GEDA computer.

Application of Dynamic Optimization techniques to nuclear reactor control systems continued. Approaches tried last month have still not yielded simple results that appear practical. The non-linearities in the Xenon equations will probably have to be handled by some type of approximation not yet tried.

SEPARATIONS

Experiments with Plutonium Solutions

Criticality experiments were initiated with a water-reflected assembly during the month. These are the first measurements with the 14-inch sphere, fully reflected with water. The concentration of Pu in the nitrate solution was ~ 44 g Pu/ℓ, with a nitric acid molarity of about four. From the inverse multiplication curves during the critical approach, the critical volume and mass were ~ 21 liters, and 924 g Pu.

The data obtained from the water-reflected experiments will provide a needed tie point for comparing the current experimental results with the early Hanford P-11 experiments which were conducted with dilute Pu solutions. The acid molarity and Pu concentration will be adjusted to obtain criticality in the full sphere (23.2 liters) for a more direct comparison with the P-11 results.

In the above experiment the system was not taken to full criticality because of mechanical difficulties; there was some indication of a small Pu leak back through one of the valves.

After being inoperative for a period of time because of the assignment of key personnel and equipment to the Recuplex incident, considerable time was involved in placing the equipment at the Critical Mass Laboratory back into proper operating condition. After checkout of the instrumentation and safety system, the critical assembly fuel addition pumps were found to be inoperative. Investigation disclosed two air operated valves to have broken diaphragms; upon replacement of the diaphragms the pumps worked satisfactorily. At this point the control rod became inoperative. The unit was disassembled and examined; a stretch wire to the magnet coil was found broken; the unit was repaired and now operates satisfactorily. Pu contamination was handled as a matter of routine and without incident during these operations.

During the month exhaust fans were installed in the control room to increase the air flow for improved cooling of the instruments which were overheating. New type radiation dosimeters were mounted in the control room, hallway, and mixing room at the facility.

A practice evacuation for a criticality incident was held on May 29. Approximately four minutes were required to evacuate Building 209-E, assemble in the lunch room at the Hot Semi Works, and complete an accounting of all personnel in the Laboratory at the time the criticality evacuation alarm was sounded. Seventeen persons were at the Laboratory and participated in the evacuation.

Experiments with Plutonium Oxide Plastic Mixtures

The critical assembly (split-half) machine, for use in criticality measurements with PuO_2-plastic mixtures, was moved to the Critical Mass Laboratory from the Technical Shops where it was constructed. The electrical work and subsequent testing of this device are to be completed at the Laboratory. Experiments will begin after completion of a Hazards Summary report.

Limiting Critical Concentration of Pu^{239} in Aqueous Solutions by Monte Carlo Techniques

The limiting critical concentration of a plutonium-water mixture was previously determined from k_∞ measurements in the PCTR to be 8.4 ± 1 g Pu/ℓ[1]; whereas, the limiting critical concentration was calculated to be ~7.25 g Pu^{239}/ℓ utilizing the 9-Zoom Multi-Group Diffusion Code[2]. In an effort to resolve this difference, a series of calculations were made of k_∞ for Pu^{239}-water mixtures with the HISMC Monte Carlo Code. The infinite multiplication constant was computed for concentrations of 7.5, 8.4, and 11.4 g Pu^{239}/ℓ. A total of four thousand neutron histories were traced for the 7.5 and 8.4 g/ℓ concentrations, and three thousand histories for the 11.4 g/ℓ concentration. The results of the calculations are summarized in the table below.

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(1) Masterson, R. H., et al., "Limiting Critical Concentrations for a Plutonium-Nitrate Solution and for a Uranium-235 Solution", Physics Research and Development Operation Quarterly Report, January, February, March, HW-73116, April, 1962.

(2) Reardon, W. A., Private Communication.

Monte Carlo Calculations of k_∞ for Pu^{239}-
Water Mixtures

Run No.	7.5 g/l Absorptions	7.5 g/l Fissions	8.4 g/l Absorptions	8.4 g/l Fissions	11.4 g/l Absorptions	11.4 g/l Fissions
1	996.69	336.99	989.62	351.77	1021.15	420.40
2	980.00	332.26	987.55	350.40	996.52	402.86
3	1041.02	362.81	1040.84	383.31	990.49	408.08
4	980.18	303.83	985.79	316.81		
TOTALS	3997.89	1335.89	4003.80	1402.29	3008.16	1231.34
k_∞	0.9724		1.0192		1.1912	

From the Monte Carlo calculations, the limiting concentration of a Pu^{239} water mixture is 8.0 g Pu^{239}/l; this value is in better agreement with the experimental value, and is within the uncertainty of the experimental error.

Measurement of k_∞ in the PCTR for Dilute Solutions

Stainless steel tanks were used for the containment vessels in PCTR experiments for determining the limiting concentration of a Pu-water mixture. The uncertainty in the measured value of 8.4 ± 1 g Pu/l is the result of the uncertainty in the correction for the effect of the stainless steel on the measured value of k_∞. An experiment is planned during June in the PCTR to study the effect of the stainless steel and to further evaluate the magnitude of the correction. In this case, the experiments will be conducted with highly enriched UO_2F_2 solutions, since the results are expected to be applicable to the measurements with dilute Pu solutions, and there will be no potential hazard to the PCTR from Pu contamination.

Subcritical Interactions

The calculation of interaction probability functions between pairs of subcritical assemblies has been extended to include both parallel and perpendicular slabs, of equal area parallel cylinders with and without partial shadowing, and a parallel slab and cylinder. Using these functions, the calculated k_{eff} agree with measured critical systems of these geometries to about ± 3%. Some theoretical work is now being done on n component systems, $n > 2$. An effort is being made to reduce the problem to an eigenvalue matrix with better definition of the parameters than has been used previously.

[redacted] [redacted]

Transport Theory Analysis of Plutonium Metal-Solution Systems

The analysis of this problem, which was described in the March monthly report, has been completed by use of Program S-X and cross sections based on LAMS-2543. The final form of the data has been obtained in a form suitable for nuclear safety applications, namely, an "envelope" curve which gives the minimum critical mass for a particular volume of the system, without specific regard to the ratio of plutonium metal to plutonium in solution for which this minimum mass occurs.

The transport theory curve as obtained gives a lower minimum critical mass for the metal-solution systems than was obtained with earlier diffusion theory (18-group 9-ZOOM) calculations. For homogeneous systems (i.e., all solid metal dissolved), however, the transport theory calculations did not agree with experiment, in contrast to the diffusion theory results. On the basis of the reasonable assumption that the cross sections used in the Program S-X calculations are satisfactory for the faster spectra of the metal-solution systems, while the 9-ZOOM cross sections are satisfactory for the homogeneous systems, it was decided to join these two curves together rather than search for the difficulties with the Program S-X calculations of the homogeneous systems.

Computational Programming

QUENCH, a version of TRIP, has been prepared to do kinetics calculations on a range of space independent reactors with compositions and time-dependent reactivity behavior simulating possible conditions in the K-9 nuclear excursion. The purpose of these calculations is to identify plausible explanations of the estimated magnitude of the initial excursion. A few preliminary calculations have given encouraging results.

Instrumentation

One method of using instrumentation to monitor the buildup of plutonium in Hood Nine at the 234-5 Building, CPD, was tried. The attempted method was discarded after a series of measurements proved it to be impractical in its present form. A gamma energy spectrometer was assembled and used to measure moderate quantities of plutonium; however, the high gamma dose rates at Hood Nine swamped the system. The mechanical arrangement of the hood prevents the use of adequate lead collimators while still permitting proper observation of the required areas. An alternate scheme remains to be devised. The work is being done in cooperation with Finished Products Chemical Technology, CPD.

The initial phase of the "C" column simulation was completed this month. This simulation is concerned with the evaluation of several possible mathematical models of a chemical separations column. These models were derived by Chemical Research and Development. All previous work was performed using only one of the tentative mathematical models. This model proved fairly satisfactory in matching the results of eighteen of the twenty experimental column runs. Two of these runs did not fit the model satisfactorily. A new model incorporating an additional variable (organic backmixing) was programmed on the analog computer this month. The new model gave considerably improved results in matching the two experimental runs in question.

A radioactive waste disposal heat transfer study, concerned with the determination of the maximum sphere of influence of the heat produced by a buried cylinder of radioactive waste, was started. Since the radioactivity decays at a known rate, the temperature in the cylinder will reach a maximum at a time determined by the initial heat generation rate and the thermal characteristics of the soil. The magnitude of the maximum temperature and the time of its occurrence are to be determined.

Maintenance at the Critical Mass Lab has continued to be a problem. One trouble that still remains unsolved is the inability to drop the control rod. Residual magnetism seems to prevent the rod magnet from releasing the rod when the rod drop button is pressed. Reverse magnetic current seems to have no effect. All necessary components are on order for the new control rod drive.

The split-half machine (the critical assembly for solid criticality studies) has been delivered. It is presently in the shop being repaired, as it was delivered in an inoperative condition.

An as-built program has been started on the instrumentation drawings for the existing Critical Mass Lab. A similar program will be started in the near future for the electrical drawings. Scoping of instrumentation for a Phase II expansion of the laboratory has been started.

Consulting Services on Nuclear Safety - Criticality Hazards

Meeting of Industrial Nuclear Safety Group and Standards Committees on Fissionable Material Outside Reactors

A meeting of the Industrial Nuclear Safety Group and of the Standards Committees on Fissionable Materials Outside Reactors was held at Hanford on May 9-11. Twenty-four persons attended from off-site, including the top experts in the field of nuclear safety in the United States. General subjects for discussion were: 1) Reviews of Experimental Criticality

Studies, 2) General Discussion of Nuclear Safety Guide, 3) Discussion on Draft of Compilation of Criticality Data, 4) Standards and Regulations, 5) Transportation of Fissile Materials, and 6) Theory and Special Topics.

The group toured the Critical Mass Laboratory on May 11, and E. D. Clayton was Chairman of the meeting. The next meeting will be held in the Spring of 1963, at Washington, D. C. C. D. Luke, Chief, Criticality Evaluation Branch, Division of Licensing and Regulation, USAEC, will be host for the next meeting.

Nuclear Safety in HLO

Nuclear Safety specifications F-4 and H-2 were approved and issued. F-4 covers the fabrication and storage of 2.5 w/o oralloy-thorium alloy fuel elements that will be made in the 306 Building by the Reactor and Fuels Research and Development Operation. H-2 covers the storage of fissile materials in the two wet storage basins in 327 Building by the Radiometallurgy Laboratory.

Nuclear safety specification J-3 was prepared for the Plutonium Metallurgy Operation. This specification covers the handling and storage of plutonium carbide, which is to be fabricated in Building 231-Z. The PuC will be in the C/Pu range of 0.7-2.0. On May 18, 1962, a meeting was held with personnel of the Plutonium Fabrication Operation to discuss the nuclear safety aspects of the plutonium carbide process. The potential hazards and safety requirements of each step of the process were reviewed.

A meeting was held with Plutonium Metallurgy Operation personnel on May 29, 1962, to discuss the general nuclear safety limits for dry glove boxes in Buildings 308 and 231-Z. It was concluded that the "2.8 Kg per batch - 4.5 Kg per hood" limit currently in use could be revised to a single limit of 3.5 Kg. A specification is to be prepared to cover this revision. The specification will also give the requirements necessary for a glove box to be classified as "dry".

Nuclear Safety in CPD

Participation as a member of the CPD Hazards Analysis Group continued throughout the month. This group meets regularly to evaluate the deactivation activities that are currently being carried out in the Recuplex facility of Z Plant. Twenty-one hazards evaluations were completed during May. The group also evaluated the hazards associated with the startup of

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Tasks I, II, and III.[3]

At the request of the Redox Operation, a nuclear safety review was made of Project CGC-913. Under this project, equipment in the 202-S and 233-S Buildings was rearranged to permit the accumulation and recovery of neptunium without interfering with the production of plutonium. The facility was inspected on May 8, 1962. It was concluded from the review that the facility is safe from the standpoint of criticality under the operating controls specified; comments were submitted to Redox personnel in a meeting on May 14, 1962. Calculations are now being made (at the request of Redox personnel) to determine the minimum critical plutonium concentration for each vessel in the facility. A report will be issued when these calculations are completed.

Nuclear Safety in Transportation

The shipment of several containers of slightly enriched uranium on a Mound Laboratories railroad car was approved May 31, 1962.[4] The shipment is to include: a) 5 lb. of 1.0-1.75 w/o U^{235} enriched uranium metal, b) 255 lb. of 2.6 w/o U^{235} enriched UO_2 pellets, and c) 1337 lb. of 1.47 w/o U^{235} enriched uranium fuel elements.

NEUTRON CROSS SECTION PROGRAM

Quasi-Elastic Scattering of Neutrons from Water

Several measurements are in progress which are designed to confirm or clarify the results of previous measurements of the quasi-elastic scattering of neutrons from room temperature water. Measurements have been made at an initial neutron energy of 0.15 ev with a higher resolution in neutron energy than previously used. Measurements have been completed at four degrees scattering angle for two different water sample thicknesses. Measurements are in progress at a larger scattering angle.

Two papers were accepted for presentation at the IAEA Symposium on Inelastic Scattering of Neutrons in Solids and Liquids, Chalk River, Canada, September 10-14, 1962. The papers report work done on the KE neutron spectrometer.

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(3) C. L. Brown, M. J. Stedwell, R. L. Stevenson, "Hazards Analysis of Task I, II, and III Reactivation (Supernate Handling by Load Out)", HW-73646, May 11, 1961.

(4) Letter from P. F. Gast to F. J. Zelley, "Nuclear Safety Approval for Slightly Enriched Uranium Shipment", May 31, 1962.

They are authored and titled as follows:

1) D. A. Kottwitz, B. R. Leonard, Jr., and R. B. Smith, "Quasi-Elastic Scattering by Room Temperature Light Water" and

2) D. A. Kottwitz and B. R. Leonard, Jr., "The Scattering Law for Room Temperature Light Water."

Inelastic Scattering of Neutrons from Water

Additional measurements on the scattering of slow neutrons with large energy change from room temperature water are in progress. Angular distribution measurements have been completed for the scattering of neutrons of initial energy of 0.3 ev to a final energy of 0.125 ev. Calculations have continued on multiple interaction effects in scattering samples.

Total Cross Section of Graphite

Measurements have been made of the total cross section of a sample of pyrolytic graphite over the neutron energy range from 0.008 ev to 0.3 ev. Measurements were made in two different sample orientations with the neutron beam parallel to and perpendicular to the direction of preferred orientation of the basal planes of the graphite crystals. The significant differences in total cross section which were observed occur primarily for neutron energies less than 0.028 ev. Analysis of these data has not been completed.

Fast Neutron Cross Sections

Samples of lead, sulphur, and calcium were prepared and tested for use as transmission samples for fast neutron total cross section measurements.

The phase-quadrature deflection plate assembly to suppress alternate charged particle bursts of the swept beam was installed in the Van de Graaff. Operation of the system was satisfactory but it produces a small horizontal displacement (about 0.5 mm) of the beam which can be controlled.

The preparation of a computer program, BIG NED, for the reduction and computation of fast neutron cross section data has continued. The initial portion of the program which performs a polynomial fit to the open-beam spectrum data has been successfully compiled.

Instrumentation

Construction of the 1024-channel time-of-flight analyzer continued this month. Wiring of the analyzer has presently been completed and debugging of the logic is in progress. It is expected that debugging will be completed by the first of June.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE

Zero-Dimensional Analysis of Pu-Al-H_2O Assemblies

Work on the zero-dimensional analysis of the 5 w/o Pu-Al-H_2O assemblies has been completed. An informal report, HW-73653, "Zero-Dimensional Analysis of Some Pu-Al-H_2O Assemblies", is being issued.

Heterogeneous Pu-H_2O Reactor Physics Calculations

Final results of a three-group diffusion theory analysis of heterogeneous 5.0 w/o Pu-Al rods in H_2O have been obtained. A complete tabulation of significant interim results is also given in Table I including calculations I, II, and III which were reported last month. Whereas, a ν value of 2.84 had been used in the calculations of I, II, and III, the more accepted value of 2.91 was used to obtain result IV. This perturbation increased the multiplication by a nearly constant 24 mk for all lattice spacings.

Result V reflects the results of a perturbation calculation to correct result IV for the lower ratio of epithermal to thermal fluxes near the top and bottom reflectors. Two approaches to that correction were used. The first is to deduce a change in the specific effective multiplication rate in the core due to the spectral change. The one-dimensional exact solution in the radial direction yields the three-group fluxes and adjoints and the vector multiplications $\phi^* J\phi$ and $\phi K\phi$ are performed. The quantity, J, is the three-group production matrix ($\nu\Sigma_f$) in each region and K is the three-group loss matrix (by diffusion, absorption and group transfer) in each region. ϕ^* is the three-group adjoint flux (row vector) and ϕ is the flux (column vector). Summing in the radial direction only over the core, the specific effective multiplication in the core is

$$k_{eff_1}\,(\text{core}) = \frac{\phi^* J\phi\ (\text{core})}{\phi^* K\phi\ (\text{core})}\ .$$

However, the specific multiplication deduced from,

[redacted]

[redacted]

TABLE I

SUMMARY OF RESULTS OF ANALYSIS OF 5.0 WT % PU-AL - H_2O HETEROGENEOUS

LATTICE EXPERIMENT

Lattice Spacing (in.)	Exp. Crit. No. of Cells	Calculated Effective Multiplication I	II	III	IV	V	VI	VII	Final Calculated Critical No. of Cells
0.75	356.1	0.9900	0.9922	0.9707	0.9943	1.0073	1.0110	1.0189	330.4
0.85	230.2	0.9706	0.9724	0.9559	0.9793	0.9888	0.9920	0.9945	235.1
0.90	192.0	0.9538	0.9554	0.9408	0.9638	0.9722	0.9736	0.9785	208.7
1.00	170.1	0.9596	0.9610	0.9489	0.9721	0.9792	0.9785	0.9829	182.6
1.10	166.5	0.9695	0.9705	0.9588	0.9823	0.9884	0.9852	0.9890	174.4
1.20	181.1	0.9839	0.9845	0.9759	0.9999	1.0052	0.9991	1.0033	178.8
1.30	215.5	0.9984	0.9989	0.9907	1.0150	1.0194	1.0102	1.0133	200.1
1.50	307.8	0.9811	0.9816	0.9742	0.9986	1.0019	0.9871	0.9902	337.0

I No Correction
II Bell Correction Added
III Fuel Temp. Added
IV ν Correction Added
V Axial B_3^2 Correction Added
VI Mod Temp. Correction Added
VII Axial $B^2_{1,2}$ Correction Added (Final Value)

[redacted]

$$k_{eff_2}\ (\text{core}) \;=\; k_\infty\ (\text{cell})\ \exp - \left(\tau_1 B_1^2 + \tau_2 B_2^2 + L^2\ B_3^2\right)(\text{core})$$

would yield the same result if the effective spectral ratio in the cell case (radial boundary conditions of zero current) were the same as in the core in the reactor case.

The fast and epithermal group ages and the thermal diffusion length are core-averaged values in the above equation and the values of the group radial bucklings were calculated as

$$B_i^2 \;=\; \frac{\nabla \phi_i^* \ \nabla \phi_i}{\phi_i^* \ \phi_i}$$

with summation extending only through the core.

The effect of the reflector-induced spectrum change in the core (in the radial direction) is:

$$\Delta\, k_{eff}\ (\text{radial, core}) \;=\; k_{eff_1}\ (\text{core}) - k_{eff_2}\ (\text{core})$$

If the statistical weight (relative to a core-center value) of such a spectral change is nearly the same near the ends of the cylindrical core as near the core radius, the effect near the top and bottom reflectors is approximately proportional to the amount of core-reflector interface involved so that

$$\Delta\, k_{eff}\ (\text{axial, core}) \;=\; \frac{2\pi\ R^2}{2\pi\ RH} \cdot \Delta\, k_{eff}\ (\text{radial, core})$$

where

R is the core radius

H is the core height

A second approach to this correction yields substantially the same result. This method is to deduce the proper thermal axial buckling from the thermal radial buckling as calculated by

$$\frac{\nabla \phi_i^* \ \nabla \phi_i}{\phi_i^* \ \phi_i} .$$

In the tightly packed lattices this quantity is negative. The new thermal buckling attributed to the axial direction is simply the calculated radial value multiplied by the ratio of the geometric axial and radial bucklings. Then the correction amounts to

$$\Delta k_{eff} \text{ (axial, core)} = \left[\frac{\nabla \phi^* \nabla \phi}{\phi^* \phi} \text{ (reactor)} \cdot \frac{\left(\frac{\pi}{H}\right)^2}{\left(\frac{2.4048}{R}\right)^2} - \left(\frac{\pi}{H}\right)^2 \right] L^2 \text{ (core)}$$

where ϕ^* and ϕ are thermal values.

R and H are the extrapolated values. Result V is the exact solution which expedites this second method.

Result VI is again an application of the Deutsch neutron temperature calculations. In this case, where a constant 80 C neutron temperature had been used to determine the thermal parameters for H_2O, the neutron temperature range from 140 C (0.75" lattice) to 40 C (1.5" lattice).

Result 7 has corrected fast and epithermal axial bucklings incorporated into the exact solution. These are based on extrapolation distances calculated for these two groups in the radial direction, and are slightly greater than the constant value of 9.0 cm assumed previous to the earlier results.

A graphic comparison of the computed and measured critical number of rods is shown on Figure 1.

<u>PRTR Coolant Loss Effect with All Moxtyl Loading</u>

A theoretical estimate of the multiplication change due to the complete loss of coolant in some or all PRTR coolant channels has been obtained. The reactor condition assumed was a uniform loading of green moxtyl fuel with 210 gms Pu/element and 16 w/o Pu-240. The calculation was performed using the SWAP code in a manner used before to determine the effect of coolant loss with the reactor in a spike-loaded configuration (three-zone UO_2 and L_x Pu loading)[5]. The effect of complete loss of 0.25% H_2O degraded D_2O coolant in all channels (85) was -16.9 mk. The radial distribution is shown in Figure 2.

(5) "PRTR Critical Test Results", HW-61900 BA, PRTR Physics Subcouncil, December 31, 1961.

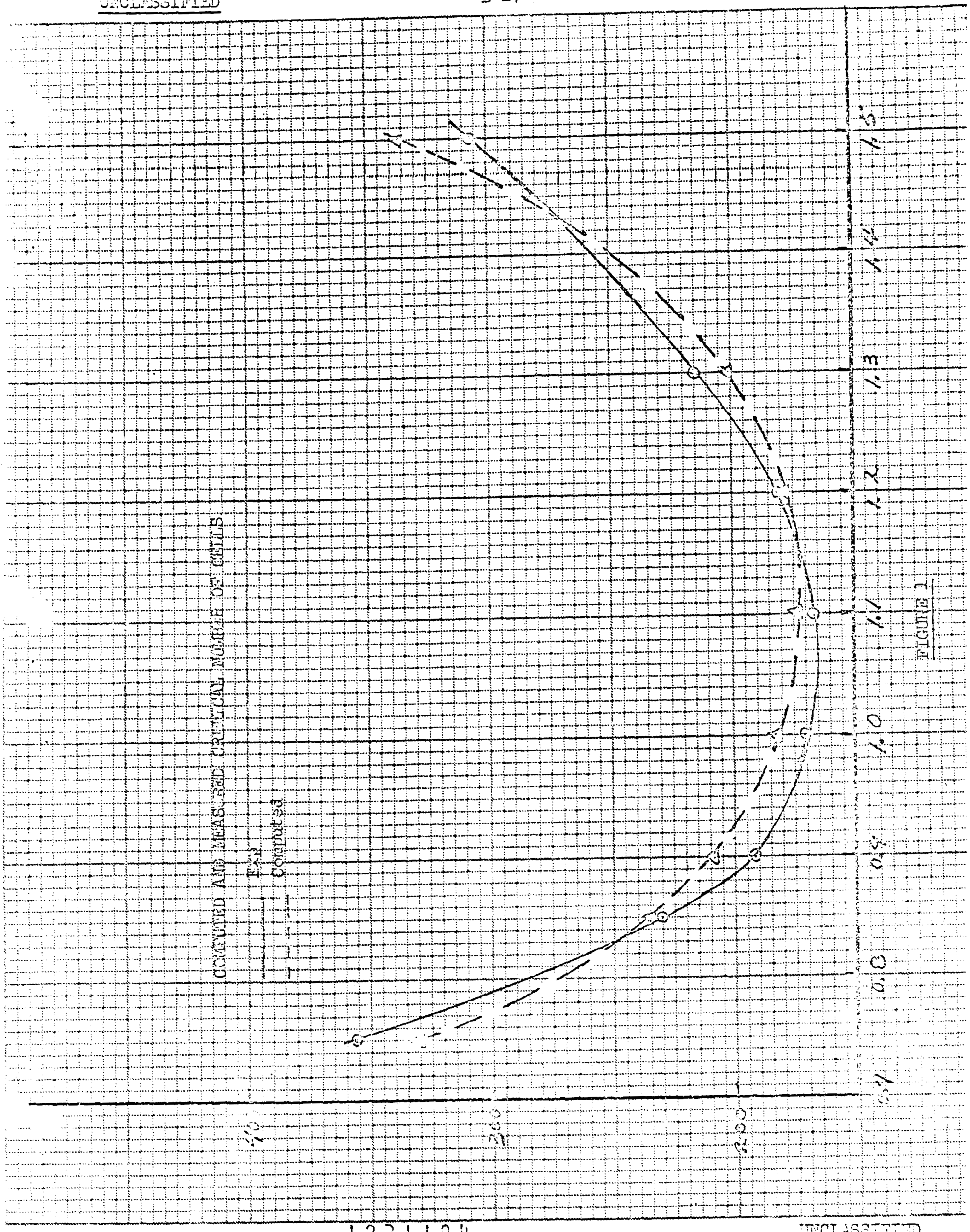
COMPUTED AND MEASURED CRITICAL NUMBER OF CELLS
Exp
Computed
0.8
0.9
1.0
1.1
1.2
1.3
1.4
1.5
200
300

FIGURE 1

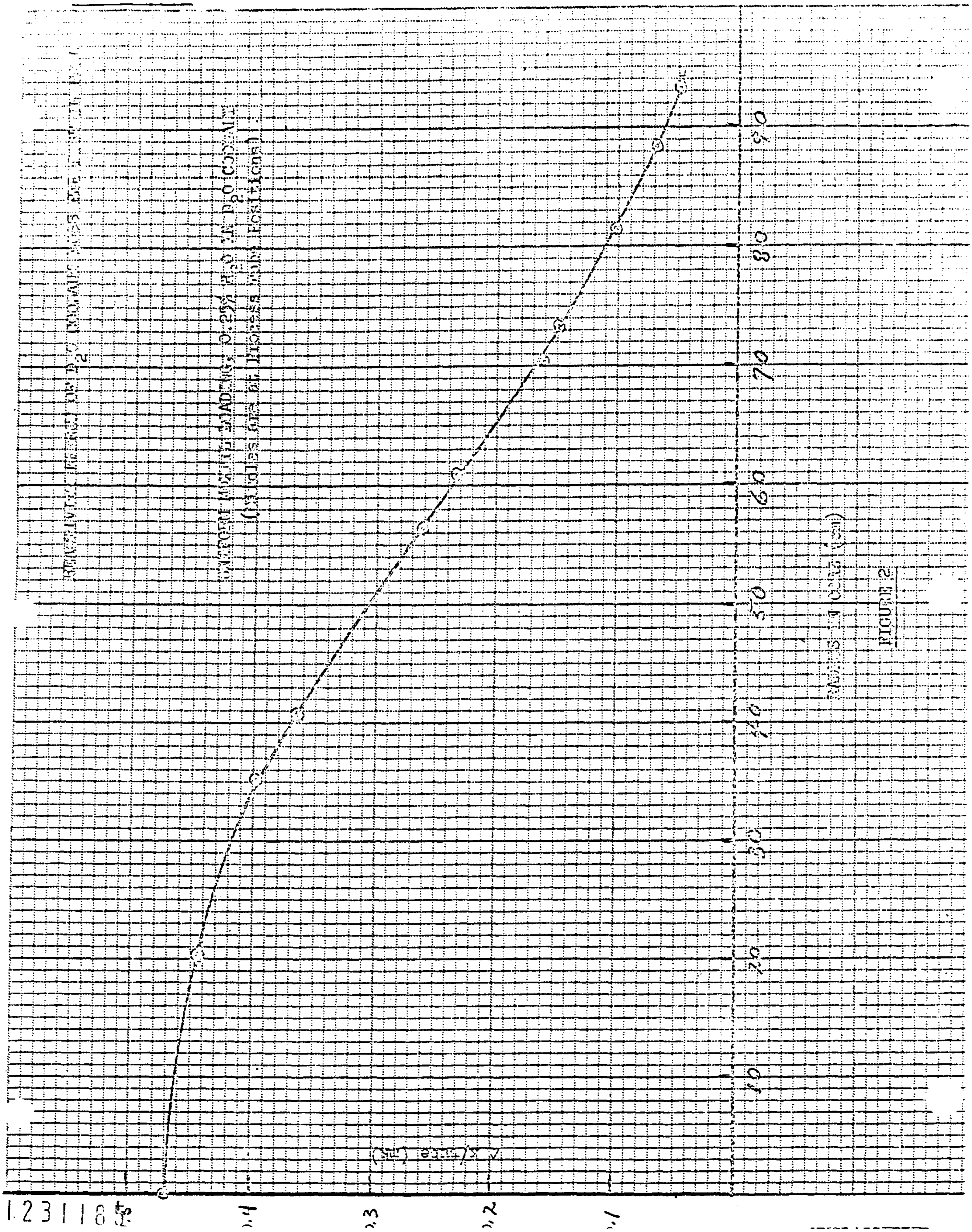

FIGURE 2

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Pu-239 Resonance Flux in L_x Pu-Al fuel Element

At the request of Reactor Lattice Physics an eleven group HFN calculation of the flux in a 19-rod low exposure Pu-Al cluster was made. The emphasis was on the flux in the 0.3 ev Pu-239 resonance to provide a means of correcting lutetium foil data for the flux dip when attempting to deduce Westcott r and T values. The 19-rod cluster was annularized by preserving volumes of each material. The 19-level slowing down diffusion program G_2 was used to obtain a set of fast group (0.6826 to 10^7 ev) constants, and the slowing down density at 0.6826 ev. Group transfer cross sections below 0.6826 ev were hand calculated assuming isotropic elastic scattering and transfers from the fast group were normalized to $\frac{q_{0.6826\ ev}}{\phi_{fast}}$. Program C-Fine was used to obtain flat flux weighted Pu-239 cross sections. All other elements were assumed to be 1/v absorbers below 0.6826 ev. A single Maxwellian thermal group at 100 F with cutoff at 0.1798 ev was assumed. The epithermal group fluxes and the Maxwellian thermal group were converted to $\phi(u)$ and are given in Table II for the three fuel regions.

The G_2 results show that $\phi(u)$ is virtually flat from 0.68 ev to 1.8 ev. Taking $\phi(.6338$ ev) as the epithermal flux in the absence of the Pu-239 resonance, the spectral r value can be obtained from the relation

$$r = \frac{1}{\frac{\phi_{th}}{\phi_{epi}} + \sqrt{\frac{16}{\mu\pi}}}$$

Taking the cutoff energy to be where the eipthermal flux equals the Maxwellian flux, μ kT = 0.18 ev and $\mu = 6.7$. The r values are computed below:

	ϕ_{th}	ϕ_{epi}	r
Inner Rod	3.381	.2493	.0693
Middle Ring	3.495	.2499	.0673
Outer Ring	3.820	.2515	.0623

Error Analysis of PRTR Data

In support of the work on the PRTR theory-experiment correlation, an analysis of the relative errors associated with the PRTR operating data was initiated. The objective of the analysis is to assign reasonable uncertainties to the recorded data and perhaps recommend more appropriate measure-

[redacted]

TABLE II

PRTR HFN FLUXES CONVERTED TO $\bar{\phi}(u)$

Pu-Al L_x

	E*(ev)	E_{lower}	$\bar{\phi}(u)$ in Center Rod Unnorm.	$\bar{\phi}(u)$ in Center Rod Norm.	$\bar{\phi}(u)$ in Middle Ring Unnorm.	$\bar{\phi}(u)$ in Middle Ring Norm.	$\bar{\phi}(u)$ in Outer Ring Unnorm.	$\bar{\phi}(u)$ in Outer Ring Norm.
Slowing Down	.6338	.5885	.2493	1.000	.2499	1.000	.2515	1.000
	.5464	.5074	.2401	.9631	.2411	.9649	.2441	.9705
	.4711	.4375	.2296	.9210	.2312	.9253	.2358	.9375
	.4062	.3772	.2050	.8223	.2084	.8340	.2183	.8680
	.3502	.3252	.1426	.5720	.1509	.6039	.1762	.7006
	.3020	.2804	.07409	.2972	.08623	.3451	.1289	.5125
	.2604	.2418	.08488	.3405	.09703	.3883	.1358	.5399
	.2245	.2085	.1233	.4946	.1325	.5303	.1592	.6330
	.1936	.1798	.1489	.5973	.1556	.6227	.1745	.6938
Maxwellian	.18		.1729	.6935	.1787	.7151	.1953	.7765
	.15		.3678	1.475	.3802	1.522	.4156	1.652
	.10		1.128	4.525	1.165	4.662	1.274	5.065
	.05		1.822	7.308	1.883	7.536	2.058	8.183

* E is the geometric mean of E_{upper} and E_{lower} for the slowing down region.

[redacted]

ments to facilitate the correlation.

PRTR-Banked Shim Rod Test

A test was performed in the PRTR, whereby the shim rods were all held at the same vertical height and the corresponding flows and differential temperatures recorded. These data were processed by the PRTR Data Processing Program(6) which computed the appropriate "Tube Factors".

An effort is currently under way to theoretically compute these experimental tube factors. At present, a model to describe the various cells (UO_2, Pu-Al, and moxtyl) is being derived. From these models, the appropriate constants will be obtained for use in a two-dimensional analysis(7). Presumably, this two-dimensional analysis will yield the power distributions matching the experimental values.

Advanced Concepts - Space Applications

A set of 16-group cross sections were obtained for two isotopes of rubidium (Rb-85 and Rb-87) from the RBU basic library. The cross sections were weighted by a spectrum computed by Yiftah, et al.,(8) and inserted into a library tape for use with HFN. Computation of reactor systems using rubidium coolant, tantalum structure, and PuO_2, UO_2, and PuN as fuels is now proceeding. A 10 cm beryllium reflector is assumed in all cases.

In addition to rubidium, nitrogen, beryllium, U-234 and U-236 have been added to the HFN cross section tape.

Lattice Parameters for Low Exposure Pu-Al Fuel

The attempt to analyze plutonium-aluminum fueled lattices with Program S(9) has continued. Several runs were made during the month as input

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(6) Fishbaugher, J. R., "PRTR Data Processing Program", HW-64631, April 5, 1960.

(7) Stone, S. P., "9-Angie - A Two-Dimensional, Multi-Group, Neutron Diffusion-Theory Reactor Code for the IBM 709 or 7090", UCRL-6076, October 28, 1960.

(8) Yiftah, Okrent, and Moldauer, "Fast Reactor Cross Sections", Pergamon Press, 1960.

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

inconsistencies were found and removed. The most recent run reproduces an experimental copper activity traverse through a cell to within 1% but the calculated value of k_∞ for the poisoned cell is 0.85 rather than unity. The reason for this is unknown but is probably due to remaining errors in cross section data.

Effective Resonance Integral of Pu^{240}

Work has continued during the month on the interpretation of the experiment to determine the effective resonance integral of Pu^{240} relative to the dilute resonance integral. The diluent for the plutonium is aluminum. This experiment has been designed to provide general resonance integral information applicable to Pu-Al fueled reactors.

Progress has been made in the determination of the sensitivity of the reactor to the addition of dilute Pu^{240}. An approximate correction for self shielding needs to be made on several points for which the self shielding is small.

The work of Dresner[10] is applicable to this problem. His treatment takes into account the scattering in the aluminum diluent, and includes Doppler broadening. Nordheim's analysis[11] also takes these effects into account, but Dresner's work can be applied without the use of machine calculations.

For each of the i sets of Pu-Al rods that differ only in Pu^{240} content, the difference in reactivities measured in the PCTR between the rods in a set is given by

$$\Delta\rho_i = c \left[(\Delta N)^{40} I_a^{40} + N_b^{40} (\Delta I)^{40} \right]_i$$

The subscript "a" refers to the rod with the smaller concentration of Pu^{240}, and the subscript "b" refers to the rod with the greater concentration of Pu^{240}. If the Pu^{240} in both rods were perfectly dilute, the second term would vanish, and the constant "c" could be evaluated directly from the experimental data. For sets of rods which are nearly

(10) Dresner, L. Resonance Absorption in Nuclear Reactors. New York: Pergammon Press, 1960.

(11) Nordheim, L. W. "A New Calculation of Resonance Integrals", Nuc. Sci. and Eng., p. 2, 457-463, 1962.

dilute, the second term is small, and is being evaluated from the calculations discussed above.

Preliminary analysis indicates that the effective resonance integral of Pu^{240} for the rod with the highest concentration of Pu^{240} is reduced by a factor of approximately five from the dilute integral.

Plutonium Recycle Critical Facility

The final drafts of the PRCF Startup Procedures are being prepared. Approximately 16% of these procedures have been completed. Additional work is being done in preparation for Pressure and Temperature Coefficient Experiments and Reflector Experiments.

Three physics orientation meetings have been held for PRCFO personnel. The purposes and physics background for the startup program have been covered in these meetings.

The pulse-height distribution of BF_3 counters which are available have been measured with the 256 channel analyzer. The standard pile was used as a source of neutrons for the measurement. The purpose of the measurements is to determine which counters will be satisfactory for use during PRCF start-up.

A report of invention HWIR-1519 which describes the thickness gauge that was used to measure the thickness of cadmium on the PRCF safety rods has been written and forwarded to the AEC for evaluation.

Measurements on Phoenix Fuel Standards

Because of the difficulty of accurately determining the boron content of the poison standards for the MTR/ARMF Phoenix Experiment by chemical analysis, it has been proposed that the boron and plutonium content be determined from reactivity measurements in the PCTR. Two cylindrical water jackets have been designed and are now being fabricated. These jackets will surround the samples and will allow the relative worths of thermal and fast neutrons in the PCTR to be varied. The Pu and B content can be found from comparisons with Cu and U^{235} standards.

Approach to Critical Measurements with PuAl-H_2O Systems

Experiments to determine the nuclear parameters for 1.8 w/o Pu-Al fuel in light water were conducted in the tank in the TTR reactor room. All of the approach to critical experiments and exponential experiments have been completed. During the month the exponential experiments for the 0.75 and

0.80 inch lattice have been completed. In addition, the approach to critical for the 0.75 inch lattice was repeated in order to check an inconsistency in the data. The results are 563 ± 2 and 510 ± 2 rods for lattice spacings of 0.75 and 0.80 inches, respectively. This corresponds to a critical mass of 3.63 and 4.10 kg of Pu. Additional experiments which will employ zoned loadings of 5 and 6 w/o Pu^{240} are being planned for a lattice spacing of 0.85 inches. The lattice structure has been assembled for these experiments.

PRTR Fuel Irradiation Experiments

Bare and cadmium covered lutetium foils were attached longitudinally on two unirradiated 1.8 w/o Pu-Al fuel elements. Each of the fuel elements was irradiated separately in PRTR tube 1556. Each irradiation lasted 30 minutes at 50 kw with full pressure and coolant flow. The first element could not be removed immediately from the reactor because of PRTR instrument difficulties; however, it is believed that the delay did not adversely affect the experiment. Some of the foils were too radioactive to count with the foils lying in the calibrated position on the NaI(Tl) crystal. It was concluded from measurements with a 256 channel analyzer that much of the activity was coming from the 6.8 d Lu^{177}. The counts from the 3.7 hr Lu^{176m} activity would be lost if the foils were not counted until the total activity had decayed so that the foil could be counted on the crystal. Thus, several arrangements were established so that the foils could be counted at fixed distances from the crystal and data were obtained to allow normalizations between the various distances. All of the foils have been counted and the analysis of the data has begun.

The development of computational methods related to this work is reported under "Code Development".

Neutron Rethermalization

A trip report on the BNL Conference on Neutron Thermalization was prepared. This conference produced a significant set of papers on neutron thermalization that are available upon request.

Code Development

Program to Correct Sensor Activity

A program to compute the time evolution of the radioactive daughter atom density when a stable parent atom is exposed to an arbitrary flux history has been written and checked out. The objective is to provide a means of correcting observed sensor activity to give accurate flux-times, for instance, the activities obtained from the monitors on the physics test elements in PRTR. Another application might be the calculation of radiation history dependent parameters in lutetium foil irradiations.

The program assumes that the exposure history can be described by alternate periods of constant power and zero power; however, by taking small enough time steps it is possible to approximate a continuously varying exposure. A total of 1425 up-down cycles is provided. The program computes:

$$N(T+t_n) = \frac{N(T)\, \sigma_1\, \phi\, e^{-\sigma_1 \phi t}}{\phi\, \sigma_2 + \lambda - \phi\, \sigma_1} \left[1 - e^{-(\phi\, \sigma_2 + \lambda - \phi\, \sigma_1) t_n} \right]$$

or

$$N(T+t_n) = N(T)\, e^{-\lambda t_n}$$

depending upon whether t_n is an exposure period or a decay period. The values of flux are given in an input table, where as many as 100 flux values and a corresponding reference power level are given. The flux values are sealed to the appropriate powers for each exposure period. The resulting corrections to flux-time for each flux level provide a means of obtaining the integrated flux-time for the exposure.

RBU

The revision to the Monte Carlo collision routine modifying the treatment of collisions with bound scattering centers was inserted and partially debugged. The first results predict an effective neutron temperature of about 0.015 ev, and a flux peak-to-tail ratio measured at 0.5 ev of about 265. These results do not compare favorably with SPECTRUM results of effective neutron temperature 0.027 ev, and a peak-to-tail ratio of about 200.

Work is now under way to determine exactly why the spectrum does not approximate the spectrum previously generated using a gaseous hydrogen moderator. The leakage estimate seems to be in error as well. The internally calculated leakage expectation value differs by approximately a factor of two from the directly (Monte Carlo) calculated particle leakage from the system. The cause of this discrepancy is unknown, and it is assumed that the new model is not yet performing correctly.

CALX

SIGMA-3, the program which prepares cross sections for use in the CALX multi-group burnup code, has been coded and is being debugged. TEMPEST, the thermal neutron spectrum code, and GAM, the epithermal neutron spectrum code, were successfully run as a single package, and generated the "basic library" from which SIGMA-3 will construct a CALX data tape. A "merging library" which enables SIGMA-3 to relate CALX, GAM, and TEMPEST notation, is being recorded on data sheets. Until the full merging library, covering 39 fuel and 120 non-fuel isotopes, is complete, a dummy library will be used for debug runs.

Preliminary runs on CALX indicate satisfactory performance. Further debugging awaits successful data tape generation.

Non-Linear Fuel-Cycle Functions

Optimization work on the non-linear aspects of nuclear fuel-cycle analysis progressed through formulation, FORTRAN programming, and check-out of a versatile complex-field subroutine for generating, differentiating, and integrating the entire family of doubly-periodic elliptic functions. The work was motivated by the previously reported observation(12) that elliptic functions satisfy first-order differential equations having non-linear structure resembling that of the isotope balance equations and the neutron energy-spectrum balance equations for an infinite homogeneous medium.

The logic of the subroutine revolves about systematic use of four of the elliptic generating functions:

(1) The Weierstrass sigma-function(13), defined as having a doubly-periodic lattice of simple zeros in the complex plane, with an essential singularity at infinity, and computed directly from this definition as the uniformly-convergent doubly-infinite product

$$\sigma(z, \omega_1, \omega_2) = z \prod_{m,n}' \left[1 - z/(2\, m\, \omega_1 + 2\, n\, \omega_2)\right]$$

$$\cdot \exp\left[z/(2\, m\, \omega_1 + 2\, n\, \omega_2) + z^2/2\,(2\, m\, \omega_1 + 2\, n\, \omega_2)^2\right]; \qquad (1)$$

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(12) PIRDO Monthly Report, HW-72902 B, p. B-13, February, 1962.

(13) Neville, E. H., Jacobian Elliptic Functions, Clarendon Press, 2nd Edition (1951), Eqs. 0.52, 0.53, 0.61, 0.65.

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(2) The Weierstrass zeta-function[13], defined as having a doubly-periodic array of simple poles, and computed from this definition as the uniformly-convergent doubly-infinite sum

$$\zeta\,(z,\ \omega_1,\ \omega_2)\ =\ 1/z + \sum_{m,n}{}' \Big[1/(z-2\ m\ \omega_1-2\ n\ \omega_2) + 1/(2\ m\ \omega_1 + 2\ n\ \omega_2) + z/(2\ m\ \omega_1 + 2\ n\ \omega_2)^2 \Big]; \qquad (2)$$

(3) The Weierstrass "p"-function[13], a doubly-periodic lattice of second-order poles, computed by the uniformly-convergent double sum

$$p(z,\ \omega_1,\ \omega_2)\ =\ 1/z^2 + \sum_{m,n}{}' \Big[1/(z-2\ m\ \omega_1 - 2\ n\ \omega_2)^2 - 1/(2\ m\ \omega_1 + 2\ n\ \omega_2)^2 \Big];\ \text{and} \qquad (3)$$

(4) The Neville N-th order zeta-function[13], a doubly-periodic array of N-th order poles computed by the uniformly-convergent double sum

$$\zeta_N(z,\ \omega_1,\ \omega_2)\ =\ \sum_{m,n} \Big[1/(z-2\ m\ \omega_1 - 2\ n\ \omega_2)^N \Big]. \qquad (4)$$

In all these expressions, the argument z and the dual half-periods ω_1 and ω_2 are complex numbers, the product-or-sum indices (m,n) run through all positive and negative integers, and a primed product-or-sum symbol signifies omission of the singular (m=0, n=0) term.

Elliptic functions having any specified singularity structure are obtainable quite clearly from these generating functions. Products and ratios of the multiple-zero sigma-functions may be used to obtain any specified doubly-periodic pattern of zeros and poles. Weighted sums of the multiple-pole zeta-functions may be used to obtain any desired distribution of poles and residues.

These elliptic generating functions form a sequence of successive derivatives, having clean and powerful doubly-periodic orthogonality properties, which essentially transform the intricate properties of elliptic functions into the simple formalism of elementary partial fractions.

The logarithmic derivative of the first function gives the second function. Each successive function thereafter is the renormalized derivative of the preceding function. Correspondingly, the subroutine evaluates derivatives and integrals of elliptic functions simply by shifting the order N of the generating function.

Mass Spectrometry

Six more plutonium samples from PRTR fuel element No. 5075 were isotopically analyzed. Alternate analyses were made on a single plutonium sample to serve as a standard for calibration purposes and quality control. The standards results indicate that the systematic bias error in the measured mass ratios was only 0.3 percent as compared to a bias of 2.0 percent encountered in April. In addition, the external precision (reproducibility of repeated analyses) agreed with the internal precision of a single analysis.

Instrumentation and Systems Studies

One experimental transistorized scintillation PRTR Liquid Effluent Gamma Monitor has satisfactorily completed laboratory operational and drift tests and is ready for installation at PRTR as soon as PRTR personnel install the cables and prepare the probe and instrument locations. A second generation [redacted]

NEUTRON FLUX MONITORS

Continued investigations and more computer studies have shown that three radionuclides, U^{234}, U^{238}, and Pu^{240}, are of interest for use in regenerating "Phoenix" neutron in-core detectors. The absorption cross sections of the three are sensitive to neutron temperature and to spectral hardness; however, the cross sections are of the order of magnitude of 10, 100, and 1000 barns, respectively, so an evaluation of them should encompass the field. Burnout of the fertile material limits the useful lifetime of the detectors, and since burnout is proportional to $e^{-\phi\sigma_a t}$, either high fluxes or large absorption cross sections will accelerate burnout. Detector sensitivity or fission rate is proportional to $\phi\sigma_f N_f$ where N_f is the number of fissile nuclei, with fission cross section σ_f, in the detector. In a regenerating detector, N_f is related to the initial fertile nuclei content; therefore, detector sensitivity in a given flux can be established by the number of fertile nuclei used in the detector with the limit of detector size possible for the specific application.

From the foregoing, it can be concluded qualitatively that the best fertile material for a given detector is one with the smallest cross section which will provide adequate sensitivity in the space available. Further detailed calculations are necessary to establish solution criteria. An initial attempt to utilize the neutron flux monitor detector program, as written by Operations Research and Synthesis personnel, was unsuccessful because of recent modifications to the computer system. The program was revised so detailed calculations can be made to investigate the stated nuclides to establish detector selection criteria as a function of reactor in-core flux environment.

The initial feasibility study report of regenerating in-core neutron flux monitors was rewritten, in part, to permit release as an unclassified report.

Work with the microwave equipment was confined to efforts to reduce the noise level and to increase the sensitivity.

HIGH TEMPERATURE LATTICE TEST REACTOR

A document describing the HTLTR was completed and submitted for declassification (HW-73693).

[redacted] [redacted]

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

The multiparameter eddy current testing equipment is being modified to permit its use and evaluation on metal test specimens. Assistance was given to Physical Testing by developing a small internal test probe and associated eddy current test equipment for testing 1/4-inch Inconel tubing installed in the NPR project. This tubing was encased within a concrete structure and was accessible only through the bore.

Final circuit values and system configuration for the multiparameter eddy current test equipment were recorded. A series of waveform photographs indicating signal flow through the equipment were taken. Steps initial to the next phase of development were taken. Four eddy current test specimens are now on order from the machine shops. Each is to be fabricated from 1-3/8-inch I.D. Zircaloy-2 tubing with a 0.030-inch wall thickness, and will be 18" in length. Two types of manufactured defects will be used, axial electro-machined notches and changes in wall thickness. Two samples will be made for each type of defect, one with the defect starting at the inside of the tube and the other with the defect starting at the outside. The set of notches to be used will consist of notch depths of 5, 10, 20, 25, and 30 mils. Each notch is to be about 3/8 inch long by 2.5 mils wide. For the wall thickness variation, the tube wall will be reduced from 30 to 15 mils in three five-mil steps by machining.

Brass forms to permit accurate construction of test probe coils on recessed 1/4-inch and 1/2-inch General Ceramics Ferramic type cores were fabricated. These were used to build two each of three types of test-probe heads. Each head consists of both a drive and a pickup coil wound on the Ferramic cores just mentioned and set in a phenolic mount. All coil leads come to terminals and are potted in plastic for mechanical protection.

Preliminary eddy current test measurements were made on 1/4-inch Inconel tubing with nominal wall thickness of 0.049 inch to obtain data on which to base the design of a multiparameter test for separating test indications from cracks near the inner surface and those near the outer surface. Tests were made with samples having some electromachined notches opening to the inner surface and others with electromachined notches opening to the outer surface. Measurements were made with an internal probe operating at 500, 800, and 1200 kilocycles.

Assistance was given to Physical Testing in providing eddy current test equipment for use in testing installed Inconel tubing. Available commercial testing equipment had insufficient sensitivity for this test; thus, an instrument and associated test probe were developed for this application. The instrument was developed, checked out in the laboratory, and placed in service in the field all within a period of about one month. The instrument operates at approximately 500 kilocycles and uses a test probe 0.140 inch in diameter mounted at the end of a 30-foot section of 0.045-inch diameter cable. The standard used in the field test has a notch in the inside wall estimated to be 0.010 inch deep, and 1/8 inch long, and an O.D. notch estimated to be 0.035 mils deep, from the outside and 1/8 inch long. Both of these notches give good signal-to-noise ratio compared to signals obtained from an average normal tube. A second instrument was fabricated and will be used for evaluation tests using external encircling coils for testing 1/4-inch diameter Inconel tubing.

Heat Transfer Testing

A dual radiometer for use in minimizing the effects of differences in surface emissivity has been tested. Heat from the plasma arc jet caused cracking of the radiometer lenses. Masks were installed to alleviate this problem, and the cracked As_2O_3 lenses replaced with a coated germanium and a coated silicon lens available on loan from the optical shop. Subsequent tests showed a lack of agreement in signals from the two radiometers. This lack of agreement is believed due either to differences in the lenses, or differences in the angle that the two radiometer optical axes make with the fuel element surfaces. Replacement lenses have been ordered, and tests of the dependence of radiometer signals on the angle made with various fuel element surfaces are continuing.

Transformers were installed between the pre and post amplifier systems of the radiometers to break up electrical ground loops. Such ground loops cause severe problems when photoelectromagnetic infrared detectors are used because of the low detector output voltages. It was also necessary to reduce time constants in the amplifiers to reduce recovery time due to transients generated by the high frequency arc-starting unit in the plasma arc jet.

Zirconium Hydride Detection

Attenuation measurements were made on seven Zircaloy-2 samples having the same metallurgical history, and having hydride concentrations ranging from 0 to 600 ppm ± 3%. These measurements were made at 15 Mc, with salol as a coupling material. Due to coupling uncertainties, several different mountings were made on each sample. The smallest value of attenuation was recorded for each sample, since deviations from perfection in mounting increase the attenuation. These measurements indicate that an increase in hydride concentration increases the ultrasonic attenuation. This increase, however, is too small to be significant because any variation in the surface finish, or metallurgical history of the sample, could possibly give a greater change in attenuation. This change in attenuation indicates, however, that the internal friction of Zircaloy-2 increases the hydride concentration.

A mounting jig was fabricated which permits salol to be used as a couplant for attenuation measurements made with variable stress. Preliminary measurements indicate that dynamic stress increases attenuation in hydrided Zircaloy-2 more than static stress. However, the many variables involved make it difficult to ascribe this increase only to effects within the metal.

A selective literature search on the Hall Effect for Group IV and V metals was completed by the library reference staff. Letters to research labs and information centers confirmed our opinion that such a comprehensive compilation does not now exist. Editing of the data will be a major task. Work on the Hall Effect hydride detection probe was temporarily suspended in favor of more pressing problems.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

The feasibility of using drilled holes as primary calibration standards for the ultrasonic defect test in Zircaloy tubing is being determined. Ultrasonic response measurements are in progress on two sets of drilled holes, one set with varying diameters, the other with varying depths. Drilled holes offer advantages over machined notches in simplicity of fabrication and in reproducibility.

Work continued towards establishing tentative specifications for transducers for sheath tube testing. Preparations were completed for measurements on drilled hole standards for comparison with measurements of reflection from ball bearings and other evaluation methods. The carrier frequency of an UR 600 Reflectoscope system was measured when used with each of twelve representative 10 Mc Li_2SO_4 crystals. The system carrier frequency varied with transducers from 9.2 to 9.8 Mc. This demonstrates one of the variables that may have to be controlled for a successful test, particularly for Lamb-wave testing.

The development of alignment and calibration procedures for the electronic instrumentation continued. A new faster method for adjusting the center frequency and bandwidth of the Immerscope agreed within 10% with the method previously used. Some side effects due to amplitude distortion are being further examined.

Experimental verification of theoretical Lamb-wave curves continued. The velocity of ultrasound in water for our conditions was measured to be 0.57 x 10^5 inches per second rather than 0.585 x 10^5, the value previously assumed. The new value provides better agreement of experimental data with theory.

The longitudinal and shear velocities of ultrasound in Zircaloy are also important in Lamb-wave theory. Using the Schlieren system to observe mode propagation, the shear velocity was found to be 0.92 x 10^5 inches per second, in exact agreement with the theoretically assumed value. Measurement of the longitudinal velocity, V_L, was also attempted. One would expect ultrasound to propagate down a thin plate with the least attenuation when the velocity is very near or exactly at V_L. Under those conditions, the motions of particles at the plate surfaces are predominantly parallel to the surface and minimum leakage of energy surrounding water would occur. Conversely, leakage at the edge of the thin plate would be expected to pass through a maximum when the Lamb-wave phase velocity, V, is equal to V_L. The Schlieren system was used to observe that generation of the fifth symmetric Lamb-wave mode gave maxima in edge leakage at fd products of 4.43 and 4.51 x 10^5 inch cycles per second and a sharp decrease in leakage at an fd product of 4.47 x 10^5 inch cycles per second. If one sets $V = V_L$ in the Lamb-wave equations, the result is not solvable. It is postulated that propagation of modes with $V = V_L$ may be physically impossible also; however, the sharp decrease in edge leakage has not been observed with the fifth asymmetrical mode. Assuming that the decrease observed with the symmetric mode does define the condition for V_L, one can calculate that $V_L = 1.8 \times 10^5$ inches per second, in good agreement with the theoretical value of 1.82 x 10^5.

Good agreement with theory has been obtained for experimental data on the ultrasonic responses from notches ranging from 0.0014 to 0.0175 inch in depth. A number of Schlieren photographs were also made to aid an attempt to more fully describe the manner in which surface particle vibrations affect the ultrasonic response. This may help clarify some of the more subtle requirements for Lamb-wave propagation.

PHYSICAL RESEARCH - 05 PROGRAM

Mechanism of Graphite Damage

An improved apparatus for thermal conductivity measurements is about finished by the shop. Work is continuing on adding an ion pump to the electron accelerator.

A decision has been reached to terminate this study at the end of FY-1962. Preparation of a terminal report is under way.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Investigations continued into the variability of wind direction observed at the meteorology tower as related to horizontal spread of a continuous plume from a point source. Last month, results were reported for a diffusion model which assumed that the wind direction observed near the source was propagated unchanged over the entire sampling grid during the period of release and sampling. This assumption is equivalent to requiring that the Eulerian autocorrelation of wind direction fluctuations at the source be identical to the Lagrangian correlation following the motion. Other investigators have suggested that the fluid-attached Lagrangian time scale be identified with the fixed-point Eulerian time scale by a simple proportionality, β, ranging in value from one to ten. The power spectrum of one to sixteen cycles per hour, derived for one moderately stable experiment, showed that 90 percent of the total variation of wind direction was within this range, representing "meander" frequencies. Using the calculated spectral density with the observed crosswind plume width, σ_Y, the Eulerian-Lagrangian proportionality factor, β, was found to be unity for the experiment. An IBM 7090 program was developed for spectral analysis of wind velocity fluctuations and β-computations to facilitate evaluation of additional experiments.

In Air Force supported programs, diffusion data obtained at Cape Canaveral, Florida, were processed and transmitted to the Geophysics Research Directorate. Equipment from the Canaveral site was received at Hanford for use in further studies here. Data from Vandenberg series II were processed and editing started. The Vandenberg series III experiments started on May 28 and will terminate June 30.

In the search for suitable fluorescent pigments for use in multicolor tracer techniques for atmospheric dispersion studies, the original list of twenty-five potential fluorescent or phosphorescent powders was reduced to five. Selection of these five was on the basis of compatibility with our current tracer, zinc sulfide #2210. The phosphorescent pigments generally have a slightly larger particle size distribution than the fluorescent pigments, permitting possible studies of the relation between size distribution and dispersion.

HW-71400, Volume I, titled, "The Green Glow Diffusion Program," was distributed during the month. The first volume includes descriptions of the field site, forecasting techniques, diffusion-measuring methods, meteorological equipment, and operating procedures during the experiments. The second volume contains tabulations of the diffusion data and the meteorological data collected during the program and should be ready for distribution in July. Results of data analyses will appear in Journal articles as they are completed.

Dosimetry

Preparations for measurement of radioactivity in Alaskan eskimos were completed. A new shadow shield was assembled. It was found to be contaminated with Zr-Nb-95. The contamination was located in the surface of the iron mounting plate and removed by grinding off the surface of the plate. The beds in the shields were provided with motor-driven, screw drives. The shadow shield system was calibrated for Cs-137. A system was devised for using a standard 500 cm^3 plastic carton for counting samples of food and vegetation in the shadow shield counter. This system was calibrated with Ce-141, Cr-51, Cs-137, Zr-Nb-95, Mn-54, and Zn-65. A portable motor-generator was obtained and proved satisfactory for powering the whole system. All other supplies that are expected to be needed were obtained. The lead and other heavy items were shipped by truck and barge to Fairbanks. The instruments were shipped by air to Fairbanks. All material must go by air from Fairbanks to Kotzebue, the first working location.

While calibrating the shadow shield for Cs-137, the calibration of the whole body counter was checked. It was within the errors of measurement of its old calibration.

An experiment was conducted that confirmed our ability to measure amounts of P-32 in the body in the range 50 to 150 nc. The counting of a subject on a weekly diet of white fish is continuing.

One of the men involved in the Recuplex accident was examined for P-32 ten days after the accident and then several more times. The P-32 was easily detected. The counting rate due to P-32 decreased exponentially with a 14.5 day half-life (i.e., the radioactive decay half-life) rather than the 8 to 10 day half-life observed for subjects who received P-32 intravenously. This indicated that most of the P-32 being observed was formed in the relatively tightly bound phosphorus, probably that in the skull, rather than the more mobile portion in which the intravenously injected P-32 appears. Thus the calibration of the counter, which was done with intravenously injected subjects, was not applicable; if applied anyway the calibration would have indicated two to three times as much P-32 as predicted from the activity of the Na-24 that had been present. The other two men who might have been counted could not be because of the interference from Au-198 in fillings in their teeth.

Radiation Protection Operation was assisted in installing a new multi-channel analyzer for the whole body counter.

The positive ion accelerator operated satisfactorily during the month.

The neutron detection instruments that were set up near Recuplex were taken down and restored to normal use. Several investigations related to the Recuplex accident and involving activation studies in the large moderator were carried out. These required setting up scintillation and Geiger counters and calibrating the thermal flux in the moderator; this work had been planned for much later this year. With the help of Radiological Chemistry, the thermal flux was measured by measuring the activation produced in sodium; copper foils were used as flux monitors. They then performed activation studies of various objects that had been on the persons of the people at Recuplex. We first tried to evaluate the gamma-ray dose recorded by the sensitive film in the film badges by activating the silver in the developed film; these films were too dark to be read by an optical densitometer. This did not provide a useful result. The measured activities were right at the first maximum of the activity vs. gamma-ray dose calibration curve and could not be interpreted with any worthwhile accuracy. Next we prepared insensitive films with different combinations of thermal neutron and gamma ray exposures. By matching these with the personnel film, it was possible to make estimates of the slow neutron flux to which the people were exposed. These estimates agreed reasonably well with those from activation of the objects the people were carrying.

Four precision long counters were compared using neutrons from the T (p,n) reaction. An examination of the data for six such counters that we have tested shows that they have responses within ±0.7% of each other. Work continued, but still without success, on trying to find a better characterization of the scattered neutron flux in our laboratory.

The plutonium source that we had measured calorimetrically was also measured calorimetrically at Mound Laboratory during a visit there by one of our physicists. They also measured more heat output than calculations based on the plutonium analysis indicated should be there. They report that this has been a common finding in their work, and that, because of the confirmation provided by our work, they intend to make a special study to uncover the cause of the discrepancy. There was a disturbing 2 to 3% difference between the results found by us and by Mound. We are exploring the possibility of errors in temperature mreasurement due to heat flow patterns in the calorimeter.

Radiation Instruments

A method was developed for providing manual keyboard entry of data into the printer and tape punch of the new Whole Body Counting Facility 400-channel analyzer. Components for the manual entry device are being procured.

Two off-site fabricated, modified pencil dosimeters of the illuminated fiber and CdS light sensor type were received and are ready to be tested with one type of audible signalling dose meter. Following a suggestion, one self-reading type dosimter was disassembled to accurately measure fiber diameter, length, and movement. With these data, the expected change in capacitance with fiber movement can possibly be calculated. If this capacitance change is large enough, it may be possible to vary the frequency of an oscillator in proportion to fiber movement. An invention report is being prepared on this idea.

Extensive testing of the automatic recharging type of dosimeter, as used in one type of experimental pocket dose meter, indicates the dosimeter with an Aquadag-coated center quartz rod and stainless steel fiber will operate correctly to an indicated exposure dose in excess of 1000 r. Circuitry development for this type of dose meter is complete.

Two circuits are available with one using an amplifier and miniature register and the second employing five binaries and an alarm circuit to provide readout and selectable audio alarming at 10, 20, 40, 80, and 160 mr. Fifty automatic recharging dosimeters, which are being fabricated off-site, are to be delivered in June. Four complete dose meters will then be assembled.

Methods of detecting airborne zinc sulfide particles for use in atmospheric studies was continued with emphasis on dual pigments. It was determined that several pigments can be discriminated with the use of proper detectors and pulse height analysis circuitry; however, some of the pigments may be unacceptable for use in diffusion studies. A second portable experimental model of the single pigment "in-field" detection method was fabricated and partly tested. This field unit employs a solid state dc amplifier and a recorder along with the activation light and phototube detector. Mockup laboratory tests were successful.

The developed low voltage dc to dc converter and a rechargeable nickel-cadmium battery were installed in an experimental portable transistorized G.M. monitor which has both a count-rate meter and audible output. Following successful initial tests, the instrument was started on a scheduled program of complete evaluation testing.

Design was started on a special miniature G.M. detector and transistorized circuit instrument which will have either a flashing light or audible indication of increasing gamma field intensity. The detector, which will be sealed in thin aluminum tubing, will be immersed in a Biology Operation fish tank. Several of a number of fish in the tank will have small Co^{60} wires implanted in them; thus, as the suitably-marked carrier fish swim by the detector, the signal output frequency rate will markedly increase.

Accelerated fabrication continued on one prototype six-decade, 1 mr to 1000 r, scintillation, transistorized, logarithmic-response, area radiation monitor. All printed circuit boards, including four electronic alarm-trip circuits, have been completed and the chassis has been prepared for parts installation. In addition, several circuits were completed for one second-generation, prototype, combination logarithmic and linear response, four-decade area radiation monitor.

Four successful transistorized, selectable-level, alarm-trip circuits were developed and tested during the month. Two circuits were specifically designed for use in line-operated instruments and the other two are useful for either portable or line-operated instrument use. All circuits can drive either audible resonant air column oscillator-type alarms or relays to be used for heavy duty light or audible klaxon-type alarms. Of the two line-operated types, one employs a pulse amplifier and integrating circuit used to control a reed relay and the second employs a solid state operational amplifier to control a relay. The two trip circuits for either portable or line-operated instrument use are of the solid-state long-tailed pair differential amplifier type and were designed for an input signal voltage range of 0 to 0.5 VDC. One circuit requires two batteries or power supplies for plus and minus voltages and the second uses a constantly-energized Unijunction audio oscillator and diode pump circuit to provide the necessary minus voltage. Typical power supply voltage levels are 4 to 10 VDC with current requirements of 1 to 6 milliamperes. Temperature tests of the dual-battery type indicated alarm trip-point stability to be within ± 2% of full scale range from +30 to +125 F and within ±4% of full scale range from 0 to +150 F. If necessary, the stability can be improved with added temperature compensation circuitry. The differential amplifier type circuit will be applied to the experimental gamma background compensated beta-gamma scintillation hand and shoe monitor to eliminate the need to use contacting meters for alarm purposes. The circuit design for this conversion was completed.

Portions of the programmer section of the portable mast system for Atmospheric Physics were completed and initially tested. The completed work included the power supply and regulator, the frequency division circuits, the ring counters of 20 and 6 for timing, gating control, and distribution, the pulse generator and buffer amplifiers, and the automatic reset circuits for the ring counters. The ring counters use standard flip-flop storage and diode AND gate cards which were modified to be useful over a wide range of input pulse conditions. Twenty commercial, inexpensive logic cards have been employed to date with only two faulty diodes found.

The Ampex tape recorder electronics equipment purchased by Atmospheric Physics Operation is being checked out. It is planned to use this equipment to obtain power density spectra and pertinent correlation coefficients related to measured wind velocity components.

Three design modifications were completed for the Atmospheric Physics Radio-telemetry System. A special mounting plate was designed to adapt the new wind chargers to the existing towers. A specification was prepared for obtaining new sine-cosine potentiometers to replace the defective units. Design was completed on a low voltage dropout circuit for incorporation into the data stations. The circuit will prevent the data stations from answering and staying energized when the supply storage battery is nearly dead. Only some Zener diodes need to be received to complete the prototype circuit. The scanning sequences of both the central station and the data stations were changed to provide more reliable and logical sequence wind direction readout. A general report was started to describe all system changes, test equipment and methods developed, complete circuits, and a detailed analysis and adjustment procedure.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples as received during May. These analyses were performed using the single-filament ion-source mass spectrometer.

An electron multiplier structure (RCA-6810A) for ion detection was washed with dilute nitric acid to attempt to reduce the internal leakage current. This technique is used by the mass spectrometery personnel at the National Bureau of Standards. This procedure had no measurable effect on the leakage current in this test.

Work continued on setting up the ion-optical test bench for experimental studies. Fabrication of several components was completed. Preliminary measurements were made of the detection of 60 kv electrons by scintillation detection.

TEST REACTOR OPERATIONS

The PCTR was operated intermittently during the month with no unscheduled shutdowns. Three sets of foils were irradiated for the Non-Metallic Materials Development Operation. The reactor was used two afternoons during the month by the University of Washington Graduate Center. One week in May was used for maintenance items.

The TTR was not operated during the month. The completion of the critical approach measurements using 1/2 inch diameter rods of 1.8 w/o Pu-Al in a 3/4 inch triangular water lattice finished the first series of tests. This work is being done in the Critical Approach Tank which uses the TTR safety and control circuits.

CUSTOMER WORK

Weather Forecasting and Meteorological Service

The mid-May forecast of prospective 1962 crests of the Columbia River flow at Hanford was issued. Predicted levels show no change from the two earlier forecasts.

Consultation service was rendered on meteorological and climatological aspects of the Recuplex incident to RPO, dispersion and deposition calculations to OR&S, and Carbon 14 releases to RPO for IPD. In addition, meteorological and radiation data from the meteorology tower were summarized for the Recuplex Study Committee and a document reporting the incident was reviewed for RPO.

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

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	93	82.4
24-Hour General	62	83.7
Special	211	86.3

May was much cooler and wetter than normal, although not record-breaking in either respect.

Instrumentation and Systems Studies

All laboratory test work was completed on the Automatic Conveyor Laundry Monitor for alpha-beta-gamma detection. Sketch drawing information as to mechanical features and electronic circuits was completed and given to Drafting Operation for use in preparation of complete drawings of the system. Final alpha detection tests with actual contaminated garments showed that Pu^{239} contamination spots as low as 200 to 400 d/m could be detected and consistent detection of 1000 d/m spots was obtained when passing over the scintillation alpha probes. Large brushes were added to the unit to provide smoothing and positioning of the garments before they are passed over the probes. Other changes included a conveyor speed reduction from 10 feet per minute to 6.5 feet per minute, an added complete overhead light shield painted with black non-reflecting paint, and a change of power circuits so the drop-drive motor is controlled by the conveyor start switch to prevent possible mechanical jamming. A design was also completed for large coat hangers which will be shop-fabricated.

Modifications were completed on a special holding and positioning mechanism for use with a nine-inch diameter NaI crystal for Analytical Laboratories, HLO. The holder had been fabricated off-site and had to be modified to obtain correct performance.

A number of standard HAPO air filters from the 325 Building were counted for plutonium activity on a rush basis in the experimental coincident-count filter counter, since the regular counters in 325 Building were inoperative due to lack of good quality double-aluminum-coated Mylar light shielding material.

All necessary commercial instrumentation was received for the assembly of one coincidence-count type alpha air filter counter to be used by Radiation Monitoring for counting of standard HAPO 4-inch x 8-inch air filters. Design was completed for the transistorized emitter followers, coincidence gates, and multivibrator circuits to be used in the system. The detection head assembly has been fabricated.

Initial testing was started on the scintillation transistorized Columbia River Radiation Monitor designed and fabricated for Environmental Studies and Evaluation, RPO.

A special circuit using a rechargeable nickel-cadmium battery was designed for use by Radiological Engineering, IPD, in their twelve "Sentinel" scintillation approximate-gamma-dose-rate monitors which have adjustable set-point alarming features both on the unit and at a remote point. The design will permit the portable units to be 110 VAC line operated with the battery being constantly trickle-charged.

Maintenance suggestions were presented to Environmental Monitoring, RPO, concerning the sensitive scintillation transistorized gamma monitors used in serial survey work and on the Columbia River.

At the request of Chemical Effluents Technology, HLO, a transistorized circuit, consisting of an amplifier and trigger, was designed for use with a tachometer which would not perform correctly with the available commercial circuit. The project was successfully completed.

Discussions were held with Fuels Technology, HLO, personnel concerning methods of detecting the surface level of uncompacted UO_2 in fuel rods during vibratory compaction. One novel idea will be investigated.

Bids were reviewed for the sixty-point data logging system to be used in the Physical Metallurgy studies of radiation effects on fissionable materials. The low bid of $25,000 by Goodyear Aircraft Company was rejected as not meeting the specifications. The second low bidder at $31,000 was Non-Linear Systems, Inc., who proposed a system which meets all requirements.

An estimate was prepared of the cost of a device for the digital readout of fuel element dimensions. The instrument will be used by Quality Control, FPD, and is similar to that developed for Fuels Design in the 306 Building installation.

Work was started on the design and fabrication of a solid-state programmer-scanner for use in Physical Metallurgy Operation's creep capsule data logging system. The new system will incorporate all solid-state logic and drives, and will use ultra-reliable mercury-wetted relays for the low-level switching. The programmer-scanner will initially handle 96 points but will have capabilities for expansion to any number of points due to the shift-register matrix design. The system will scan and record at the optimum speed compatible with the Clary printer and punch (approximately one point per second). The log cycle rate will be variable from continuous to one cycle per 24 hours and will automatically increase during a reactor transient. Every sixth log cycle the programmer will make a calibration check on the creep measurements. Other features to be included in the programmer-scanner are:

1. Push button random access to any channel.
2. Push button selection of capsules to be scanned and logged; capsules not selected to be skipped.
3. Patchboard selection of signals per capsule to be scanned and logged.
4. Scan and log cycle to be initiated automatically, by digital clock or manually from front panel.

5. Digital clock to indicate decimal time.

The programmer-scanner will be integrated with a Non-Linear Systems digital voltmeter system which will measure and digitize the input signals to ± 0.1% of reading plus one digit. A dummy expansion capsule was installed at 100-KW with a complete temperature control loop. Tests and analyses were performed on the system and optimization of static and dynamic response was achieved.

A method of reducing the adverse effects of the rate mode of control on the autoclave vessel control temperatures is being studied for FPD. It appears that insertion of a properly designed low pass filter in the controller input circuit will reduce the controller's response to the step changes produced when the two auxiliary temperatures are monitored while retaining the necessary phase lead during the autoclave startup period. FPD has been requested to install the filter circuit in one of the controllers for test purposes.

Consultation was provided Messrs. Fernard and Smith of the U. S. Army Corps of Engineers regarding methods of wind and temperature data telemetry from existing snow-depth measuring sites.

The reference system for calibrating micro-displacement readout systems, to be used by Physical Metallurgy Operation for in-reactor creep measurements, is being checked out in preparation for calibrating several third generation transducer readout systems. Operations Research and Synthesis has tentatively completed a 7090 program for correlating data generated by this program. Preliminary analysis of the December 1961 reference system self-calibrations indicates that the precision of this system has been improved to approximately ± 20 micro-inches over a 0.040 range, or about ± 0.05%, on two independent instruments.

Optics

During the five-week period (April 29-June 3) included in this report, a total of 528 man hours of shop work was performed. The work included:

1. Fabrication of a photometer for Process Control Development, HLO.
2. Fabrication of a periscope for Maintenance Engineering, IPD.
3. Fabrication of two quartz pieces for Ceramic Fuels Development Operation, HLO.
4. Fabrication of components for Traversing Mechanisms for Irradiation Testing Operation, IPD.
5. Repair of three crane periscope heads for Purex and one for B Area.
6. Modification of an infrared radiometer for Physical Measurements.

7. Fabrication of a small filter for Radiological Development and Calibrations, HLO.
8. Fabrication of a radiometer for Coextruded Product Engineering, FPD.
9. Coating of 14 lamps with stainless steel for Structural Materials Development, HLO.
10. Fabrication of 12 quartz wheels for AlSi Product Engineering, FPD.
11. Servicing in field of four crane periscopes at Purex.
12. Servicing in field of two B-Building crane periscopes.

Physical Testing

Service testing work proceeded routinely. A total of 4,164 tests were made on 2,497 items, representing some 10,134 feet of material, mostly still tubular components. However, the tubing test work has been steadily declining as various projects have been phasing out. Test work included: autoclaving; borescoping; dimensional measurements (micrometric); eddy current; heat treatment; magnetic particle; mechanical tests (bend, flattening, hardness, and tensile); metallography (macro and micro examination, and fracture); penetrant (fluorescent O.D. and I.D.); radiography (fluoroscopy, gamma-ray and X-ray); surface treatment (conditioning, pickling, steam detergent cleaning, and vapor degreasing); and ultrasonic (flaw detection and thickness measurements). Work was done for 29 different HAPO components representing most of the operating departments and service organizations, and other AEC contractors. Advice was given on 49 different occasions on general testing theory and applications.

The testing and treatment of NPR process tubes was resumed on May 7 and continued until the NPR work stoppage on May 14. During the work stoppage, IPD has been assisted on a special problem involving NPR rupture monitor sample lines. Each process tube is sampled with a 1/4-inch O.D. x 0.049-inch-wall Inconel tube. At locations on each side of the reactor, the tubing penetrates eleven feet of shielding. To make the penetration, the tubes (approximately 525) are contained in an 18-inch-diameter carbon steel pipe. Tubes are fitted for their passage through the pipe by attachment to Inconel tube sheets at each end. A high density concrete grout (up to 93% iron powder) completely surrounds the tubes. In the course of making hydrostatic tests on connecting tubing to be attached to the penetration tubes, failures occurred ascribed to discontinuities in the as-received tubing. As a consequence, the installed tubing became suspect and a nondestructive test was desired for establishing the integrity of the installed tubing. An eddy current method was developed by Physical Measurements (see Nondestructive Testing Research section of this report). Sensitivity to 7-to-10 mil deep electromachined notches on the I.D. surface was successfully demonstrated, with noise due to diameter and wall thickness changes minimized. To date some 300 tubes have been successfully

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probed in an effort to determine the extent of defective tubes. Results are still being analyzed. If the number of defective tubes proves to be less than the number of spares provided, the defective tubes will be sealed and the penetration used as installed at a considerable savings in time and cost.

In addition to the examination of the installed tubing, tubing not yet installed was examined using an encircling coil for the eddy current test. About 80 tubes between 20 and 40 feet long were tested and served as pilot runs for the development of the I.D. probe test. Approximately 100,000 feet of tubing remains to be examined.

Other tube testing work included: borescoping of a prototype process tube in the charging machine after simulated loadings of fuel; special conditioning, pickling, and autoclaving of NPR test tubes for machining and corrosion tests; and precise ultrasonic wall thickness measurements of bent tubing to referee tests made by NPR subcontractors.

Field testing activity consisted of X-ray and gamma-ray work at PRTR (rupture loop, primary system, and helium system), NPR (instrument leads), De Laval pump impellers (100-H, 100-D, and 100-DR), carbon steel, splitter blades (B, D, F, and H reactors), and graphite coolant piping for 105-N reactor fabricated by J. A. Jones at North Richland. Fluorescent penetrant tests were conducted on over-bore nozzles and De Laval pump impellers. Ultrasonic thickness measurements were made on vessels in the process cell and service basement of the PRTR reactor. Calls continued to be made for work on an emergency basis. Special assistance was given during a recent PRTR outage on radiographic examination of piping welds involving both new construction and repair work.

Testing work on NPR primary piping continued. Forty-six pieces ranging from 27" to 48" long, of NPR connector tubing, were nondestructively and destructively tested to evaluate the extent and nature of surface discontinuities. Magnetic particle tests revealed areas with discontinuities, and samples were cut out for flattening, micro, and fluorescent penetrant tests. Seven sections of the tubing were ultrasonically examined for discontinuities. The immersion test was used. No discontinuities were detected giving a response greater than the signal return from a notch 8 mils deep x 1 inch long. The results of the tests were submitted to NPR project engineers for an evaluation.

Five rings cut from NPR primary piping (18" O.D. x 1.036" wall) were ultrasonically tested for discontinuities prior to start of fatigue testing. Ultrasonic tests will be made during the fatigue test at intervals to determine if propagation of the discontinuity will occur. Ultrasonic, macro, fluorescent penetrant, and side bend tests were made on a sample which had failed in fatigue test. The failure occurred in the base metal about three inches away from the weld. Inspection of the weld disclosed discontinuities ranging from 5 mils to 90 mils. Discontinuities found in the weld gave no evidence of influencing the failure of the base metal.

A major problem was the examination of chromium steel roller-bearings used to expand the ends of NPR process tubes for installation. A set of five roller-bearings (pills) are placed on a mandrel and inserted in the tube and rotated to flare the tube. Some pills broke during the flaring process. We were asked to help determine the cause of the failures. The pills, designed with a long taper on one end and a short taper on the other, failed in the area of the thickest diameter. Magnetic particle inspection conducted on ten new pills found them free of defects, but of 93 used pills, fifty percent were defective in the area where the failures occurred. No correlation was apparent between the extent of the indications and the amount of usage. Five of the new pills now in service are being given periodic checks to determine the extent of indications permissible before failure can be expected. Procedures are in effect to periodically test all pills in service. Metallographic examinations revealed the magnetic particle indications to be cracks. Rockwell hardness tests showed the pills to be in the Rockwell C-63 to C-67 range, indicating the elasticity of the material is almost negligible. New pills are now being manufactured with specification changes as to the degree of hardness in an effort to eliminate failures.

A procedure was set up to fluorescent penetrant test on NPR pump sleeve after recycle corrosion tests. The penetrant test will be done intermittently during the corrosion test to determine if cracks occur.

A number of miscellaneous examinations included: Inconel to stainless steel weld samples to determine the quality of the weld; Inconel socket welds for discontinuities in the weld; stainless steel weld samples taken from welded chains; and stainless steel mechanical seal fittings.

An ultrasonic test was developed to detect three-mil-wide x three-mil-deep x 1/4-inch-long electromachined notches located on the O.D. and I.D. surfaces of a 4-1/2-inch O.D. pipe standard. This was designed for testing header piping in the 189-D Building thermal hydraulic laboratory.

The radiographic density of uranium was evaluated in the molten and solid state with gamma-radiography. The test setup consisted of induction heating a tungsten clad uranium fuel element in an argon atmosphere. A brass water jacket cooled the fuel elements sufficiently to allow placement of film on the outside. Temperature was determined and controlled by pryometer readings.

A study was made of methods for radiographically determining density changes in Pu alloy fuel elements. Twenty-five samples with Pu density variations from 0.5 to 16.8 gm/cm^3 were used. The study utilized 250 KV X-ray, Co^{60}, and Ir^{192}; all sources of radiation were evaluated for contrast, density latitude, and detail. The X-ray results indicated a material density change of 2.0 maximum with one exposure and necessitated several setup changes and exposures to cover the complete range of densities. In addition, the material had to be placed in a masking template to prevent undercutting. The contrast and detail was good in the low density materials but was limited in penetration of higher density material. Cobalt-60 covers a wider density latitude but loss of detail and contrast is great. Good results were obtained with the Iridium-192 by the use of two films of different speed; it was found the complete ranges could be covered with one exposure; no masking template was necessary; detail and contrast were good over the entire density range.

An unusual problem was encountered when a piece of tool steel was lost while loading and compacting UO_2 PRTR fuel elements. It was necessary to attempt to find the steel in the uranium element. It was shown that Iridium-192 could give good definition of the steel in a test sample.

Work on slit radiography was continued with back-scatter still presenting a masking problem. Successful radiographs were obtained on a variety of pieces including pipes from 2" O.D. up to 8" O.D. A 1/4" slit is used at the sample surface and a speed of rotation of 1 rpm with appropriate exposure times gave good results.

Development of equipment for detecting process tube cracks is complete. A functional test was made on two 105-H process tubes. Tube No. 2371 had one defect indication which was equivalent to a calibration standard. This standard was a 0.020" deep x 0.125" long transverse notch in the I.D. of a 105-H process tube. A total of 10 other indications ranging from 20% to 55% of the standard indication were found. Tube No. 2371 has been removed from the reactor for visual examination. Sections where indications occurred will be fully examined by sectioning for discontinuities.

A reactor Parker fitting which was deliberately subjected to intergranular corrosion and cracking was tested. No defect indications were observed. The depths of the cracks are unknown. Since the process used for causing cracking causes uniform attack, the entire fitting was exposed. The tapered end where the ultrasound enters was also cracked. This may have prevented or reduced ultrasound entry. Further testing will be delayed pending acquisition of samples having cracks localized in the vicinity of the fitting thread relief.

ANALOG COMPUTER FACILITY OPERATION

The major computer problems considered during the month were:

1. NPR Simulator
2. "C" Column Simulation
3. Reactor Instrumentation Studies
4. Waste Cask Heat Transfer

Eighty-two percent of the GEDA equipment (average), and ninety-one percent of the EASE equipment was in good operating condition during the month. Computer utilization was as follows:

GEDA	EASE	
144	147	Hours Up
21	21	Hours Scheduled Downtime
3	0	Hours Unscheduled Downtime
0	0	Hours Idle
168	168	Hours Total

Routine maintenance of the computers was good although minor troubles were experienced with the GEDA function generators and patch bay. The EASE scanner would not scan through the alphabits class at the beginning of the month, but this problem was corrected by placing 220 K resistors in series with the set lines feeding the alphabits registers.

INSTRUMENT EVALUATION

Tests were started on the use of rechargeable nickel-cadmium batteries in a prototype transistorized portable G.M. monitor. Tests indicate at least 25 hours of constant service can be obtained before recharge is necessary.

Substitution of a new diode pump circuit oven in the logarithmic and linear response area monitors has proved the worth of using the new oven, because the noise level was considerably reduced. Both the logarithmic and linear ranges calibrated correctly on one monitor scheduled for use in the 326 Building.

All evaluation and acceptance tests were satisfactorily completed on one vendor prototype Model II Scintran of 65 on order by RPO. The units are being fabricated by Instrument Laboratory, Inc., of Seattle.

All circuit prints were completed for the prototype scintillation transistorized alpha portable "poppy" instrument which successfully completed all evaluation tests. Following preparation of mechanical layout prints, some 30 units will be ordered off-site by Radiation Protection Operation.

Of some 200 Bendix 0-200 mr self-reading pencil dosimeters tested in cooperation with RPO, about 20% had a response of minus 10% for a given exposure whereas the acceptable limit is minus 5%.

Preliminary acceptance tests were completed and one minor circuit modification was made to eliminate interference on two scintillation transistorized alpha-only hand and clothing monitors being fabricated by the 328 Building Electronics Shop; one for use at Biology Operation, and the second for use by J. A. Jones Company at a waste treatment facility. The register readout signal-to-background ratio is 5 to 1 for a 500 d/m Pu^{239} source distributed over a 4-inch x 8-inch area. This is quite satisfactory. Layout photographs of the units will be made before they are sent to the field so future instruments will have a similar construction.

To eliminate minor occasional erratic operation of the 30 aluminum chassis Model II Scintrans fabricated off-site, one unit was modified to use a buss-wire common ground soldered connection to all ground points. This change eliminated, at least to date, the spurious action apparently caused by aluminum oxide film buildup on the many ground points.

RS Paul for
Manager
PHYSICS AND INSTRUMENT RESEARCH
AND DEVELOPMENT

PF Gast:RSP:mcs

CHEMICAL RESEARCH AND DEVELOPMENT OPERATION

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Characteristics of NPR Decontamination Wastes

A document, "Crib Disposal of the NPR Decontamination Wastes," HW-73482, is being prepared. A summary from this document is as follows:

Waste liquid permeability experiments showed the permeability of specified NPR decontamination wastes to be 43 to 65 percent that of water. A percolation rate for the sludge-free waste solution is expected to be 5 gal/ft^2/day. No evidence of soil plugging by the filtered wastes was detected. Percolation of the waste solution was almost stopped by the presence of sludge. More than six hours of standing was required to settle completely the slowest settling Phos-1® (a proprietary 10 percent phosphoric acid solution containing corrosion inhibitors) waste solids. The annual volume of waste sludge is expected to be in excess of 25,000 ft^3.

Waste frothing was recognized as a possible problem in waste handling.

Treatment of NPR Decontamination Wastes

Dibasic ammonium citrate is a current candidate to replace Phos-1 for decontaminating the non-stainless steel portion of the NPR primary coolant loop. Preliminary experiments with one percent citrate solutions, simulating the waste concentration, and containing Co-60 were carried out. The best scavenging procedure found to date is the addition of ferrous sulfate followed by sodium hydroxide to at least pH 12. Potassium permanganate and cobalt nitrate showed limited scavenging while calcium and aluminum salts were ineffective.

The addition of 100 ppm Fe^{+2} removed 10, 95 and 99.7 percent of the cobalt at pH 10, 11 and 12, respectively. Increasing the Fe^{+2} concentration to 400 ppm increased the Co-60 removal to 80, 98 and

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> 99.9 at the corresponding pH values. Scavenging with permanganate, with or without Fe^{+2} present, showed a maximum Co-60 removal of 75-80 percent, as did a cobalt salt.

Citrate solutions containing cerium, zinc, strontium and zirconium were effectively scavenged (> 99 percent removed) at pH 12 by the addition of either the Fe^{+2} or the permanganate. Cesium was not removed and only 40-70 percent of the ruthenium was removed by either iron or permanganate.

Ground Water Temperature Studies

A temperature profile obtained from an abandoned farm well, located one-half mile east of 100-B Area, showed the temperature of the ground water at this site to be 43 C. This new information indicates that thermally warm water from beneath 100-B Area is moving toward the east as well as toward the west and north. Ground water samples from four wells located six miles east of 100-B Area, which show temperatures several degrees above background, are being analyzed for radionuclides (primarily Cr-51) present in reactor effluent.

Airborne Particulates in Reactor Operations

Final samples were collected in the program to characterize airborne particulates from reactor tube replacement. A report of this work is being written.

Effluent Monitoring

A faulty voltage regulator tube in the power supply for the scaler circuit caused the only malfunction of the As-76 monitor during the month. The sample supply to the monitor was interrupted on May 26, by failure of a flexible coupling on the 107 inlet sample pump.

CCl_4 extractions were used to measure iodine interference in effluent samples from 100-D and 100-H during the extended reactor outage at 100-F. Interference ranged from 10 to 20 percent in these samples. The iodine interference in the 100-F effluent decreased from approximately 50 percent before the outage to 8 percent after the outage.

A carbon tetrachloride extraction treatment of the monitor sample appears to be the most satisfactory method of removing the iodine interference. Heating the sample and purging with noble gas caused only minor reductions in interference. The extraction techniques may be useful as a method for rupture detection.

Reactor Effluent Water Radioisotope Studies

A reactor tube study of the effect of addition of silicate on the production of radioisotopes in the effluent water was started at the KE reactor on April 30, 1962. The tube to be tested and a control tube to which no silicate was to be added were chemically cleaned prior to reactor start-up. Ten parts per million silicon as sodium silicate were added continuously for five days when the test had to be interrupted because an improperly mixed batch of sodium silicate caused a high pressure drop in the tube. After the five days the concentrations of P-32, Mn-56, As-76, Cr-51, Np-239 and Zn-65 were lower in the test tube than in the control tube by about a factor of two. The addition was restarted four days later although regular process water without added silicate had been used during that period, and the radioisotopes concentrations in the two tubes were about equal. The factor of two decrease in the radioisotope concentrations was less than expected from the laboratory studies.

Reactor Water Treatment Process Studies

A laboratory water treatment plant study of the effect of alum addition on phosphate ion removal under conditions similar to those at the K areas was completed. An indication was obtained that the poor removal observed when alum addition was adjusted to establish an optimum zeta potential of the floc may well have been due to the delay of some 30 seconds which elapses between alum addition and pH adjustment in the plant. This hypothesis will be tested in future studies.

The particulate nature of some of the parent materials in Columbia River water was determined by filtration in a study of means of removal of these materials. River water was passed through a succession of eleven membrane filters with decreasing pore sizes and the residue on each filter was neutron activated and analyzed for P-32, Sc-46, Co-60 and Zn-65. The particulate material appeared to be in two particle size ranges. The amounts of the particulate parent material held are tabulated below:

Pore Size (mμ)	Isotope (%) P-32	Sc-46	Co-60	Zn-65
> 3000	78	80	39	26
[redacted] 50 - 300	15.5	19	57	51

[redacted]

[redacted]

The small particulates would probably not be retained by the process filters and would contribute to the radioisotopes formed in reactor effluent water. Further studies of the particle size distributions of additional parent materials and of other water streams are in progress to guide further removal studies.

SEPARATIONS PROCESSES

Purex Process

Studies were continued to discover the source of material causing the reduction in uranium transfer rate noted when Purex plant 2DX (R-7) is contacted with simulated 2DFS solution.

Transfer rates continue to show the presence of a surface active material in R-7 which is removed by permanganate treatment but not by carbonate washing. Severely degraded Soltrol-TBP was carbonate washed and added to laboratory solvent. No decrease in extraction rate was found. Thus, although degraded Soltrol has undesirable physical properties, it does not appear to be the source of the material adversely affecting the transfer rate. Both anion and cation types of ion exchanger resin were pulverized and extracted with warm water. When added to the 2D extraction system, no effect on transfer rate was observed. An appreciable reduction in transfer rate was observed, however, upon either addition of methanol or water extracts of material held up by the glass wool water filter downstream from the plant ion exchange water treatment beds. These extracts also very markedly increased coalescence times in alkaline washing steps; but the effect was much less in acid systems. The infra-red absorption analysis of these extracts showed -OH and -COOH bands as well as others tentatively identified as due to -ONO or $-NO_2$ groups. The source of the organic material on the filters has not been established, although its presence is unquestionable.

Diluent Studies

A sample of South Hampton Company's molecular sieve process n-paraffin, SOHO (our nomenclature) has been examined. The molecular weight was too low, but the chemical behavior was satisfactory. More comprehensive tests will be made when the sample of regular production material is received.

Of the immediately available petroleum diluents, Penn Refining Company's 2251 oil still appears to be well ahead with respect to chemical stability and would be an excellent candidate for interim Purex operation pending the larger scale production of pure n-paraffins.

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Plutonium Recovery from RMC Line - Continuous Task I Oxalate Supernate

The work reported last month on the development of an anion exchange process for the recovery of plutonium oxalate-containing Task I supernate was completed during the month and a final summary report will be issued as soon as remaining analytical work is completed. Process variables investigated in May included (1) the effect on resin capacity of higher nitric acid concentrations than previously used, (2) the effect of higher temperature and flow rate, and (3) the effect of a large amount of fluoride. Resin capacity at ambient (25 C) temperature, 8.8 M HNO_3, and a flow rate of 8 ml/min.cm^2 was about 45 grams Pu/liter of resin. Operation at 58 C and at a flow rate of 17 ml/min.cm^2 increased the loading to 75 grams/l, which is essentially the theoretical capacity of the resin. Increasing the acidity to 10.0 M HNO_3 decreased the capacity only to 68 g/l, indicating a wide operating range. Addition of fluoride equivalent to 0.2 M (plus a 1:1 mole ratio of aluminum) did not affect capacity. It is possible that less aluminum would be equally effective.

In summary, the solution to the precipitate and resin loading problems encountered in the initial stages of this study is to increase the feed nitric acid concentration to 9-10 M and to add aluminum, if fluoride is present. All but a trace of precipitate is dissolved at the high acidity and the oxalic acid ionization is suppressed sufficiently to prevent excessive complexing of the plutonium by oxalate.

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

Studies on the formation of uranium(IV) nitrate by the reduction of uranyl nitrate with aluminum metal were continued. The starting solutions contained only uranyl nitrate, nitric acid and hydrazine or sulfamic acid. Highlights of the findings include:

1. Laboratory scale runs indicate that a bed of aluminum turnings can be substituted for the powdered aluminum used in previous studies.

2. The reduction rate increases with an increase of temperature.

3. The source of the uranyl nitrate solution does not appear to affect the reduction reaction.

4. Runs made on a 200 liter scale produced results consistent [redacted]those experienced on a 500 ml scale.

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Charging of Overbore Slugs in a Multipurpose Dissolver

An increase in the diameter of the fuel elements used in the production reactors is being contemplated. Some concern has been expressed about possible jamming of these overbore slugs in the annular crib of the multipurpose dissolver.

Steel slugs, 2 inches in diameter and 8.9 inches long were charged to a carbon steel mockup (OD of 70 inches, ID of 49 inches) of the Redox multipurpose dissolver slug crib. No jamming in the slug crib was observed during the dumping of 16 buckets of the steel slugs. When the slugs were vertically stacked in the slug bucket, some jamming between the bucket and the conical distributor was observed. Raising and lowering of the slug bucket was effective in breaking the jam.

The "overbore" steel slugs occupied 50 percent of the crib as compared to 44 percent of the crib with "standard" steel slugs (1.5 inches in diameter and 8.9 inches long) used in prior tests on the same mockup.

Disposal to Ground

A tritium material balance around the Purex and Redox plants for the month of November, 1961, revealed that less than one-half of the fission product tritium present in the irradiated fuel elements can be accounted for in the waste streams discharged to ground. Analyses of several composited process condensate samples showed up to 300-fold variations in tritium concentrations; this stream was expected to contain most of the tritium. Arrangements were made to collect and analyze samples of coating removal waste, ammonia scrubber solution and stack gas to assist in determining the prominent paths of tritium discharge from the separations plants.

Soil column tests on limestone-neutralized and unneutralized Purex process condensate waste were completed at the Redox laboratory and the results were plotted and analyzed. Results, nearly identical to those noted several years ago, showed 85-95 percent breakthrough of Sr-90 in less than three column volumes of waste in each case. Thus, there appears to be no significant benefit resulting from the neutralization of this waste stream with limestone. The Sr-90 concentration in the waste is consistently below the recommended ground water limit of 1×10^{-7} μc Sr-90/cc.

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Iodine-131 in Airborne Effluents

The adsorption on charcoal of I-131 from streams carrying volatiles and particulates was further studied. These experiments were exploratory to disclose atmospheres from which I-131 would not be efficiently removed, and to help identify the reasons for lower than anticipated I-131 removal on charcoal used in stack sampling and in recent tests with oxidizer off-gases. The following table presents the data obtained relative to adsorption of I-131 in various atmospheres:

Charcoal Adsorption of Molecular I-131 and Removal of Particulates and Vapors Carrying I-131 by High Efficiency Filters

Charcoal Pretreatment	Gas Carrying Stream	% First * Charcoal Capsule	% Second* Charcoal Capsule	% Filter ** (Molecular)
Saturated with Trichloroethane	Air + trichloroethane	99.4	0.6	-
None (new)	Unfiltered air + metal fumes from sparking copper	89	3	8
None (new)	Air passed over simulated boiling oxidizer solution containing I-131	99	1	-
None (new)	Air + oil mist from pump	20	63	17
None	Air + hot oil vapors	86	5	9
None	Air + oil vapors passed through a high efficiency filter	78	11	11

* 1/2-inch diameter by 1-1/2 inch long

[redacted]lter was downstream of second charcoal trap

[redacted]

Trichloroethane vapors which may be present from degreasing-decontamination does not lower the efficiency of charcoal. The presence of oil or fine particles materially reduces the effectiveness of charcoal for removing I-131.

Temperature measurement in a small bed of charcoal revealed that a strong exothermic reaction occurs when approximately 30 percent NO_2 in air is passed through hexone-saturated charcoal. The temperature rose rapidly to 240 C under these conditions. When NO_2 passed through clean charcoal, temperatures of about 80 C were recorded. This observation may be of considerable interest in potential applications of charcoal to streams which may have hexone and NO_2 present. When NO_2 was adsorbed on charcoal, then hexone vapors passed through, no evidence of reaction was observed.

WASTE TREATMENT

Cesium Removal from Formaldehyde-Treated Waste

Research continued on the separation of cesium from FTW waste using mineral columns and on the subsequent loading of the separated cesium on high capacity inorganic exchangers.

Zeolon and Linde AW-500 were evaluated for separation of the cesium from FTW. The waste volume processed to cesium breakthrough on Zeolon was about 80 percent that of clinoptilolite. The AW-500 was not stable at the acidity of the FTW.

Gas evolution during elution of cesium from clinoptilolite was minimized by washing with 1 $\underline{M}$ NH_4OH prior to eluting with a solution of 2 $\underline{M}$ $(NH_4)_2CO_3$ + 0.5 $\underline{M}$ NH_4OH at 55 C.

Eluates from clinoptilolite separations of cesium from FTW were boiled to remove $(NH_4)_2CO_3$ and then passed through beds of candidate cesium packaging materials. Cesium loading capacities for these solutions on Linde 13X, AW-500 and Zeolon were 2.2, 2.1 and 1.6 meq Cs/g.

Cesium was also loaded on 13X and AW-500 from a solution containing 0.02 $\underline{M}$ cesium and 0.5 $\underline{M}$ HNO_3 neutralized to 0.5 $\underline{M}$ $NaNO_3$. This solution is a simulated cesium product from the dipicrylamine extraction process. Loadings were 0.43 and 1.32 meq Cs/g for 13X and AW-500, respectively.

Strontium, Cerium, Ruthenium, Niobium and Neptunium Behavior During Cesium Removal from Formaldehyde-Treated Waste

The behavior of major fission products and neptunium during the loading of cesium on clinoptilolite from a simulated FTW solution was investigated. A simulated FTW solution spiked with Purex 1WW was passed through a column of clinoptilolite until 80 percent cesium breakthrough was reached. Strontium, cerium, ruthenium and neptunium reached 50 percent breakthrough within one column volume. About 95 percent of the niobium was adsorbed on the clinoptilolite. Measurements were not possible for zirconium because it was not detectable in the spiked feed solution.

The clinoptilolite column was subsequently flushed with one molar nitric acid, water and one molar ammonium hydroxide. The cesium was then eluted with two molar ammonium carbonate at 60 C. Decontamination factors for strontium, cerium, ruthenium, and niobium were 2000, > 1000, > 1000 and 10, respectively. Most of the niobium remained on the clinoptilolite during flushing and elution.

Elution of Cesium from Zeolites after Heating

Additional results were obtained on the elution of cesium with ammonium nitrate from clinoptilolite which had been heated to 200 and 400 C after loading. Ninety-nine percent of the cesium was eluted with 20 column volumes of 2 $\underline{M}$ NH_4NO_3 from the mineral which had been heated to 200 C for periods up to 510 hours. After heating to 400 C for the same time period, 93 percent could be similarly eluted.

Cesium loaded on Linde 13X and heated to 600 C for 24 hours was only 61 percent eluted with 20 column volumes of 2 $\underline{M}$ NH_4NO_3; however, 99 percent was eluted with 0.5 $\underline{M}$ HNO_3 in 50 column volumes. There was no evidence of Cs-137 volatilization during the 24-hour heating period.

Greater than 99 percent of the cesium could be removed from Zeolon in equilibration experiments with 2 $\underline{M}$ NH_4NO_3 after the loaded mineral had been heated for 24 hours at 600 C.

Recovery of Neptunium from Purex Formaldehyde-Treated Waste with D2EHPA

Work was continued toward developing a satisfactory flowsheet for recovering neptunium from Purex plant FTW solutions. The flowsheet currently under study involves reduction of Np(V) to Np(IV)

[redacted]

and Pu(IV) to Pu(III) with iron(II) sulfamate and hydrazine. Over 95 percent of the neptunium is extracted in a single batch contact with one-tenth volume of 0.04 M to 0.1 M D2EHPA in Soltrol. Neptunium is removed from the organic by stripping with 0.5 M $H_2C_2O_4$. Experiments with synthetic FTW showed that neptunium extraction is not significantly altered by eliminating TBP from the extractant or by substituting Soltrol for Shell Spray Base. Neptunium extraction equilibrium was obtained within 10 minutes contact at either 25 or 50 C; neptunium distribution ratios were lower at the higher temperature. In decontamination studies with tracer Zr-95 - Nb-95 appreciable co-extraction of these elements with neptunium occurred. Complexants such as EDTA, HEDTA and citric acid suppress extraction of zirconium-niobium. Their effects on neptunium extraction are being studied.

Solvent Extraction Removal of Fission Products from Purex FTW

Miniature mixer settler runs were made with feed solutions prepared from synthetic Purex FTW solution to test a study flowsheet in which citric acid was used both as buffering and complexing agent. Feed pH was 4.0; extractant was 0.2 M D2EHPA - 0.2 M TBP - Soltrol; scrub was 0.25 M citric acid at pH 2.9. With seven extraction and five scrub stages and at relative flows of 1.0/0.21/0.98 feed, scrub and extractant, greater than 99 percent of the strontium and over 90 percent of both cerium and promethium were extracted. Decontamination from cesium, zirconium-niobium and sodium was excellent; sodium concentration of the organic product was only 0.002 M. Decontamination from iron was poorer than with HEDTA complexed feeds but still acceptable.

Product Forms

The thermal stability of calcined strontium oxide was studied in a pressure bomb experiment. Calcined SrO was heated in a closed system to 900 C with a resulting pressure of 160 psi. After recalcination at 900 C in an open system, the molten material changed to a hard white solid apparently due to decomposition of $Sr(OH)_2$. No pressure was experienced on subsequent reheating in a closed system.

Strontium carbonate plus lithium fluoride sinters to a hard porous cake at about 650 C with as little as 3 w/o LiF. To produce a freely-flowing melt, at least 12 w/o LiF and heating to 800 C are required. The dense product formed by complete melting of the fluxed $SrCO_3$ has a bulk density of 3.2 g/cc and is 50 w/o Sr. The leach rate in tap water is 0.002 g/(hr)(cm^2).

[redacted]

Prototype Loading Station

A conical filter 7.5 inches in diameter by 16 inches high with a one-inch annular cake space inside a conical filter can has been tested with precipitated $SrCO_3$. To prevent the precipitate passing the filter screen (16 x 150 mesh wire twill), diatomaceous earth filter aid was added to the feed slurry. The filter cake contained about 0.35 g Sr/cc and filled completely the annulus.

When the conical filter was removed from the filter can the filter broke away cleanly from the filter cake. The can was then heated for dehydration of the cake. There were minor splashes of wet cake during the boiling period, but there was no foaming. The whole filter can eventually attained the desired temperature of 800 C. Heating was by a 15 KW induction heater.

Fission Product Binding Materials

Two high capacity synthetic alumino-silicate zeolites, Linde 4A and Linde 13X®, have been proposed for the fixation of cesium or strontium. Thermobalance measurements on these two materials showed them to be stable to weight loss at temperatures in the 500 to 960 C range. Attempts were made to fuse these two zeolites using 2 to 25 w/o of NaF, KF, CaF_2, LiF, PbF_2, CuF_2 or lead oxide as a fluxing agent. Maximum temperature reached was ca. 925 C. Of these, only lithium fluoride shows promise as a fluxing agent. The 13X material with 10 and 25 percent and the 4A with 25 percent lithium fluoride formed dense hard compacts with apparent fusion.

In-Tank Solidification

In laboratory studies synthetic "old coating removal waste" was concentrated at constant temperatures of 91 and 103 C to form slurries which solidify completely when cooled to 25-50 C. These tests at temperatures well below the boiling point simulate more nearly conditions expected when concentrating by sparging with a hot gas than do previous tests in which the solutions were maintained at the boiling point. The final slurries (before cooling) were estimated to contain 10-20 percent solids which tended to settle out and which could interfere with operation of an air lift circulator. The solids appear to have the same tendency toward deliquescence as those formed at the boiling point. In either case, the solids hold more water than can be accounted for by known hydrate forms; they are apparently gels.

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Studies were initiated in a 4-foot diameter by 15-inch deep insulated tank on hydraulic and thermal behavior of in-tank solidification of coating waste. Particular emphasis is being placed on evaluation of solids deposition patterns and sphere of influence of the circulator. The tank is heated with an external circulation loop. Design and operation is aimed at simulation of the hydraulic conditions occurring in a 75-foot diameter underground waste storage tank undergoing solidification by hot air sparging.

In an initial test, 313 Building coating waste was evaporated 2.2 fold over an eight-day period without carbonation. Sludge gradually built up to a depth of 1/4-inch fairly uniformly over the bottom of the 4-foot diameter model. Viscosity of the supernatant liquor measured by Brookfield viscosimeter increased from 4.5 to 23 (relative scale). The run was terminated by plugging in the heated recirculation loop.

The model tank was successfully solidified in a second run. Details will be reported next month when all data are available.

TRANSURANIC ELEMENT AND FISSION PRODUCT RECOVERY

Assistance to Purex Head-End Fission Product Recovery

Previously reported laboratory work demonstrated that cerium and promethium were not precipitated (as the sulfates) from tartrate-complexed FTW when hydrogen peroxide was added to the solution, provided the pH was between 2 and 2.5. Partial precipitation was observed at pH 1.5. Further experiments have shown that peroxide has no effect on the lead-carrier precipitation of strontium, raising the interesting possibility that peroxide addition could be used to bring about separation of strontium and rare earths. This would permit eliminating the current rare earth oxalate precipitation step, and the strontium (carbonate) concentration step which follows rare earth removal (and is required because of the dilute character of the oxalate supernate). Overall effect would be to reduce Purex strontium recovery from a four-step process to a two-step process, with a substantial increase in plant capacity and decrease in cumulative strontium loss. Alternately, using peroxide while retaining the oxalate step should yield a strontium product more highly decontaminated from cerium and rare earths, perhaps permitting simplifications in the Hot Semiworks solvent extraction flow sheet.

The effects of pertinent variables were as follows: (1) Peroxide had no effect on strontium recovery, which generally averaged greater than 90 percent in the precipitate after washing and metathesis, (2) the presence of both tartrate and peroxide are requird to prevent rare earth precipitation, (3) variation of hydrogen peroxide concentration between 10 and 40 ml (of 30 percent H_2O_2) per liter of FTW and tartrate (5 $\underline{M}$) between 100 and 400 ml per liter had only slight effect on cerium hold-back, although higher concentrations of each yielded better cerium decontamination factors, and (4) cerium decontamination factors were generally in the range 5 to 10. These are about the same as are achieved in the plant in the oxalate step and are adequate to reduce the heat load in the CR vault to an acceptable value, the primary reason for rare earth removal.

The fact that both tartrate and peroxide were required, plus visual observations on the color of the ferric tartrate complex, suggests that a tartrate oxidation product, probably an alpha-hydroxy acid, is responsible for the observed rare earth behavior. The expected decomposition products of tartaric acid are being tested to determine whether any of them exhibit unusual complexing ability for cerium. Hot cell runs with actual plant feed are also planned.

Removal of Cesium from Purex Supernate by Ferrocyanide Precipitation

The carrier precipitation of cesium from Purex waste with ferrocyanides, ferricyanides, etc., has been extensively investigated in the past and demonstrated in hot cell experiments with plant waste. However, the flowsheets used were designed to yield a product of high specific activity rather than to produce very high degree of cesium recovery. The study has accordingly been extended to define conditions for achieving the 95-98 percent cesium removal required by the Waste Management Program to control salt-cake self heating. Laboratory experiments indicated that this degree of removal is easily achieved, at least with synthetic solutions and efficient phase separation. Two one-liter scale hot cell experiments were run during the month with actual 103A Purex supernate. A continuous overflow solid bowl centrifuge (20 min. residence time at 700 G) was used for precipitate removal to realistically simulate plant centrifuge behavior. Over 99 percent cesium removal was achieved in the first run, using full strength supernate, 0.001 $\underline{M}$ Ni, and 0.005 $\underline{M}$ potassium ferrocyanide. Feed diluted with an equal volume of water (and with only half the above concentrations of nickel and ferrocyanide) was used in the second and yielded 97 percent cesium removal. The observed loss with diluted feed was somewhat poorer than laboratory results but could doubtless be improved by using more precipitant.

The cesium-bearing nickel ferrocyanide precipitates from the two hot cell experiments were slurried from the centrifuge and metathesized with one gram each of solid silver carbonate (to form soluble cesium carbonate and insoluble silver ferrocyanide and nickel carbonate). Recoveries in this step were very poor (17 and 6 percent, respectively), probably because of mechanical slurry transfer problems. Additional experiments are being performed to resolve this point.

Cesium Solvent Extraction with Dipicrylamine (DPA)

New information was obtained during the month on selective scrubbing to remove sodium, disengaging times were measured, and a multiple batch extraction of cesium from 103A supernate was performed in B Cell.

When DPA (dipicrylamine) is used to extract cesium from a waste solution containing large quantities of sodium, any DPA not associated with cesium will be converted to the sodium form (Na.DPA), and subsequent stripping with dilute acid will remove both together, resulting, for example, in a Na/Cs ratio of ten if 0.001 M cesium is extracted with 0.01 M DPA. While further enrichment of cesium with respect to sodium could be achieved by additional cycles, use of a selective scrub to remove the extracted sodium prior to cesium stripping would be desirable and would result in a much simpler process. Ammonium ion was found to be an effective scrubbing agent since its extractability into DPA falls between sodium and cesium. Furthermore, most ammonium salts can be readily volatilized from the strip solution or product to yield cesium free of all other metallic ions. In experiments simulating an extraction-scrub column followed by a stripping column, cesium was extracted from either 103A supernate or 1965 FTW with 0.01 M DPA in nitrobenzene, scrubbed twice with 0.02 M ammonium nitrate, and then stripped with dilute nitric acid. The decontamination factor for sodium (feed to product) exceeded 10^4. The cesium recovered from FTW contained only 3 w/o sodium and that from 103A supernate 19 w/o. Use of more than two scrub stages would yield essentially "CP" cesium.

The disengaging times of dipicrylamine-nitrobenzene solution in contact with various aqueous phases were determined in support of current cold Semiworks pulse column studies. Disengaging times against synthetic 103A supernate were 176 seconds, aqueous phase continuous, and 76 seconds, organic continuous. Against 1965 salt waste, the corresponding numbers were 54 seconds, aqueous continuous, and 80 seconds, organic continuous. While these values may be satisfactory, it was found that addition of 25 to 75 ppm of "Mistron" (a magnesium aluminum silicate surfactant) shortened the disengaging times by a factor of about three.

A hot cell experiment was performed in which a liter of actual 103A supernate was extracted batchwise three times with equal volumes (O/A = 1) of 0.01 M DPA diluted with a 50-50 mixture of nitrobenzene and "Tetralin" (tetrahydronaphthalene). (Purpose of the Tetralin was to decrease the specific gravity of the organic.) After three extractions, approximately 97 percent of the cesium had been removed from the aqueous. Other analytical results are not yet available.

ANALYTICAL AND INSTRUMENTAL CHEMISTRY

Cesium Analytical Procedures

Results were reported last month on development of an improved, rapid radio-chemical method for cesium-137, based on the use of dipicrylamine-nitrobenzene solvent extraction (to replace the tetraphenyl boron extractant previously used). The analytical use of DPA solvent extraction has been extended this month to include flame photometric determination of cesium (which would include both active and inactive cesium), rubidium and potassium. The general scheme is to neutralize acid samples with lithium hydroxide and to complex iron, and other elements which are insoluble in basic solutions, with the lithium salt of EDTA. (Lithium is used because it interferes less with the extraction than does sodium, particularly in the case of rubidium and potassium, and its greater solubility allows higher concentrations of DPA to be used.) The neutralized, complexed sample is then extracted with DPA and the organic phase read directly on the flame photometer. The procedure has been applied with good success and is being used in the Analytical Laboratories.

EQUIPMENT AND MATERIALS

Corrosion of 304-L by HNO_3-HF-$Al(NO_3)_3$-$Pu(NO_3)_4$ Solutions

Samples of 304-L stainless steel were exposed at room temperature to a variety of solutions containing up to 16 M HNO_3, up to 200 g/l $Pu(NO_3)_4$, up to 1.0 M HF and up to 0.10 M $Al(NO_3)_3$. The purpose was to evaluate corrosion potential in 234-5 Building storage and transport containers. Corrosion rates observed were $\leq$ 0.6 mils/mo. Two samples exposed at higher temperatures, 60 C and boiling, corroded rapidly, 25 and 250 mils/mo, respectively, as expected. [redacted]

Corrosion in Formaldehyde Treatment Reactor

As reported last month, corrosion rates for 304-L stainless steel in chromium-free solutions simulating the conditions expected in the Purex formaldehyde treatment vessel were about 0.5 mils/mo. In the presence of 0.01 M Cr(VI), rates of five to seven mils/mo were found; the presence of Cr(III) did not significantly affect corrosion rates. It is expected that chromium will be present as Cr(III) in this vessel.

Corrosion of Tantalum by Some Purex Solutions

Samples of commercially pure tantalum were exposed in the liquid and vapor phases of boiling synthetic Purex 1WW containing 0.004 M, 0.02 M and 0.04 M HF. Other samples were so situated that the condensate from the boiling solutions flowed over them. In all cases, corrosion rates were less than 0.01 mil/mo.

Commercially pure tantalum was also exposed to various decontaminating solutions used in the Purex plant--caustic tartrate at 80 C, oxalic acid-nitric acid solution at 60 C, alkaline permanganate at room temperature and nitric acid-ferrous sulfamate at room temperature. Corrosion rates in all cases were less than 0.01 mil/mo.

Stress Cracking in 1020 Mild Steel

Testing of notched and stressed 1020 mild steel C-rings was continued. No cracking failures have been obtained to date in simulated Purex stored waste, neutralized current formaldehyde treated waste or coating removal waste. Some cracking has been observed in 50 w/o $NaNO_3$ at 80-90 C on specimens which have been (1) given a spheroidizing treatment, (2) given a quench hardening treatment, (3) etched in hydrochloric acid, or (4) stressed to initiate cracking in the base of the notch. From these and prior studies by other workers, it appears that three approaches to minimize susceptibility of underground tanks to cracking offer merit; namely, (1) anneal the fabricated tank, (2) utilize cathodic protection, and (3) use alternate weld filler material. Three 3-foot by 3-foot by 1/2-inch welded plates are being fabricated to study the latter two approaches. Two will be welded with 6010 weld rod commonly used for 1020 steel. One of these will have a 95 Cd - 5 Ag alloy rod spot welded along the weld seams. The third will be welded with Inconel 82 welding rod. All three will be exposed in simulated waste solution containing six molar nitrate.

Evaluation of Corronel 230®

Corrosion test specimens of Corronel 230, a 65 a/o Ni - 35 a/o Cr alloy developed to resist corrosion by hot nitric acid, were tested in typical HAPO environments. The alloy appears to have good resistance to HNO_3-HF solutions and may prove useful in certain 234-5 Building applications. Huey tests and tests in boiling synthetic 1WW solution indicate the alloy is inferior to 304-L stainless steel for use in Purex waste concentrators.

PROCESS CONTROL DEVELOPMENT

Purex 2DW Photometer

A dual filter photometer developed for monitoring uranium concentration in the 2DW stream began operation in March 1962. The instrument has performed quite well and is accepted by CPD personnel as a useful tool in controlling operation of the 2D column.

Turbine Meter Evaluation

The 150 gpm turbine meter under evaluation ceased functioning after 200 hours total running time under intermittent operating conditions. Bearing failure was the direct cause of the malfunction. Probable cause of the rotor bearing failure is the high axial loading experienced by the bearing as the rotor was accelerated from zero to 5000 rpm during approximately 350 on-off cycles. This intermittent operation was deliberately designed to be more severe than a probable plant application. However, failure in this short time indicates that further bearing improvements are needed. The meter has been returned to the vendor for repair and bearing modification. The vendor's proposed modification will decrease the rangeability and sensitivity of the instrument but should substantially improve its operating reliability.

Column Mathematical Models

It was found that the addition of an organic backmixing term to the describing differential equations is needed to obtain an adequate representation of the experimental data from the column. This backmixing parameter becomes important in regions near the flooding curve where the volumetric fraction of organic is quite large. The need for this term indicates the invalidity of neglecting organic backmixing in the usual methods of calculating HTU's for a C-type column.

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REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Precipitation of Plutonium Oxide - Plutonium oxide was precipitated from various melts containing uranyl chloride and europium chloride to test the separation of plutonium from europium in this system. The europium decontamination factors obtained are not as high as those reported for neodymium with promethium tracer, but neither are the uranium decontamination factors. Also, the decontamination factors did not vary as much with changing salt system or oxygen chlorine ratios, leading to the belief that some unknown extraneous effect limited rare earth decontamination in this experiment.

Precipitation of plutonium oxide from a melt containing one weight percent plutonium left about the same concentration of soluble plutonium chloride in the melt as precipitations from a 0.1 weight percent plutonium melt (i.e., 0.012 to 0.03 w/o Pu). This indicates some solubility effect may be limiting the completeness of precipitation of PuO_2 by O_2-Cl_2 mixtures in molten alkali chlorides.

UO_2 Crystal Growth Studies - X-ray diffraction techniques have been used to identify crystal faces on electrodeposited UO_2 and to determine crystal growth patterns. The study of crystals representative of the various stages of growth in the $PbCl_2$-2.5 KCl and the LiCl-KCl systems showed the initial growth pattern to be the same in both melts, with (when dry) rapidly growing 111 faces. In the final stages of growth in $PbCl_2$-2.5 KCl, the crystals developed 100 and 110 faces and became cubes. In LiCl-KCl, 311 and 511 faces developed and the crystals became odd shaped polygons. Crystal growth in NaCl-KCl follows a pattern similar to that in $PbCl_2$-2.5 KCl, with the formation of cubes. In "wet" melts, the development of dendrites apparently proceeds with rapidly growing 210 faces appearing during the entire growth period.

Electrochemistry of Uranium in Molten Chloride Salt Solution - Molten chloride salt solutions (KCl-NaCl at 715 C) of uranyl(V) have been prepared by reacting solid UO_2 with 3×10^{-3} molar uranyl(VI). The reaction was accompanied by a change in solution color (by reflected light) from yellow-orange to brown. An anodic chronopotentiogram showed an oxidation wave which is not seen in either uranium(IV) or uranyl(VI) solutions. A reduction showed only the wave characteristic or the uranyl(V) to UO_2 reduction.

Pilot Plant Electrolyses - Pilot plant electrolyses tests were continued during the month in the 60 w/o KCl - 40 w/o LiCl molten salt system. Four runs (LK-28 - LK-31) were made in an 80 liter quartz crucible to further investigate voltage and impurity effects and the dissolution and melt drying time cycle.

Runs LK-28 and 29 were made in melt previously used for two other runs. In LK-28, 47 pounds of small crystal "walnut type" UO_2 was electrodeposited at 0.76 volts reference potential and a maximum current density of 0.54 amp/in^2. The initial bulk density was 94.4 percent of theoretical and the O/U ratio 2.0071. Analysis of the melt showed the presence of 0.54 w/o nickel which may account for the poor crystal characteristics. The results of the electrolyses showed that nickel impurities produce the same bad effect on crystal characteristics as previously noted with iron. The nickel entered the melt through corrosion of Hastelloy D® and Duranickel® cathodes and hence the use of these metals without further development cannot be tolerated in a process depending on a reusable melt.

A fresh melt was prepared from pure salt and U_3O_8 feed for use in the next two runs. In LK-30, 73 pounds of UO_2 were electrodeposited on a smooth graphite electrode at a reference potential starting at 0.55 and ending at 0.66 volts. (This voltage range was caused by a changing potential drop in the cathode as the current decreased during the electrolysis.) The maximum current density was 0.95 amp/in^2 and current efficiency 75.7 percent. Melt concentration decreased from 30 to 14 w/o uranium during the run. The product was thin and columnar appearing, had a bulk density of 96.8 percent of theoretical and an O/U ratio of 2.0023.

The melt from LK-30 was recharged with U_3O_8 and used for run LK-31. A 44.5 pound deposit was produced at 0.4 to 0.45 volts reference potential at a maximum current density of 0.636 amp/in^2 and current efficiency of 75 percent. The crystal characteristics of the product were excellent and are attributed to the lower reference potential used. The bulk density was 98.6 percent of theoretical, O/U ratio 2.006 and impurities after a preliminary water rinse only 40 ppm Li, 180 ppm K, and 285 ppm C.

Pilot Plant Dissolution Studies - During melt preparation for Run LK-31, a 24 hour dissolution test was made at 600 C to determine what a minimum dissolution and drying cycle might be. A U_3O_8 feed was added in two 30-pound increments at zero and eight hours, and a 20-pound increment at 16 hours. The melt was circulated

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through a gas lift with 4.4 liters/minute of chlorine and an [redacted] ional three liters/minute of chlorine was introduced through a sparge tube providing a total chlorine gas flow of 7.4 liters/minute. A total of 81 pounds, or a four-fold excess of chlorine was used. The initial melt concentration was 14 w/o uranium and the final concentration 28.3 w/o uranium. Drying was carried out with two carbon anodes in place using three liters/minute hydrogen chloride and 4.4 liters/minute chlorine flow at 530 C.

As anticipated, maximum dissolution occurred immediately after each incremental U_3O_8 addition with very little dissolution during the last four hours of each eight-hour period. Dissolution was complete before the end of the 24-hour period. It appears that with properly timed incremental additions the dissolution could be completed in approximately 12 hours. Drying was completed in four hours, and may be completed in two hours with appropriate adjustment of gas flow rates.

Molten Salt Density Measurement - Molten salt density is an important variable in the Salt Cycle Process, and measurement of its variation with uranium concentration may be useful in controlling the electrodeposition process. Density of a liquid can be determined from the apparent weight of a submersed plummet of known volume and weight. Using this principle, an instrumentation system was devised consisting of a quartz plummet, a strain gauge load cell, and a recorder. This system was fabricated, tested, and installed in the Pyrochemical Test Facility for evaluation. The instrument is designed to measure specific gravity in the range 1.5 to 3.0.

Initial calibrations indicate an accuracy of about two percent of full scale, with a reasonable likelihood that one percent accuracy can be obtained. The range and span of density measurements can be adjusted to cover a variety of applications.

Electrode Materials - Successful operation of the Salt Cycle Process requires a non-corroding cathode from which the UO_2 deposit can be removed easily without inclusion of the electrode material in the product. All metallic materials tested to date have corroded preferentially at the liquid gas interface or have produced a corrosion layer at the electrode-deposit interface which was incorporated in the UO_2 product. An electrode composed of Duranickel (which shows good overall resistance but produces a corrosion layer on the deposit) plated below the bath surface with a thin layer of platinum (which corrodes severely at the interface but produces no corrosion layer on the deposit) was tested with encouraging results. The platinum coating adhered well to the Duranickel during deposit removal but there was corrosion of the substrate at flaws in the platinum coating. A 3-mil layer of Al_2O_3 was sprayed on a Duranickel cathode at the liquid gas interface. Corrosion was greatly diminished at the interface although UO_2 deposited on the Al_2O_3 below the interface due to its inherent porosity. Because of thermal shock, part of the Al_2O_3 separated from the electrode after it was removed from the salt bath.

Reactor grade graphite electrodes (sp gr 1.67) have been used in recent large scale runs, but strong adherence of the deposit to the graphite made removal difficult and resulted in 100 to 300 ppm carbon in the UO_2 product. In several pilot plant runs, the removal of the deposit has been facilitated by immersing the hot electrode in cold water. (The thermal shock effect separated the UO_2 from the electrode without apparent damage, although it did not reduce the carbon content of the deposit.) Small-scale tests with non-porous pyrolytic graphite (sp gr 2.18) cathodes have shown that the adherence of the deposit to the electrode is due to the intrinsic surface roughness of reactor grade graphite and not to chemical bonding. Deposits on pyrolytic graphite came off easily, leaving the electrode surface smooth and shiny as before deposition and with apparently no graphite on the UO_2 surface. Future pilot plant tests with pyrolytic graphite are planned.

Materials of Construction - Samples of Ni_3Al, nickel-rich Ni_3Al, NiAl and Ni-12.5 a/o Al-12.5 a/o Cr were exposed for single six-hour periods to chlorine-sparged LiCl-KCl melt at 600 C. All of the Ni-Al alloys corroded at rates of 5 mils/mo or less; the Ni-Al-Cr alloy corroded at 180 mils/mo. In the same melt but with 10 w/o UO_2Cl_2 present, NiAl alloy corroded at 83 mils/mo. Larger castings of the Ni-Al alloys are being prepared to permit studies of heat treating and mechanical properties.

RADIOACTIVE RESIDUE FIXATION

Synthetic Zeolites

The strontium capacity of several synthetic zeolites, estimated at 50 percent breakthrough, was determined with 0.05 M $SrCl_2$ solution at 25 C. A standard flow rate of 0.8 column volumes per minute was used. Results expressed as milli-equivalents strontium per gram of exchanger are as follows: clinoptilolite, 0.64; Linde AW-500, 1.18; Decalso, 1.40; Linde 13X, 2.56; and Linde 4A, 3.38.

Zeolite 4A strontium loading capacities also were determined for several strontium feed compositions. With an influent containing

0.001 M $Sr(NO_3)_2$ + Sr-85 + 0.01 M $NaNO_3$ at pH 6, the capacity was 3.37 meq Sr/g at a flow rate of 0.8 column volumes per minute. A feed solution containing 0.0035 M $Sr(NO_3)_2$ + Sr-85 + 0.0015 M $Ca(NO_3)_2$ + 0.01 M $NaNO_3$ at pH 6 yielded a capacity of 2.49 meq Sr/g. A ten-fold increase in the concentration of all components of the latter solution did not change the strontium capacity of the zeolite.

Nitric acid equilibrium data equivalent to a total of 40 column volumes of 0.15, 0.74, 1.47, 2.94 and 7.35 N HNO_3 solution passed through Linde 4A and 13X were obtained. Zeolite weight loss data indicate that nitric acid concentrations greater than 0.05 N for 4A and 0.1 N for 13X begin to dissolve the aluminum present in the zeolite structure. Since greater than 0.05 N HNO_3 may be required for strontium elution from 600 C calcined 4A, an aluminum separation may be necessary during strontium recovery procedures.

Condensate Treatment

Micro Pilot Plant Run 26 was performed to evaluate the decontamination ability of Amberlite 200 (a sulfonated polystyrene cation resin) in the hydrogen form and to observe the hydraulic characteristics of the bed with the feed acidified to a pH 4, but not prefiltered. The 0.5-liter bed which was used during Run 25 was regenerated with four liters of 8 N HNO_3 for use during Run 26. Over 3000 liters of steam-stripped Purex Tank Farm condensate waste were treated before the run was terminated. Except for Ru-106, all radioisotopes present in the waste were being effectively removed by the bed. The addition of nitric acid to the feed significantly improved the hydraulic characteristics of the bed. The pressure drop per foot of bed depth across the resin column at a flow rate of about 1 gpm/ft^2 increased from 0.1 psi at the beginning of the run to about 0.2 psi at the end of the run. During an equivalent period for Run 25 the pressure drop under the same flow conditions, but without feed acidification, reached about 5.5 psi.

Passage of the effluent from the cation resin during Run 26 through a 0.5-liter mixed resin bed column (Amberlite XE-150) removed Ru-106 from the first 500 liters treated with a decontamination factor greater than 100. The concentration of this isotope in over 100 liters of mixed resin bed effluent was below its MPC_w. Even after the effluent concentration exceeded its MPC_w the isotope was being removed with a decontamination factor of 4 to 8. The increasing concentration of Ru-106 appears to have some relationship with the saturation of the anion resin portion of the mixed resin bed with nitrate ion.

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In-Cell Calciner

Testing of the mock-up of the in-cell equipment has been completed and can be moved into the cell as soon as the prerequisite electrical work is completed.

The power supply to be used with the electrostatic bubble scrubber arrived, and performance was tested during a spray calciner run. Performance was excellent, the decontamination factor across the scrubber exceeding a value of 100 (analytical detection limit). The design is, therefore, believed to be satisfactory, and a more durable but functionally similar linear polyethylene and stainless steel inner assembly is being fabricated to replace the somewhat fragile glass liner used in these tests.

The spray calciner run (during which the above measurements were made) was uneventful except for trouble early in the run with metal chips in the steam line. This difficulty is believed to have been eliminated by thorough purging of the lines and installation of a micro-porous metallic filter on the line. The powder product from the run was melted down in the melt pot while corrosion coupons of 304-L stainless and Ni-O-nel were exposed in the off-gas line. Corrosion rates were 1 mil/month for Ni-O-nel and 30 mil/month for 304-L. On the basis of these observations, and similar ones during pot calcination run, Ni-O-nel is recommended for off-gas line and condenser coil service.

A second pot calciner run was made. The previous run had apparently resulted in foaming since a high concentration of sodium was found in the condensate, corresponding to a decontamination factor of only 16 for sodium. Therefore, a draft tube was built into the pot to minimize foaming by creating a rolling motion in the liquid. This appears to have been effective, since the decontamination factor for sodium in the second run was 6×10^6 to the condensate. Liquid level control was accomplished with thermocouples as before, but a running inventory of the feed and the condensate gave an additional check of the boildown process. Eighteen liters of simulated Purex waste were added to the pot while boiling down in 7-1/2 hours. The pot was cooled (during overnight shut-down) and opened. The draft tube was completely plugged with green salt. However, there was no evidence of foam having reached the off-gas pipe, and the corrosion samples were still bright. Three more hours were required to complete boildown, calcination, and melt-down.

The off-gas rate was generally very low and the particle samplers showed no iron detectable above the blank value. The corrosion

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coupons again showed Ni-O-nel to be superior to 304-L stainless for off-gas line service and corrosion rates were 1 and 2 mils/mo, respectively, based on a 12-hour run. However, only for 1/6 of this time were the extremely corrosive sulfate off-gases coming past the coupons. There is indication (due to coupon placement) that more rapid corrosion results where the metal can become damp from moisture condensation. The pot upon completion of the run was half full of melt. The bulk density was 2.7 g/cc, corresponding to a volume reduction factor of 9.7.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

Evaluation of the airborne geomagnetic survey data is in progress. An earlier study located major anomalies in basalt contours and correlated them with known basalt features. The present study will yield finer detail and be more quantitative. The rectilinear tapes are being converted to graphs showing changes in total magnetic field strength along each flight line. The end result will be a map of the project showing contours of equal magnetic anomaly. Detailed correlation with the known buried basalt surfaces can then be made.

The work on lapse time of flow from a crib to a nearby river was modified since satisfactory results were obtained only when the crib was assumed to be somewhat distant from the river. The necessary expressions have been obtained to permit calculation of arrival time distributions by the modified method and programming is in progress. The solution to this problem is expected to closely approximate the actual flow from a crib to an adjacent river.

Head losses through the internal orifices in the well packer have been determined for various water addition rates. These losses range from 0.1 to 36 centimeters for flows of from 1 to 16.6 gallons/minute. Since the packer is used for in-well permeability measurements, it is necessary to subtract these inherent equipment head losses from the taped potentials to arrive at the true head attributed solely to soil permeability.

The three-fold division of the Ringold Formation into a lower "blue clay", a middle conglomerate, and an upper silt and fine sand member was demonstrated to be an oversimplification or arbitrary division, although useful for general purposes. Beneath 200 West Area only 40 feet of silty clays characteristic of the "blue clays" exist at the base of the Ringold Formation. East of the Columbia River, opposite 100-F Area, more than 400 feet of uniform silts and clays are present in beds stratigraphically equal to all three members of the Formation. Gross lateral variations and discontinuity of beds indicate that the Formation beneath the Hanford Works is more fluviatile than lacustrine in origin. Aquifers and aquicludes thus will be less continuous but more complexly interfingered than if the Ringold Formation were dominantly lacustrine. The correlations now made resolve differences in thickness and nomenclature of the three members at different locations.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Radiation Chemistry

A mechanism of radiation induced hemolysis of human red blood cells was postulated, reduced to an equation, and tested experimentally with striking agreement. It was suggested that the damage to the cell which ultimately results in hemolysis is caused by free radicals, that a cell can undergo numerous reactions with the radicals giving rise to cells visually unchanged but chemically modified, that all the reactions between radical and cell are bimolecular with equal rate constants, and that the time it takes a cell to hemolyze is a function of the number of reactions it has undergone. One parameter, the protection index of the cells, has not been measured as yet and was adjusted to the experimental data. This parameter and further experimental data will be evaluated.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition in Conduits

Initial experiments were performed to establish the magnitude of particle deposition in elbows of tubes and conduits. Some 20 experiments were performed under laminar and turbulent flow conditions to measure the fraction of ZnS (4.2 μ mass median diameter) deposited in a 1/2-inch OD tube bent to a radius of 1-1/2 inches. A definite increase in deposition with velocity was shown; however, at the higher velocities used some re-entrainment caused an apparently lower deposition.

Of considerable significance was the marked effect of the elbow on the pattern and magnitude of deposition in the straight portion of the tube immediately downstream of the elbow. The particles were preferentially deposited along the inside radius of the tube, and a large fraction was deposited in the downstream 3 to 4 feet of straight tube. The deposition was unsymmetrical, occurring largely along the side of the tube nearer to the bend radius center. As much as 44 percent of ZnS entering a bend and passing through four feet of 1/2-inch tubing was deposited. This is far more than expected for an equal length of straight tube, strongly suggesting that the bend has a significant effect on the flow pattern in the downstream portion of the tube. Although equations have been developed by others to predict deposition during laminar flow, the results obtained could not be predicted because the assumptions made in the derivations did not recognize the complex deposition pattern actually observed.

Further experiments are planned to establish deposition parameters for particles deposited in tube elbows and in immediately downstream sections.

W. H. Reas
Manager
Chemical Research and Development

WH Reas:cf

BIOLOGY OPERATION

A. ORGANIZATION AND PERSONNEL

Dr. L. A. George, a Biological Scientist in the Biological Analyses Operation, passed away on May 19.

B. TECHNICAL ACTIVITIES

FISH INCIDENT

On the evening of April 30, nearly our entire population of fish - about 8,000 being held in troughs and ponds containing river water, was wiped out. The cause was sought by IPD and HLO personnel for the following two weeks.

IPD obtained evidence that chlorinated water from the reactor water treatment plant backed down through pipe lines into the intake of the pump supplying water to 146-FR. Conditions believed to exist on April 30 were established, and an experimentally controlled kill of fish in 146-FR qualitatively identical to that on April 30 was obtained.

State and Government fish hatcheries and the University of Washington's Laboratory of Radiation Biology are generously cooperating to restock our laboratory.

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

Test on the effect of reactor effluent on young chinook salmon fingerling was continued at 1706-KE. No adverse growth rate or mortality rate was evident for the 3 per cent treatment level, but the 7 per cent treatment groups are experiencing greater mortality and growth depression.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Phosphorus

Cichlids fed food containing P^{32} levels of 0.25, 1.0, and 4.0 μc/g have shown no differences in mortality attributable to four months of isotope feeding.

Alkaline Earths

Liver and kidney mitochondria were incubated in vitro at 30 C in solutions containing Sr^{85} and Ca^{45} with 0 or 0.9 μM calcium, ATP, cytochrome C and/or magnesium. The kidney mitochondria required ATP for absorption of Ca^{45} and Sr^{85} whether or not calcium was present. Cytochrome C was not necessary. Liver mitochondria in the presence of 0.9 μM calcium also showed this requirement for ATP and indifference to cytochrome C. In the absence of calcium in the medium, the reactions to the absence of cytochrome C or ATP in the medium were quite variable. This variability might well be due to residual amounts of ATP and cytochrome C in the liver being at times adequate to promote the absorption of Ca^{45} and Sr^{85} in small amounts of carrier. Magnesium appears to be required for absorption of Sr^{85} and Ca^{45} by liver and kidney mitochondria in the presence of 0.9 µM calcium but not in its absence.

Strontium

A female miniature pig placed on 625 µc Sr^{90} per day at 9 months of age succumbed 9 months after the initiation of feeding. During the period of daily Sr^{90} feeding this animal conceived, gave birth, and weaned a litter of offspring, which, except for a slight reduction in size, were apparently normal. The animal had exhibited a progressive neutropenia and thrombocytopenia since shortly after initiation of Sr^{90} feeding. Erythrocyte and lymphocyte numbers were reduced in number during the last few months of feeding.

At post-mortem, widespread hemorrhage was observed, which was particularly prominent in the lungs. No gross alterations in the skeleton were noted. The estimated radiation dose rate to portions of the animal's skeleton approached 100 rads per day before death.

Analysis of the Sr^{90} and calcium content of the skeleton of a number of miniature pigs being fed Sr^{90} daily revealed that the younger animals are apparently not discriminating against Sr^{90} relative to calcium to the extent that older animals are. The Sr^{90}/Ca ratio of the skeleton of the newborn is ~0.03 that of the dam's diet, or 0.12 that of the dam's plasma. By weaning, the skeleton Sr^{90}/Ca ratio has increased to ~0.07 that of the dam's diet. However, in the passage of Sr^{90} and calcium from the dam's diet to the milk, the sow discriminates against Sr^{90} relative to calcium by a factor of about 10. Thus the Sr^{90}/Ca ratio of the weanling's skeleton is ~0.7 that of its diet - the milk of the mother. By three months of age, the Sr^{90}/Ca ratio of the animal's skeleton is ~0.25 that of its diet. The skeleton at three months of age has significant portions that were formed prior to weaning. Thus to attain an over-all skeletal Sr^{90}/Ca ratio of 0.25 that of diet, the skeleton of uremia. At post-mortem a severe bilateral nephrosclerosis was noted with apparent detachment of the kidney capsule. The space between the shrunken kidney and capsule was filled with over a liter of dark red fluid.

Radium-226 does not concentrate to any great extent in the kidney, but it decays to Rn^{222} of which appreciable amounts are returned to the circulating blood, it then decays to Po^{218}, which is known to concentrate in the kidney. (It is conceivable that the kidney damage is the result of alpha irradiation from the decay of the short-lived Po^{218}.)

Neptunium Studies

DTPA-treated animals show 56 per cent less Np^{237} than animals receiving 0.89 μc Np^{237} alone.

Pathological examinations of animals receiving Np^{237} two to three days prior to sacrifice indicated a decrease in the mega karyocyte content and fatty degeneration of the liver.

Plutonium

Comparative toxicity of Pu^{239} and Pu^{238} indicated that, given comparable μc doses, Pu^{239} is considerably more toxic than Pu^{238} to rats.

Radioanalysis of selected tissues seven days after intradermal injection of a miniature pig with 1 μc $Pu^{239}(NO_3)_4$ per site suggested that up to 11 per cent was translocated to the regional lymph nodes, about 7 per cent to the liver and 5 per cent to the bone. The amount translocated from the skin of the animal injected with 5 μc Pu^{239}/site was considerably less than that observed in the animal injected with 1 μc/site. The concentration of plutonium in the regional lymph nodes was verified autoradiographically. Histologic examination of these nodes showed a partial to complete necrosis of the lymph nodules (and reticuloendothelial cells filled with cellular debris and a heavy infiltration of granulocytes in the cortical and sinusoidal areas. The most severe damage appeared to be in the hilar area).

Transfer of Radionuclides to Milk

The study of the transfer of Ca^{45}, U^{233}, and Cm^{244} from plasma to milk following a single intravenous administration was completed. Combining these results with earlier studies showed the rate of clearance from plasma (in decreasing order) to be: U^{233}, followed by Cm^{244}, Am^{241}, and Ca^{45} closely grouped together, and then Pu^{239} and Np^{237}. Of special interest was that the plasma concentration of U^{233}, Am^{241} and Cm^{244} showed an initial decline followed by a slight increase in concentration and then a continuous decline for the remainder of the 10-day study. It is suggested that this increased concentration at 7 hours may represent release of the nuclide from an initial site of deposition such as the reticuloendothelial system of liver. In the case of Np^{237} a slightly higher concentration was noted at 2 and 4 hours post-injection than observed at 1 hour.

On the basis of milk to plasma ratios, the radionuclides could be grouped as follows: a) Ca^{45} with ratios of 10 to 40, b) Am^{241} and Cm^{244} with ratios of 2 to 5, and c) Np^{237}, Pu^{239} and U^{233} with ratios of 0.02 to 0.15. Of particular interest was the actual concentrating effect noted for movement of Am^{241} and Cm^{244} from plasma to milk. This implies that these radionuclides are being actively transported from plasma to milk. (The marked variation in these radionuclides studied is in close agreement with what would be anticipated on the basis of their chemical properties, including valence state.)

Radiation Protection Agents

While 10 control rats died between the 7th and 10th day after 900 r, 60 per cent of a similar group receiving 100 mg/kg of Calmagite were alive 42 days later. Tiron at 200 mg/kg had 20 per cent survival while Aluminon at 80 mg/kg had 10 per cent survival on the 42nd day post-irradiation.

Mice given xenogeneic (rat) bone marrow showed an enhanced liver phosphatase which was maximal at 2 weeks. $C31F_1$ mice receiving syngeneic bone marrow showed no significant rise in acid phosphatase. In general, the phosphatase response was not as pronounced in non-x-rayed animals receiving bone marrow from foreign sources (allogeneic and xenogeneic) as in the case of animals similarly treated but with prior 950 r x-ray.

Injection of polyvinyl pyrrolidone (PVP), a biologically stable macromolecular substance, resulted in acid phosphatase levels that were comparable to the high values obtained with allogeneic bone marrow treated mice receiving prior 950 r.

Inhalation Studies

Daily treatment of a dog with DTPA aerosols beginning one week after inhalation of $Ce^{144}O_2$ increased the clearance of Ce^{144}-Pr^{144}. The dogs were monitored routinely with the whole-body counter. At the end of 30 days the controls retained 85 per cent of the estimated alveolar deposition of Ce^{144}-Pr^{144} while the dog treated with DTPA retained less than 40 per cent. These results are not as marked as those obtained in two earlier rat experiments; however, in the case of the rats, treatment with DTPA was started immediately after exposure. The results of a similar experiment with Pu^{239} in dogs are not complete.

Statistical analyses of life span data for mice were completed and indicate that pulmonary deposition in mice of 0.03 to 1.0 nc $Pu^{239}O_2$ did not significantly alter the mean age at the time of death. No lung tumors or other pathology were seen in the mice. The results of this experiment are optimistic, contributing to the often asked question of whether a few radioactive particles deposited in the lung are harmful.

Plutonium excretion data were programmed for curve fitting by the IBM 7090 computer. The results show a significant effect of particle size on excretion of plutonium following inhalation of $Pu^{239}O_2$. The rate of excretion in urine decreased with increasing mass median diameter of the plutonium particles comprising the aerosols over the range 0.23 to 7.6 µ. There were less differences in fecal excretion of plutonium suggesting that better estimates of body burden could be obtained from fecal rather than urinary excretion data.

Developmental Biology

Concentrations of H^3 ranging from 0.05 mc/ml to 0.5 mc/ml delayed hatching of cichlid eggs from 1 to 2 days as compared to controls.

About 10 per cent of cichlid eggs which had been exposed to 30 per cent D_2O hatched by the 7th day following fertilization. Development was impaired and transfer to solutions containing from 0 to 30 per cent D_2O failed to produce a survivor beyond 12 days after fertilization.

Oregon "R" flies were allowed to propagate and produce a sizable population. When equilibrium had been established, a Sr^{90} source, irradiating about 25 rads per hour, was introduced.

After two months, locale of egg deposition still lacks preference with respect to distance from the radiation source. Development of eggs in the irradiated media is greatly impaired and decelerated.

When male flour beetles, T. confusum, were present with females for 1, 2, or 10 days, the reproductive abilities of the females were unaltered. This work is prerequisite for radiation studies designed to determine the incidence and recovery of induced mutations in stored vs. fresh sperm cells.

Plants

Benzimidazole was used to pretreat young intact baby seedlings prior to measuring Rb^{86}-labeled Rb uptake. Concentrations which stimulated uptake of K in excised root studies were injurious to intact plants. Onset and severity of injury was proportional to concentrations used.

Comparison of Cs^{137} and Sr^{85} uptake by bean plants from Sr^{90} plot soil and Cs control soil showed (a) Sr^{85} uptake from both soils surpassed that of Cs^{137} by 100-fold, (b) Sr^{85} uptake was independent of soil type, and (c) Cs^{137} uptake varied with soil type.

Plant Ecology

A small floating aquatic plant, Lemna, was used to determine the ability of this plant to take up carrier-free Cs^{134} from pond water. Uptake of Cs^{134} was proportional to the concentration in pond water rather than to the amount of cesium in the system.

Rattlesnake Springs

Definite chemical changes in the chemical content of Rattlesnake Springs water during the past two months was evident. Especially noticeable was a doubling of the sodium and sulphate concentrations, and a twenty- and tenfold decrease of phosphate and nitrate, respectively. These large decreases are undoubtedly related to assimilation by the increasing autotrophic populations in the water.

HA Kornberg
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BIOLOGY LABORATORY

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C. Lectures

a. Papers Presented at Society Meetings and Symposiums

R. T. O'Brien. Studies on effects of deuterium oxide on yeast. American Society for Microbiology, Kansas City, Mo. May 10, 1962.

R. C. Thompson. Factors and conditions modifying the absorption and retention of chronically ingested radiostrontium. Conference on Transfer of Ca and Sr Across Biological Membranes. Cornell University, Ithaca, N.Y. May 13, 1962.

R. O. McClellan and L. K. Bustad. Transfer of ingested Sr^{90} and calcium to milk of miniature swine. Radiation Research Society Meeting, Colorado Springs, Colo. May 21, 1962.

Symposium on the Biology of the Transuranic Elements. Richland, Washington. May 28-30, 1962.

W. J. Bair. The effect of low levels of inhaled $Pu^{239}O_2$ on the life span of mice.

W. J. Bair. Deposition, retention, translocation and excretion of inhaled plutonium in dogs.

J. E. Ballou. The combined toxic effects of plutonium plus X-irradiation in rats.

J. E. Ballou. Preliminary evaluation of TTHA for plutonium removal.

L. K. Bustad, W. J. Clarke, L. A. George, V. G. Horstman, R. O. McClellan, L. J. Seigneur and J. L. Terry. Preliminary observations on metabolism and toxicity of plutonium in miniature swine.

J. W. Cable, V. G. Horstman, W. J. Clarke, and L. K. Bustad. Effects of intradermally injected plutonium nitrate in swine.

W. J. Clarke. Comparative histopathology of Plutonium-239, Radium-226, and Strontium-90 in Pig Bone.

H. E. Erdman. Effects of ingested plutonium-239 on fecundity, fertility, and life span of Habrobracon (Hymenoptera: Braconidae).

R. O. McClellan, H. W. Casey, and L. K. Bustad. Transfer of some transuranic elements to milk.

J. F. Park. Acute and chronic toxicity of inhaled plutonium in dogs.

R. C. Thompson. Studies with neptunium in the rat.

b. Off-Site and Local Seminars

V. H. Smith. Use of chelating agents in removing internally deposited radicactive materials. Oregon State University, Corvallis, Oregon. May 4, 1962.

c. Seminars (Biology)

B. A. McFadden, Washington State University. Control of the glyoxylate cycle in Pseudomonas indigofera. May 1, 1962.

David Hewitt, University of Washington. The somatic effects of low-level radiation exposure in man. May 3, 1962.

D. Publications

a. Documents (HW)

McClellan, R.O., J. R. McKenney, and L. K. Bustad. Metabolism and dcsimetry of cesium-137 in male sheep. Document HW-72511 (February 1962).

b. Open Literature

None

E. General

The Symposium on the Biology of the Transuranic Elements held May 28-30 in Richland was attended by about 70 non-Hanford scientists, including 10 from foreign countries.

OPERATIONS RESEARCH AND SYNTHESIS OPERATION
MONTHLY REPORT - MAY, 1962

ORGANIZATION AND PERSONNEL

Bertrand B. Field transferred from the Operations Research and Synthesis Consulting Service, New York, to the group as Specialist, Operations Research, effective May 16.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Fuels Preparation Department

Copies of the rough draft of the formal report discussing models which relate dimensional distortion of fuel elements during irradiation to reactor environment have been circulated for comment. It is planned to have the final draft ready for typing by June 8. This report is being coauthored with FPD personnel.

The revisions to the MERCY routine are completed except for debugging. Basic input data will consist of sets of measurements made by different instruments and/or operators on the same items (e.g., fuel elements). The program output will give components of measurement error contributing to the total error. Extensive use will be made of the program in view of FPD's active interest in effectively controlling sources of variation due to measurement errors.

Measurements of clad thickness for NPR fuels are made by eddy current and by autoradiograph techniques. Sets of data, using different operators and covering a period of several days, were submitted to the existing MERCY routine to provide estimates of the measurement errors. Assistance was given in interpreting the program output.

Quarterly rupture data for the last nine quarters were analyzed using the GEORGE routine (generalized regression analysis). These data were frequencies of ruptures adjusted for reactor environment and separated by reactor and enrichment level. The behavior with time was under study.

In measuring irradiated fuel elements, measurements are performed on the jacketed elements. Thus, uniform corrosion will tend to bias bare core diameter growth values. In order to adjust the existing diameter growth models for this uniform corrosion, an existing corrosion model was estimated by a quadratic model, thereby permitting direct addition to the empirical quadratic response surface for diameter growth. It was found that a quadratic surface fits the corrosion model reasonably well in the region of interest.

In 100 percent testing situations, the bases for setting specifications often take into account only the cost to the producer. That is, the specification is set to reject the worst α percent, where α is chosen such that the cost is not "excessive". A document was issued, coauthored with FPD personnel, pointing out that in setting specifications at many fuel element test stations, it is also possible to estimate the cost to the consumer associated with the occurrence of ruptures. This being the case, a logical basis for setting specs is the minimization of producer's and consumer's cost. This basis was used in providing a recommended specification for external total bad discs in the UT-4 tester, as an example. The effect of measurement error was also considered.

Analyses are being made on data from a production test designed to determine the effectiveness of the UT-2 tester in sorting out dimensionally unstable fuel cores. Designed as six replicates of basic Latin squares, the analysis is hampered by the existence of many "80 type" profiles, attributed to measurement error, which should be removed. Analyses will be made both excluding and including these fuel elements. In the former case, it will be impossible to clearly separate the effects.

Data are being analyzed to determine the preheat furnace temperature setting to be used, and the time required to reach uniform temperature in the fuel to be jacketed by the hot die sizing process. The analysis translates the time-dependent thermocouple temperature readings into measures of averages, longitudinal and transverse effects, and within core variation. It also studies how these relationships vary with time.

Two sets of data were analyzed to better estimate the cooling rate for NPR fuels after Beta heat treating. Components of variance were also estimated. The cooling rate determination must be made in order to set specifications on the transfer time such that the ending temperature is within a desired set of limits.

The analysis of clad thickness data from some 100 NPR extrusions was completed, and a report issued. This was performed primarily in order to compare vendors.

As requested, a critique was prepared on a document which reported the results of a study on the effects of brazing variables on the thickness and percent continuity of the Be-Zr braze/uranium compound layer.

Irradition Processing Department

Work continued on the problem of estimating the probability of detecting defects in welded primary piping for the NPR project. A mathematical model of defect size was constructed, and some statistical analyses of data from guided bend tests were made.

An evaluation of a sampling scheme for sampling connector tubing was made. The evaluation was given in terms of confidence intervals on a population parameter, the number of scratches or scores per 27-inch length of tubing sampled. The results were written and sent to the interested person.

Presentation of the IPD Operations Simulation study progress, and analytical report explanations were made during the month to reactor managers, maintenance managers, and reactor analysts. Considerable interest was indicated by all groups regarding the analytical reports developed for the study. The first of a series of these documents has been issued. Since comparative summaries indicated flaws in the source data, they are currently being edited. It is intended that several other analytical reports generated from these data will be issued. Further progress in the original study direction will continue upon completion of the issuing of these control reports.

Assistance was given in developing criteria for classifying process tube wall thickness versus corrosion index curves for different sets of process tubes characterized by reactor and date of installation.

Eight tubes of weight loss data were used to re-estimate parameters in two uniform corrosion models. This was done primarily in order to test the data processing IBM routine. More data will now be included to refine the estimates and determine if the existing models are unbiased.

Chemical Processing Department

In connection with controlling the density of fabricated parts, the relationship between part densities measured directly and those coming from boomerang samples was examined. Although measurement precision, as estimated by duplicate observations, indicated good control of the measurement error for both methods, the correlation between part and boomerang measurements was not too high, indicating that the two methods do not consistently measure the same quantity.

A meeting was held to discuss the technical difficulties which are encountered when design gauge data for manufactured parts are converted from the standard coordinate system to one more convenient for local gauging. The convenience and practical advantages of the proposed system were evaluated against the disadvantage of less accurate error control. Since the investment in the proposed system is relatively small, it was decided to code the gauge data conversion routine for the EDPM so that some experience could be gained. This program is nearing completion and will enter the check-out phase in the near future.

Mathematical assistance was given on constructing a mathematical model for the dissipation of heat in the earth from long cylindrical containers of radioactive materials. Solutions to the model were obtained in terms of tabulated functions.

A partial analysis of medical treatment injuries was completed, and the results were written and sent to the interested person.

Relations Operation

Initial statistical consulting is being given in planning for the next attitude survey to be conducted at Hanford.

Further salary curves were fit as requested. In addition, it was shown that a quadratic fit on the logarithms of salary data provided better fits in those instances where the cubic fits on the raw data did not yield randomness of residuals. This lack of randomness is common when fitting upper decile and quartile data. The logarithmic transformation tends to stabilize the variance along the curve.

Contract and Accounting Operation

Work was begun on the calculation of some probabilities associated with an information retrieval system. Preliminary estimates of some coincidence probabilities were calculated.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HLO

2000 Program

Pulse Column Test Facility

Statistical analysis was begun of a time series of mid-column photometer aqueous uranium concentration data to estimate the random variability of uranium concentration at a fixed point in the column during equilibrium operation. The results of this study will be used as standard information in the forthcoming flooding curve experiments. Flooding is preceded by increasing instability; so, stability can be used as an index to locate the flooding curve.

UO_2 Fuel Density

Density and homogeneity tests on actual fuel elements whose particle sizes and size distributions had been predicted by theoretical considerations confirmed their exceptional quality. Additional calculations are now being made for annular and nested annular elements.

General

Revisions and additional features are being made to the EDPM program which calculates the longevity and sensitivity of a proposed neutron detector as a function of its isotopic constituents.

Discussions were held on the subject of creating a mathematical model of the concentration and precipitation phase of a proposed piece of calcining equipment.

Calculations were completed on the weight specifications of numerous odd-geometry castings which form the components of a proposed materials test cell.

<u>3000 Program</u>

Rotary Lathe Control

Document HW-73537, "A Theory of Control for δ-ω Rotary Lathes," has been issued. This document develops the appropriate mathematical relationships which exist between the polar coordinate design specifications of a desired surface of revolution and the controlling, positioning and timing variables of a δ-ω lathe on which it is to be machined.

An automatic timing and speed monitoring routine is in the process of being incorporated into the 7090 program which generates the magnetic tapes which serve as input to the prototype δ-ω lathe control system. It is planned to generate a special tape for testing purposes as soon as the control components arrive from the Manufacturing Services Laboratories and are assembled at HAPO.

<u>4000 Program</u>

Waste Fixation

Statistical analysis was completed of data from a recent pilot study mineral bed waste column experiment. The results of the analysis indicate the effect of three independent variables, column residence time, column particle size, and waste dilution, on the waste removal characteristics of the column. The

[redacted] [redacted]

specific indices correlated with these independent variables in the statistical evaluation were the capacity of the column (column volumes of waste to 50 percent breakthrough) and the number of theoretical plates (a function related to the column saturation rate).

Fuels Research

Further work was done on the precision and accuracy studies of uranium and plutonium analysis methods. Battelle Memorial Institute reactivity measurements on uranium and plutonium fuel samples fabricated by Plutonium Fuels Development are being correlated with Analytical Laboratories uranium and plutonium analyses to determine whether or not there is a significant bias between the two laboratories. As further justification for switching from Quinalizarin method to the Azure C method for estimating trace boron content in the fuel samples, an experiment was designed to measure the within day and between day experimental errors in the Azure C method. The experiment was completed and the analysis of the data is in progress.

A frequency distribution of groove depths, caused by brackets which hold the PRTR fuel elements, was fit by a Weibull distribution in order to estimate the percentage of groove depths in the reactor exceeding a specified amount.

5000 Program

Actinide Element Research

The final version of a FORTRAN program to index cubic crystals is being debugged. This program is an improvement over earlier versions in that it reduces the input to an absolute minimum number of program parameters plus experimental data. A first attempt at programming a hexagonal crystal indexing program is currently under way. A FORTRAN program is being prepared to solve the related problem of determining the 90 degree lattice constant of indexed cubic crystals.

Division of Research Programs

In connection with the Division of Research programs, a data processing workshop was held at Hanford on May 16 and 17. The meetings were attended by several Hanford personnel involved in data generation and evaluation and persons functioning in similar positions at other AEC sites.

The activation analysis background study was completed. A procedure for graphical presentation of unanalyzed samples to help establish priorities for analyzing samples was devised and all unanalyzed samples processed with it. Three mathematical tables were calculated in connection with the report, "Fixed Time Estimation of Counting Rates with Background Corrections".

A new program, GEM, has been written to replace the SPEC program for the quantitative resolution of a time dependent gamma ray spectrum. The new program has increased capabilities and uses only about 50 percent of the memory required by the previous program. Debugging of this code is nearly complete. A new verion of the ZERO program which provides for estimation of the time dependence of the data was run successfully on several sets of data. Weekly meetings are being held with EDPO personnel to further define the IRA II file.

6000 Program

Biology and Medicine

Work continued on fitting a multicompartment model to data from a retention study on fish. A digital computer program is now being written to solve an associated system of differential equations.

A statistical analysis of data from an experiment to determine the effect of plutonium inhalation on the longevity of mice was completed. The results were written and sent to the interested person.

Other

The FORTRAN language program for routine analysis of reference system calibration data supplied by Instrument Research and Development Operation has been completed. A statistical analysis was begun of several calibration functions fitted by this program to determine the effect of temperature on a linear differential transducer.

Six fourth order matrices expressing standard spectra for the quantitative estimation of a four isotope mixture into individual peak region counting rates were inverted.

A. D. Wiggins attended the Advanced Seminar in the Statistical Theory of Reliability conducted by the U. S. Army Mathematics Research Center at the University of Wisconsin.

W. L. Nicholson

Acting for
Manager
Operations Research and Synthesis

WL Nicholson:dgl

[redacted]

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Code Development

Activities this month have centered upon adjustment of existing codes and development of adjunct codes so as to facilitate completion of the survey portions of the HLO fuel cycle analysis work. The survey work uses MELEAGER as the basic burn-up code and the major effort has been to adjust certain parameters in MELEAGER so that the results of MELEAGER computations will be in line with the results from more precise and time consuming computational codes as well as in line with currently available experimental data. It was possible to make some substantial improvements in the MELEAGER formulation because of the availability of codes based upon experimental neutron thermalization data. In addition, two additional reactor physics codes designed to reflect the reactor geometry with MELEAGER were completed and are being debugged. The first code, JASON, supplies lattice cell dimensionality for MELEAGER, and the second code will supply over-all reactor dimensionality. By use of these codes with MELEAGER, the major effects of the reactor can be taken into account, such as the relative location of coolant, fuel, jacketing, and moderator composing the lattice (JASON) and variously loaded regions of the reactor including the reflector and control rod regions. For many survey studies, the detail afforded by the foregoing codes is unnecessary and the basic MELEAGER code can be most economically used for these.

Several codes have been prepared to facilitate preparation of cases and handling of the data. The code, "DATA MAKER" was completed and applied. This code prepares input for MELEAGER CHAIN. A similar, but more sophisticated code known as "SUPERCASE GENERATOR" was also completed. This code is used to prepare input for MELEAGER, as does DATA MAKER, but is able to alter the enrichment levels so as to space the points in the range of interest to provide optimum selection of the minimum fuel cost conditions.

A test case was successfully completed on the new FEFJ code. This code will be applied to compute the cost of fuel element fabrication and jacketing from estimated labor, capital, and yields for discrete steps in the jacketing processes. The code treats reject from any step as waste or as rework to any previous step and also accounts for fissile material inventory build-up throughout the process. The code is designed to apply, not only to fuel element fabrication, but to any fabrication process. The process to be studied is broken down into individual steps with one main sequence as the primary line. As many concurrent parallel steps and recycle loops can be used as needed to properly represent the system. For each step, individual cost factors may be entered as direct material, labor cost, process time, equipment cost, equipment rental, floor space, electrical power, depreciation factors, and others. Costs can be entered in all or any number of these categories depending upon the best data source for any step.

Expenses that are best known on an annual basis are entered as such, along with interest rates and annual production. Once the plant size and costs are established for a given processing system, the importance of each step can be assesed by simply varying the yields and cost appropriately.

Burn-up experiments with the MELEAGER code for a broader range of fuels have been made possible by a careful calibration of the neutron reaction model used in MELEAGER code with SPECTRUM V (See HW-71953), a 75-energy group code. Accordingly, modifications have been made to the flux spectrum model used in MELEAGER and the selfshielding of the cross sections have been suitably adjusted to give reasonable results, even with considerable amounts of plutonium enrichment in closely spaced lattices.

Since MELEAGER and SPECTRUM codes arrive at effective cross sections by entirely different routes, it is not possible to make a direct comparison of the absolute magnitudes of cross sections between the two codes. It is, however, possible to compare the relative cross section ratios between the two codes for various isotope pairs. Fortunately, the results from a burn-up are affected much less by errors in the magnitude of cross sections than by errors in the relative values of the cross sections. In other words, if one were to reduce the cross section of all isotopes by 50 percent, the net effect would be merely to increase the neutron flux to maintain the specified power with the exception of those isotope sequences involving radioactive decay. Fuel comparisons between MELEAGER and SPECTRUM are reasonably valid only for isotopes whose major resonances lie below 4.2 ev. — such as the U-235 and the plutonium isotopes. The accompanying table compares the results of the calibrated MELEAGER with the results from SPECTRUM V which is an adaptation of the original SPECTRUM code for this work.

The calibrations were made using neutron temperatures supplied from SPECTRUM code. Earlier work by Dawson and Steves (being published) established that the neutron temperature could be related to the moderator temperature and cell blackness. The MELEAGER code now contains the following expression which is similar to that suggested by Dawson and Steves:

$$T_n + 273.1 = (T_m + 273.1)\left(1 + \frac{K_3 \left(\sum_o \sigma_{oa}^{i} \; Y^i\right)\left(RTM\right)}{(TNL)(SDPV)}\right.$$

where:

T_n is the neutron temperature $^{\circ}C$

T_m is the moderator temperature

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TABLE I

COMPARISON OF MODIFIED* MELEAGER REACTION RATES WITH SPECTRUM V REACTION RATES FOR VARYING ENRICHMENT IN HOMOGENEOUS 260 °C H_2O

Assumed Isotope Concentrations, grams/cc:					
U-235	0.028	0.056	0.140	0.28	0.56
Pu-239	0.028	0.056	0.140	0.28	0.56
Pu-240	0.012	0.024	0.060	0.12	0.24
Computed Spectral Index:					
Modified MELEAGER	0.1465	0.251	0.491	0.672	0.103
SPECTRUM V	0.1405	0.243	0.458	0.623	0.730
Computed Reaction Rate Ratio:					
Pu-239 to U-235 per atom present					
Modified MELEAGER	2.822	2.856	2.46	1.65	--
SPECTRUM V	2.902	3.124	3.39	3.35	2.97
Computed Reaction Rate Ratio:					
Pu-240 to Pu-239 per atom present					
Modified MELEAGER	0.717	0.883	1.339	2.15	--
SPECTRUM V	0.6534	0.782	1.052	1.88	1.736
Computed Pu-239 Alpha:					
Modified MELEAGER	0.548	0.561	0.584	0.581	--
SPECTRUM V	0.5515	0.565	0.581	0.581	0.5648

*The modifications employed here in MELEAGER constitute a new MELEAGER deck to be known hereafter as "Post-June 1, 1962 MELEAGER."

K_3 is a suitably adjusted constant ^{o}C

$\sigma^{i}_{oa}\ Y^{i}$ = the 2200 m/s macroscopic cross section

RTM = Fortran variable expression $= \sqrt{\dfrac{\pi\ 293}{4\ \ T_n}}$

TNL = thermal nonleakage probability

SDPV = slowing down power of the moderator normalized to the fuel.

Computations show that this expression gives excellent results for plutonium mixtures for typical power reactor moderator temperatures at a value of $K_3 = 0.7$. For moderator temperatures widely different, it is expected that the other values of K_3 will give better results.

As was mentioned in the previous monthly report, neutron absorption in Am-241 has a very marked impact on the lifetimes of plutonium enriched fuels involving significant amounts of Pu-241 (Pu-241 has a 13-year half life to decay to Am-241). The ratio of Am-241 cross section to Pu-239 was derived from MELEAGER and SPECTRUM experiments and was found to be much higher in MELEAGER, especially for a hard spectra (r values greater than 0.1). The MELEAGER values of the $\frac{\text{Am-241}\sigma}{\text{Pu-239}\sigma}$ were approximately 50 percent higher than those of Spectrum at spectral "r" values of 0.5, and 25 percent higher than those of Spectrum at spectral "r" values of 0.2. With adjusted cross sections, the error in MELEAGER at "r" = 0.2 is negligible and only about 15 percent higher at "r" values of 0.5. This change in cross section should have the effect of reducing the impact of low specific power on plutonium value as the relative probability of decaying Pu-241 to Am-241 is much greater the lower the neutron flux. This is supported by preliminary analysis of comparable Phoenix fuel computations utilizing the revised Am-241 cross section and the nonrevised models.

Fuel Reuse Cycle

"Fuel reuse" is defined as the consecutive irradiation of fuel assemblies or components in different reactor regimes, without intermediate chemical reprocessing. The aspect of fuel reuse currently under investigation involves the operation of a thermal reactor with fuel elements previously irradiated in the blanket of a fast reactor.

The fast reactor has a PuO_2-UO_2 core and a depleted UO_2 blanket. The companion thermal machine studied so far is a D_2O cooled and moderated, pressure tube reactor. Both machines operate on a constant over-all power graded cycle.

The principal effort, to date, has been to evaluate the economic incentives for developing this fuel reuse cycle. An attempt is being made to conduct the analysis in such manner as to eliminate immediate necessity of delving into the economics of the fast reactor. The method used is to compare total fuel costs for the fuel reuse cycle with fuel costs for the conventional plutonium fueling from which it is assumed that purchased (leased) plutonium is mixed with tails uranium (0.40% U-235), fabricated into fuel elements, irradiated to a reactivity-limited exposure, chemically reprocessed, and the recovered plutonium returned (sold). In the fuel reuse cycle it is assumed that tails uranium (0.40% U-235) is fabricated into fuel elements, irradiated in the fast blanket to various degrees of plutonium enrichment, reirradiated (reused) in the thermal reactor to a reactivity-limited exposure, and either returned for further "enrichment" to the fast blanket and further burn-up in the thermal reactor or chemically reprocessed and the recovered plutonium sold.

For conventional plutonium cycle, the total fuel costs must include: fabrication, shipping, and reprocessing; interest on fuel value and on investments in fabrication and shipping; and burn-up and losses of fuel materials. For the fuel reuse cycle, the fuel fabrication costs for the fast reactor blanket rods and the thermal reactor fuel elements are combined. The costs for transfer of fuel must be added as needed.

The fuel cost vs. exposure (enrichment) curve with standard plutonium fueling for the system described above exhibits a mathematical minimum at fairly high exposures. Even, with the low total charge of $30/lb for fabrication, reprocessing, and shipping, the mathematical fuel cost minimum of 0.9+ mill/kwh occurs at 30,000 to 40,000 MWD/T. Conversely, for the fuel reuse cycle, minimum fuel costs occur at quite low exposures because the apparent jacketing costs are less. For zero incremental fabrication and transfer charges, the total fuel cost increases from —0.15 mill/kwh at 1000 MWD/T to +0.6 mill/kwh at 30,000 MWD/T. For a $1/lb incremental charge, a minimum fuel cost of 0.1+ mill/kwh occurs at 2000 MWD/T. Similarly, for a $2/lb cost increment, the minimum fuel cost is 0.2+ mill/kwh at 3000 to 4000 MWD/T. At higher burn-ups, the fuel costs for the $1 and $2/lb incremental cost cases approach the zero increment case—lying within 0.05 mill/kwh at exposures of 15,000 to 30,000 MWD/T. Unless the reactors are very closely located with highly integrated fueling systems, transfer cost increments of $10/lb may be more typical. Fuel costs for repeated passes (of the fuel reuse cycle) appear to be somewhat more costly than for the first pass.

The relatively low fuel costs computed for the reuse cycle result from the current limitation of the study to a D_2O cooled and moderated reactor as well as to the accounting method assumed in which only depletion and interest on fuel value are charged against power generation in the thermal reactor. The extremely low fuel costs (—0.15 to +0.25 mill/kwh) computed for the lowest fuel exposures

(1000 to 5000 MWD/T) result from the facts that the total fuel enrichment is very low ($\sim$natural), that a substantial fraction ($\sim$half) of the enrichment is residual U-235, and that the cost of using and burning depleted uranium is negligible. The negative fuel costs result from burning small amounts of depleted uranium costing \$3/kg, and making small net amounts of plutonium which may be sold for \$10/gm. This U-235 is present in smaller or even negligible concentrations during subsequent reuse passes.

Increased fuel costs at fuel exposures and for repeated passes of the reuse cycle result principally from the fact that larger quantities of (relatively) expensive plutonium are required for reactivity and for burn-up; the increased concentrations of fission products (from both reactors) have lesser effects.

It appears that application of fuel reuse will reduce fuel cycle costs significantly. Specifically, based on preliminary results, it appears that if the thermal reactor is located near the fast reactor (minimum fuel transfer costs) and if both reactors have provisions for rapid and frequent refueling (required for low exposures on graded cycle), the fuel cycle costs of the thermal reactor applying fuel reuse may be low enough to easily compensate for the increased capital costs involved. These savings may not be as great when other thermal reactor types are considered.

Chas. A. Rohrmann

C. A. Rohrmann, Acting Manager
Programming

CAR:pc
6-15-62

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF MAY 1962

A. ORGANIZATION AND PERSONNEL

Transfers within the Section during the month included David N. Brady transferring from Radiation Monitoring to Internal Dosimetry, and Iral C. Nelson transferring from Environmental Studies and Evaluation to Internal Dosimetry, both effective May 1. R. Wayne Meisinger was transferred from Technical Administration Operation to Environmental Studies and Evaluation, and Thomas M. Beasley was transferred from Internal Dosimetry Operation to Chemical Research and Development, both effective May 1. New-hires into the External Dosimetry Operation during the month included G. B. Lage on May 7, S. B. Bridge on May 7, and R. W. Kron on May 16. Lee A. Bond joined the Internal Dosimetry Operation effective May 18. R. Doug Tillson was transferred from External Dosimetry to Technical Administration effective May 15. Wendall A. Briggs transferred to Fuels Development Operation from Internal Dosimetry effective May 21.

B. ACTIVITIES

Occupational Exposure Experience

There were no new cases of plutonium deposition confirmed by bioassay analyses during May. The total number of plutonium deposition cases that have occurred at Hanford is 288, of which 207 are currently employed.

On May 31 two CPD process operators received general hair and facial plutonium contamination to > 40,000 d/m when a glove rupture occurred in the 234-5 Building. Nasal irrigation was promptly performed. Skin decontamination was successful within two hours. Prompt analysis of aliquots from the first eight-hour urine samples showed that internal deposition may have occurred; complete collection of urine and feces was initiated.

A CPD process operator received a plutonium nitrate contaminated injury on May 29 while working in the Redox greenhouse ion exchange contactor area. Excision of tissue at the wound site was performed at Kadlec Hospital about one hour after the injury. Examination with the plutonium wound counter showed about 0.03 μc in the excised tissue, and 0.003 μc remaining at the wound site. This employee is a previous deposition case estimated to be < 1% of the maximum permissible body burden. Urine sampling was initiated to determine the magnitude of the body burden that may result from this injury.

Six other plutonium incidents involving potential inhalation or absorption of plutonium for 12 employees were reported for CPD facilities. Five other plutonium contamination incidents were reported involving possible inhalation for 14 employees in HLO facilities. Special urine sampling was initiated for all involved employees.

Air contamination resulted in a Purex analytical laboratory when a lead sample carrier containing a vial of concentrated strontium solution was removed from the Purex sample gallery without proper survey. Trace nasal contamination was detected for five employees and urine samples were requested for analysis.

An HLO Development and Corrosion Chemistry employee received a localized skin dose of about 30 rads to a small area on his thigh in a fission product contamination incident at the 222-S Building. Failure to vent a cask containing Purex 1-WW waste solution resulted in a pressure release and ejection of some of the solution. A dose estimate was obtained from autoradiographic examination of contaminated spots of the employee's coveralls.

Low-level strontium-90 contamination was spread to the hallways of the 141-F Building in conjunction with the feeding and care of an animal that was on an extremely high diet of this radionuclide. All spots of contamination were marked and cleaned to a nonsmearable condition. The large metabolism cage that housed swine receiving 15 millicuries of strontium-90 per day was successfully packaged and moved to the 200 West Area for decontamination. General surface contamination to 5 rads/hour at 6 inches was observed on the inner surfaces of the cage.

Plutonium contamination of 100,000 d/m in room 400, and 5000 d/m in the immediate hallway at the 325 Building occurred after a routine waste bag change. Failure to observe established follow-up techniques and checkout procedures caused the contaminant to extend beyond the immediate incident area. Although there were no cases of skin contamination detected, 11 cases of shoe contamination were associated with the incident.

Thorium work commenced in the 306 Building where thorium metal was pickled in the wet chemistry facility, butt-welded in the electron beam welding vacuum furnace, and reduced to an ingot in an arc-melt furnace. Air samples to 2.7×10^{-12} μc thorium/cc were collected during these operations.

A three-week scheduled outage at PRTR involving design changes, reactor and system maintenance and inspection was completed with no unusual radiation occurrences. Tritium exposures increased during the month with 44 employees receiving doses greater than 25 mrems from internally deposited tritium. The maximum accrued dose during the month was 270 mrems. Radiation levels to 2.5 r/hour were measured during process tube inspection in the storage basin. The maximum air sample taken was 4×10^{-6} μc FP/cc in A cell immediately following a reactor shutdown.

One hundred eighty-four Hanford Criticality Dosimeters were distributed and installed throughout the Plant during May. The number of dosimeters required and the location of each dosimeter was determined from a set of dosimeter placement criteria. An audit program has been established to review the location and condition of each dosimeter semi-annually.

Environmental Experience

Iodine-131 emitted from the Purex stack returned to normal (< 0.2 curies per day) after last month's high values of 44 curies in 10 days.

The sporadic increases in concentrations of air-borne fallout materials noted for April did not continue into May. The spring peak of materials generated in the 1961 USSR tests was somewhat lower in the Pacific Northwest than predicted. No positive evidence of fallout from the 1962 US tests has been found to date at Hanford. The average concentration in air was 3 $\mu\mu c/m^3$ during May in contrast to the averages of 5 to 6 $\mu\mu c/m^3$ for the first four months of this year.

A total of 108 fish were taken from the Columbia River from sampling locations at Priest Rapids, Hanford, Ringold, Richland, Burbank, and McNary Dam. One hundred sixteen tissue samples were taken from these fish and submitted for radiochemical analysis.

Ninety-seven produce samples were obtained for analysis. These include milk from the Ringold, Riverview, Benton City, Mesa, and Eltopia areas as well as composite samples from the Twin City Creamery. Milk samples totaled 120 gallons; 4 pounds Willapa Bay oysters, 4 pounds ground round steak, 40 pounds pasture grass, and 42 sets of beef thyroids were also obtained.

One aerial monitoring flight was made over the project and a second was conducted over the Columbia River from Priest Rapids to Portland. Twenty-seven flights were made over the Columbia River between Priest Rapids and McNary Dam as part of the fishing pressure study.

Columbia River flow rates exceeded 150,000 cfs. No significant changes in river pollution or temperature were observed. Seasonal increases in manganese-56 and sodium-24 transport rates persisted through the month.

Studies and Improvements

Filter samples of the ventilation exhaust during thorium reprocessing at the 306 Building were taken. Laboratory analysis of the samples is in progress.

The new personnel film badge dosimeter processing machine was operated satisfactorily during the month and delivered to the 3705 Building. Samples of the new personnel film badge dosimeters were supplied by the vendor for approval prior to the production run. Nominal modifications were requested, and accepted by the vendor.

Calibration jigs for providing radium-gamma exposure calibrations in the new personnel film badge dosimeters were designed and fabricated. A mechanized beta radiation calibration jig for the new dosimeters was also fabricated and is now ready for installation in the Calibrations building.

One week of positive ion accelerator time was available to RPO during May. The silicon diode neutron dosimeters recently purchased were studied to compare their sensitivity with diodes previously made at Batelle Memorial Institute under a Hanford contract. The new diodes proved to be less sensitive to neutron radiations than the former ones. Further consideration of neutron dosimetry through the application of solid state silicon diodes led to the prediction that n-tpye semi-conductor material may prove to be more sensitive to neutron damage than the currently used p-type material. The purchase of additional diodes fabricated of n-type material is planned to substantiate this prediction.

Air sampling calibration equipment was designed, constructed, and calibrated to provide accurate flow measurements for in-line duct air samplers and for the open-face 4" x 8" air sample heads. During the development and field testing of this air sample flow calibration equipment it was observed that the gaskets on many of the in-line duct air samplers should be replaced. The thin replacement gaskets on several of the samplers examined were allowing as much as 30% leakage around the door, thus diluting exhaust samples with clean air.

C. VISITORS

Visitors consulting with members of the Radiation Protection staff during the month included:

J. D. McLendon - Union Carbide Nuclear Company, Oak Ridge, Tennessee
G. L. Toombs - - Oregon State Board of Health, Salem, Oregon
Eiji Inaba - - - Nippon Atomic Industry Group Co., Ltd., Tokyo, Japan
D. C. Nichols - Savannah River Plant, E.I. du Pont de Nemours, Aiken, South Carolina
J. R. Muir - - - Oak Ridge National Laboratory, Oak Ridge, Tennessee
Ralph Baltzo - - University of Washington, Seattle, Washington
B.A.J. Lister)
K. Stuart) United Kingdom Atomic Energy Authority
D.B.B. Jannisch)

Visitors who toured the Whole Body Counter and the Bioassay Laboratory during the month included:

31 Walla Walla Science Class Seniors from Walla Walla College
8 Jesuit Priests from the Seminar of Philosophy and Science, Spokane
25 Purchasing Agents from the Purchasing Agents Association, Spokane and Inland Empire
12 Persons attending the Symposium on the Biology of the Transuranic Elements
P. E. Brown - Oak Ridge National Laboratory, Oak Ridge, Tennessee

C. W. Werner - Medical Service Corp., U.S. Naval Radio. Def., San Francisco, California
G. W. Gaffney - U.S. Public Health Service, Washington, D. C.
R. C. Baker - - Carbide & Carbon Nuclear, Paducah, Kentucky
R. Conti - - - AEC, Chicago Operations Office, Argonne, Illinois
J. Sedlet - - - Argonne National Laboratory, Argonne, Illinois
E. L. Ray - - - Dow Chemical Company, Rocky Flats, Colorado
A. S. Goldin - U.S. Public Health Service, Winchester, Massachusetts

Members of the Radiation Protection Operation staff visiting off-site during the month included:

A. R. Keene - Lawrence Radiation Laboratory, Berkeley, California, to conduct ASA N2.2 Subcommittee meeting.
- Colorado Springs, Colorado, to conduct NCRP Subcommittee 7 meeting.

L. C. Rouse - Lawrence Radiation Laboratory, Berkeley, California, to discuss radiation records systems.
J. R. Bovingdon - Umatilla Ordnance Depot, Umatilla, Oregon, to accompany display for Armed Forces Day.
R. F. Foster - University of California, Berkeley, California, to present an invited lecture.
- Seattle, Washington, to participate in meeting of Pacific Northwest Pollution Control Association.

L. F. Kocher - Product Engineering Co., Portland, Oregon, to discuss design specifications on new personnel dosimeters.
H. V. Larson - University of Washington, Seattle, Washington, to teach AEC Health Physics Fellowship students.

D. RELATIONS

A talk on radioactivity in the Columbia River and in Pasco sanitary water was given by R. B. Hall to the Third Inland Empire Short School on Filter Plant Operation.

A talk on Fallout was given by K. R. Heid at the Connell school for professional educators who belong to the Franklin County Education Association.

Radiation orientations were presented on four occasions to employees. Five 2-hour orientation talks were presented to Biology Research, Nuclear Physics Research, and Maintenance employees. A talk and demonstration on personnel surveys was given to the Biology Animal Farm personnel. A discussion on self-reading dosimeters was held with Firemen in the 200 West Area.

Six suggestions were submitted by personnel of the Radiation Protection Operation during May. One suggestion was adopted and six were rejected. Two suggestions are pending evaluation.

E. SIGNIFICANT REPORTS

HW-71999 - "Evaluation of Radiological Conditions in the Vicinity of Hanford for 1961" by R. F. Foster and Staff.

HW-72691-4-"Summary of Radiological Data for the Month of April, 1962" by R. F. Foster.

HW-73654 - "Contamination Control at the Hanford Laundry" by C. E. Linderoth and G. A. Little.

HW-73672 - "Dispersion of 300 Area Liquid Effluent in the Columbia River" by G. E. Backman.

HW-73898 - "Monthly Report - May 1962, Radiation Monitoring Operation" by A. J. Stevens.

"Gas Loop Facility Hazards Analysis" by L. D. Williams (undocumented).

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

	May	1962 to Date
External Exposure Above Permissible Limits		
Whole Body Penetrating	0	3
Whole Body Skin	0	3
Extremity	0	2
Hanford Pocket Dosimeters		
Dosimeters Processed	3,474	16,842
Paired Results - 100-280 mr	8	30
Paired Results - Over 280 mr	0	3
Lost Results	0	0
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	9,648	47,980
Results - 100-300 mrads	237	1,606
Results - 300-500 mrads	18	160
Results - Over 500 mrads	9	68
Lost Results	28	128
Average Dose Per Film Packet - mrad (ow)	14.75	11.54
- mr (s)	27.79	27.93
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	1,346	7,272
Results - 50-100 mrem	1	5
Results - 100-300 mrem	2	4
Results - Over 300 mrem	2	2
Lost Results	2	10
Fast Neutron		
Film Processed	347	1,997
Results - 50-100 mrem	55	305
Results - 100-300 mrem	15	340
Results - Over 300 mrem	8	11
Lost Results	0	0
Hand Checks		
Checks Taken - Alpha	17,992	150,095
- Beta-Gamma	53,567	262,902
Skin Contamination		
Plutonium	19	90
Fission Products	37	218
Uranium	0	11
Tritium	0	0

Whole Body Counter	Male	Female	May	1962 to Date
GE Employees				
Routine	2	0	2	67
Special	7	0	7	126
Terminal	1	0	1	48
Non-Routine	25	4	29	139
Non-Employees	4	0	4	11
Pre-Employment	0	0	0	3
	39	4	43	394

Bioassay	May	1962 to Date
Confirmed Plutonium Deposition Cases	0	5*
Plutonium - Samples Assayed	499	2,256
- Results Above 2.2×10^{-8} μc/Sample	24	98
Fission Product - Samples Assayed	513	2,931
- Results Above 3.1×10^{-5} μc/Sample	1	15
Uranium - Samples Assayed	197	965
Biological - Samples Assayed	61	233
Strontium - Samples Assayed	0	299

Uranium Analyses

	Following Exposure Units of 10^{-9} μc U/cc			Following Period of No Exposure Units of 10^{-9} μc U/cc		
Sample Description	Maximum	Average	Number Samples	Maximum	Average	Number Samples
Fuels Preparation	9.1	2.1	47	7.3	1.8	46
Fuels Preparation**	0	0	0	0	0	0
Hanford Laboratories	13.3	3.4	43	10.2	2.8	40
Hanford Laboratories**	0	0	0	0	0	0
Chemical Processing	0	0	0	0	0	0
Chemical Processing**	0	0	0	0	0	0
Special Incidents	0	0	0	0	0	0
Random	3.1	1.5	21	0	0	0

Tritium Samples	Maximum	Count	Total
Urine Samples			
> 5.0 μc/l	39.9	262	
< 1.0 μc/l		30	
Samples Assayed			361
D_2O Samples			
Moderator	823.0 μc/ml	6	
Primary	244.3 μc/ml	6	
Reflector	645.5 μc/ml	6	
			18
Other Water Samples			
No. 6226 "E"	223.5 μc/ml		116
			495

* The total number of plutonium deposition cases which have occurred at Hanford is now 288, of which 207 are currently employed.

** Samples taken prior to and after a specific job during work week.

Calibrations	Number of Units Calibrated May	1962 to Date
Portable Instruments		
CP Meter	1,019	5,031
Juno	279	1,359
GM	570	2,788
Other	210	1,014
Audits	110	527
	2,188	10,719
Personnel Meters		
Badge Film	1,620	8,644
Pencils	--	12,670
Other	344	2,237
	1,964	23,551
Miscellaneous Special Services	722	5,421
Total Number of Calibrations	4,874	39,691

Carl M. Unruh

For the Manager
RADIATION PROTECTION

AR Keene:CMU:ljw

FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

Special requests were established during the month as shown below:

Accounting Code	Description
.3C	R. S. Paul - Consulting for International Atomic Energy Commission in Japan. He will conduct lectures and discussions on nondestructive inspection techniques and provide advice on inspections of fuel elements to the Japan Atomic Fuel Corporation and the Japan Atomic Energy Research Institute for a period of one month beginning prior to June 30, 1962. Salary, continuity of service, travel and subsistence will be billed to the AEC.
.4X	Summer Institute in Nuclear Engineering for Engineering Faculty. Hanford Laboratories and the University of Washington Graduate Center will collaborate in offering an Institute in Richland, July 30 to August 24, 1962. Estimated costs are $2,650 which include salary and continuity of service of S. H. Bush, services, supplies and transportation.
.4Z	P. L. Hofmann - Consulting at APED for Audit of Physics Methods. Initial participation was for one week beginning April 22, 1962. Further participation is anticipated for a period of five to six weeks.
.6A	Shipment of Aluminum Clad Plutonium Foils to APED. Estimated costs for packing and shipping are $200.
.2Y	Preparation of Test Samples for Bettis. Additional funds of $10,000 were authorized in May 1962, making a total authorization for the job of $40,000.

Organizational code 7124 - Visual Displays, effective July 1, 1962 has been established to accumulate costs incurred in connection with visual displays and tour arrangements in Hanford Laboratories.

At the request of HOO-AEC, listings were supplied to them of all contracts and purchase orders over $4,000 funded by the Division of Reactor Development in FY 1961. Similar information was requested by the Joint Committee on Atomic Energy Commission from all DRD sites.

Security clearances of several Hanford Laboratories' sections were reviewed in light of the recent AEC revisions in the Weapons and Top Secret categories. Current requirements for all individuals were determined and requests for clearances needed were submitted to the Security and Patrol Operation.

An audit made of signature authorizations on stores orders issued during the week ending May 6, 1962 by Biology, Programming and Operations Research and Synthesis Operation personnel revealed no serious irregularities.

General Accounting

Following is a summary of the status of letter or other agreements covering specific actions requiring AEC concurrence:

AT-237	Summer Institute in Nuclear Energy for Engineering Faculty	Approved May 4, 1962
AT-243	Movement of Mobile Home	Approved May 23, 1962
AT-244	Participation in Wallowa County Educational Day Camp	In hands of AEC
AT-246	Special Science Seminars	In hands of AEC
AT-247	Participation in Standardizing Activities	In hands of AEC

AEC approval of agreements under Agreement AT-6 were received on the following:

Cooperative Research Program - Oregon State University
Plutonium Foils for GE-APED
IAEA Request for Consultant to Japan (R. S. Paul)

Two modifications to Appendix B (Nos. 14 and 18) were approved covering Income Extension Aid Plans for nonexempt and for exempt employees.

During the month $209,016 of equipment was transferred to classified plant accounts from Equipment Work in Progress and $27,095 from Construction Work in Progress.

A summary report of findings in connection with the FY 1961-1962 physical inventory of movable catalogued equipment in the custody of Hanford Laboratories was prepared and distributed. This report represented one complete cycle of inventories for all Hanford Laboratories' holdings, and permitted comparison between the results of this inventory and previous inventories. The FY 1961-1962 inventory disclosed 16 missing items, valued at $5,963 compared to 38 missing items valued at $8,592 in the FY 1960 inventory. As a result of the FY 1961-1962 inventory, 75 units valued at $32,842 were added to record compared to 194 units valued at $99,340 in the FY 1960 inventory. Property control within Hanford Laboratories has improved markedly since the FY 1960 inventory. The amount of missing equipment (16 items) is relatively minor when compared with the inventory of 12,842 items. The Laboratories' Equipment Pool has contributed to the fine showing by assisting property custodians in the control of equipment and by providing a central storage for equipment held for future use.

Operating personnel were notified of physical inventory schedules for movable catalogued equipment for FY-1963 which are (1) Reactor and Fuels, July 1962 (2) Physics and Instruments, January 1963 (3) Radiation Protection, March 1963 and (4) Finance and Administration, April 1963.

Results were distributed for the quarterly inventory of Other Special Materials in custody of 97 material holders as of March 31, 1962. A net shortage of 19 grams of platinum valued at $55.48 has been officially recorded and approval of write-off to Cost obtained.

Hanford Laboratories' material investment at May 1, 1962 totaling $26.9 million is detailed below:

(Amounts in thousands)

Spare Parts	$ 1 238
SS Material	369
Reactor & Other Special Materials	25 344
Total	$26 951

Heavy water costs incurred in May totaling $30,601 were comprised of losses of $29,283 and $1,318 of scrap which resulted from PRTR operations.

Heavy water scrap material returned to Savannah River Operations Office during May business totaled 20,908.66 pounds with a fund value of $253,795 and a nonfund value of $197,484. The balance of scrap material on hand at May 30th totaled 2,594 pounds having a value of $32,311.

Property Accounting, Contract and Accounting Operation, was requested to review Hanford Laboratories' accumulated depreciation accounts to (1) ensure the adequacy of composite rates and reserves, and (2) take the necessary action to adjust our reserve accounts for variances prior to the end of calendar year 1962. This action was initiated after a review by Hanford Laboratories' Property Accounting personnel revealed some rather large variances in the building, machine tools, laboratory equipment, and portable health instrument accounts. Laboratories' reserve accounts may have a net shortage on the order of $900,000. Contract and Accounting Operation is unable to begin an adequacy review until after the close of FY 1962 business.

In response to a request from Contract and Accounting, the existence of all returnable containers (189) held by Laboratories' personnel was checked. Accounts Payable was notified that 13 returnable containers are missing and believed to have been either returned to Central Stores or destroyed because of contamination or frozen valves. Accounts Payable was asked to confirm through a Central Stores search the containers which are definitely missing and should be purchased.

The AEC was advised that 6,688 grams of contaminated platinum valued at $15,459 for which there is no further need at HAPO, is available for off-site shipment.

Fifty-one items valued at $43,111 were received at the Laboratories' Equipment and Material Pool during the month of May. Fifteen items valued at $6,981 were loaned or transferred in lieu of placement of requisitions, nine items valued at $12,215 were withdrawn by custodians, 69 items valued at $21,074 were excessed. There are currently 811 items valued at $549,161 located in storage pool.

Materials in the Pool include:

Beryllium	$ 652
Gold	3 142
Palladium	2 736
Platinum	21 649
Silver	464
Zirconium	312 448
All other material held for convenience of others	246 923
	$588 014

Action as indicated occurred on the following projects during the month:

New Money Authorized Hanford Laboratories

CAH-963	Geological & Hydrological Wells, FY 1963	$11 500

Physical Completion Notices Issued

CGH-858 High Level Utility Cell

Construction Completion and Cost Closing Statements Issued

CAH-896	Stress-Rupture Testing Facility
CAH-902*	Uranium Scrap Burning Facility
CAH-919	Air Conditioning 314 Building
CGH-923	Spectroscopy Laboratory 325 Building
CAH-924*	200 KW Induction Heating System 306 Building

* AEM Services Only

The continuation of the strike by construction crafts is resulting in substantial delays in completion of projects and capital and expense work orders. Actual effect on expenditure patterns is most difficult to predict, however it is estimated that operating costs will be reduced $100,000 to $150,000.

The following new or revised OPG's were issued in May:

OPG No.	New	Revised	Title
1.13		X	Participation in the Atomic Energy Business
22.1.5		X	Biology Operation Organization
22.1.6		X	Operations Research & Synthesis Operation Organization
22.1.11		X	Programming Operation Organization
3.5.2		X	Tuition Refund Program
4.2		X	Pressure Systems
5.2 Supp.	X		Authorization for Construction or Acquisition of Plant and Equipment
55.1.4		X	Payment to Nonexempt Employees for Time Spent on Off-site Travel
55.6.1		X	Travel and Living Expense
9.3	X		Use of Equipment for Electronic Recording of Telephone Conversations

The Functional Organization Chart for Hanford Laboratories was completed and will be distributed early in June. This revision is as of May 1, 1962. Only those sections requiring changes will be reissued as of July 1, 1962.

Contracts processed during the month are listed below:

CA-330	Joseph A. Pask
CA-333	C. A. Lingafelter
DDR-110	Battelle Memorial Institute
SA-223	B&R Tug and Barge Inc.
CA-336	Amos Lane
SA-195	Sperry Products, Inc.

Personnel Accounting

Conflict of interest acknowledgments were received from 99.7% of all exempt employees. Supervision has been requested to obtain acknowledgments from the remaining .3%.

Two employees, J. T. Russell and R. H. Moore were presented patent awards of $125.00 each during the momth. Eight patent awards of $125.00 each were presented to employees of Hanford Laboratories during the first five months of 1962. One award was given to R. L. Watts as a result of a patent he submitted during employment with the Capacitor Department, Hudson Falls, New York.

Following are the payroll statistics for the month of May 1962:

Number of HLO Employees Changes During Month	Total	Exempt	Nonexempt
Employees on payroll at beginning of month	1 432	657	775
Additions and Transfers In	24	6	18
Removals and Transfers Out	17	8	9
Employees on payroll at end of month	1 439	655	784

Overtime Payments During Month	May	April
Exempt	$ 3 564	$ 5 443
Nonexempt	27 416	25 641
Total	$30 980	$31 084

Gross Payroll Paid During Month		
Exempt	$ 613 676	$ 622 613
Nonexempt	442 167	426 457
Total	$1 055 843	$1 049 070

Participation in Employee Benefit Plans at Month End	May		April	
	Number	Per Cent	Number	Per Cent
Pension	1 299	99.3	1 292	99.4
Insurance Plan-Personal	363		361	
-Dependent	1 069	99.7	1 062	99.8
U. S. Savings Bonds				
Stock Bonus Plan	91	39.7	89	38.9
Savings Plan	73	5.1	74	5.2
Savings & Security Plan	1 092	90.2	1 079	89.7
Good Neighbor Fund	966	67.1	964	67.3

Insurance Claims Employee Benefits	Number	Amount	Number	Amount
Life Insurance	-0-	$ 0	-0-	$ 0
Weekly Sickness & Accident	7	522	10	873
Comprehensive Medical	40	3 276	57	4 287
Dependent Benefits				
Comprehensive Medical	92	7 843	128	10 023
Total	139	$11 641	195	$15 183

TECHNICAL ADMINISTRATION

Employee Relations

Nineteen non-exempt employment requisitions were filled during May; 26 remain to be filled.

Professional Placement

Advanced Degree - Seventeen Ph.D. applicants visited HAPO for employment interviews. Ten offers were extended; six acceptances and three rejections were received. Current open offers total seven.

BS/MS - Two program offers and thirteen direct placement offers were extended. Offers accepted - eighteen and three; offers rejected - sixty-one and ten, respectively. Current open offers total 18.

Technical Graduate Program - Six Technical Graduates were placed on permanent assignment; two new members were added to the rolls and one terminated. Current program members total 33.

Technical Information

Revised charts of the Nuclides were obtained and distributed.

The AEC's quarterly report on its Irradiation Effects on Reactor Structural Materials program will be edited and published here and will include results of research efforts at other sites also participating in the program.

ECONOMIC EVALUATIONS

First output results of the Fuel Element Fabrication Costs computer code were reviewed. The machine calculations were quite similar to those manually calculated for the conceptual fabrication plant study, from which test input data was derived. Next step is to fully reconcile results between the two methods of calculation, after which a number of parameters of the machine input can be widely and quickly varied to analyze relative effects on total unit costs.

A variety of information on the economic factors of electric utilities was compiled to assist CR&D Operation with an economic feasibility study of the closed cycle fuel process.

PROCEDURES AND SPECIAL STUDIES

The BSDO representative assigned to the Laboratories issued the results of a superficial examination of the status of routine activity reporting in Hanford Laboratories. His observations were in line with expectations, i.e., that (1) the reports being issued are accepted as being necessary by the people in the field, and (2) no correctional duplications in reporting exist. Some misunderstanding as to the use made of some reports was found, and some related misunderstanding as to the information called for.

A procedure has been put in place to accumulate information needed to assess the amount of time spent by Hanford Laboratories' personnel on technical interchange with HAPO visitors. This study will cover a three-month period - May, June and July. The report on May volume is due June 15.

A considerable amount of time was spent during May on the preparation and installation of Hanford Laboratories' version of the ACCENT ON VALUE program.

FACILITIES ENGINEERING

Projects

At month's end Facilities Engineering Operation was responsible for 11 active projects having total authorized funds in the amount of $2,401,600. The total estimated cost of these projects is $7,462,000. Expenditures on these projects through April 30, 1962 were $1,162,000.

The following summarizes the status of project activity in May:

Number of authorized projects at month's end -------------- 11

Number of new projects authorized in May ------------------ 1
CAH-963, Geological & Hydrological Wells - FY 1962

Projects completed in May --------------------------------- 1
CGH-858, High Level Utility Cell - 327 Building

New projects submitted to the AEC in May ------------------ 0

New projects awaiting AEC authorization ------------------- 3
CAH-917, Field Service Center - Atmospheric Physics
CAH-959, Graphite Machining Shop
CAH-962, Low Level Radiochemistry Building

Project proposals complete or nearing completion ---------- 2
CAH-958, Plutonium Fuels Testing and Evaluation Labs., 308 Building
Facility for Radioactive Particle Inhalation Studies

Services

Engineering services were provided during May on the following activities with satisfactory progress shown on each:

Exhausts for Ceramic Fuels Laboratory - 325 Building
Remote Humidity Recorder - 144-F Building
Split-Half Machine - Critical Mass Laboratory
Electrical Load Study - 108-F Building
Relocate Induction Heating Equipment - 314 Building

Ten equipment requisitions totaling $4,000 were issued during the month. The total value of materials and equipment being processed is $450,000. Outstanding orders are being given special attention to assure plant site delivery prior to July 1.

Plant Engineering effort was expended on:

- Pressure system code work on: 325 Building vacuum tank, 231-Z hydraulic system, 105-C Irradiation Studies loop, PRTR rupture loop and zircaloy tube burst facility.
- 325 Analytical Laboratory glove box.
- 747 Whole Body Monitor Cell door drive.
- 209-E Compressor and motor-control center.
- 325 Ceramic Fuels area air conditioner.
- 325 "B" System duct cleaning.
- 309 M&M Building office additions
- 325 Small Particle Laboratory
- 309 Building standby electrical service study
- 309 Building emergency warning system

Maintenance and Operation

Costs for April were $122,760. Costs to date, $1,490,565 are 96.7% of forecast. Improvement maintenance costs for April were $2,958.

The following tabulation summarizes waste disposal operations:

	April	March
Concrete Barrels	3	21
Loadluggers-Hot Waste	4	5
Crib Waste	240,000 gal.	260,000 gal.

Drafting

The equivalent of 168 drawings were completed during the month. Major jobs in progress include: 280 ton extrusion press installation, PRTR as-builts, PRTR shim rod control housing, Mark V-A fuel element fabrication equipment, radial motion instrument servo, shim rod control, electrical resistivity sample holder, scintillation scanner housing, 300 Area retention and crib waste piping, cladding cutter assembly for PRTR, and ETR replacement loop.

Construction

There were 99 existing J. A. Jones Company orders at the beginning of the month with a total unexpended balance of $183,411. One hundred and eight new orders, 2 supplements and adjustments for underruns amounted to $98,191. Expenditures during the month on Hanford Laboratories' work were $65,326. Total J. A. Jones backlog at month's end was $216,276.

Expenditures for this month would have been approximately $150,000 if there had been no work stoppage.

	HL Unexpended Balance	CE&UO Unexpended Balance
Orders outstanding beginning of month	$183 411	--
Issued during the month (Inc. Supp. & Adj.)	98 191	
J. A. Jones Expenditures during month (Inc. C.O. Costs)	65 326	
Balance at month's end	216 276	
Orders closed during month	56 165	

Construction activities completed during May were:

307 Basin - Repair leak in 8" influent line
325 Building - Reroute vacuum, water and drain lines and electrical conduit in Room 130.

W Sale
Manager
Finance and Administration

W Sale:whm

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 73905

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CAH-962	Low Level Radiochemistry Building	Funds Available to AEC

AUTHORIZED FUNDS	DESIGN $	AEC $	COST & COMM. TO	$
$ -0-	CONST. $	GE $	ESTIMATED TOTAL COST	$ 1,200,000

STARTING DATES	DESIGN 7-15-62*	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN 5-15-63*
	CONST.	DIR. COMP. DATE		CONST.

ENGINEER

FEO - DS Jackson

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides a building in which extremely sensitive radioanalyses and methods development can be performed in an atmosphere protected from the environs. It consists of designing and constructing a building housing approximately 22,000 square feet of floor area including the basement.

The project proposal requesting $113,000 total design funds was submitted to the AEC, for authorization on April 30, 1962.

The HOO-AEC Board of Review recommended the Manager, HOO approve the project proposal and submit it to Washington-AEC recommending authorization of design funds.

*Based on authorization of design money by July 1, 1962.

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
	CONST.	DIR. COMP. DATE		CONST.

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73905

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CAH-959	Graphite Machining Shop - 300 Area	62-k

AUTHORIZED FUNDS			COST & COMM. TO	
$	DESIGN $	AEC $	COST & COMM. TO	$
	CONST. $	GE $	ESTIMATED TOTAL COST	$ 105,000

STARTING DATES				EST'D. COMPL. DATES	
DESIGN	8-1-62*	DATE AUTHORIZED		DESIGN	11-15-62*
CONST.	1-15-63*	DIR. COMP. DATE		CONST.	8-1-63*

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

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.

The project proposal was submitted to the Commission for approval March 2, 1962.

*Based on AEC approval by July 1, 1962.

PROJ. NO.	TITLE	FUNDING
CAH-963	Geological & Hydrological Wells - FY-1962	62-k

AUTHORIZED FUNDS			COST & COMM. TO	
$ 80,000	DESIGN $ 1,400	AEC $ 68,500	COST & COMM. TO	$ -0-
	CONST. $ 78,600	GE $ 11,500	ESTIMATED TOTAL COST	$ 80,000

STARTING DATES				EST'D. COMPL. DATES	
DESIGN	5-18-62	DATE AUTHORIZED	5-9-62	DESIGN	6-1-62
CONST.	7-15-62	DIR. COMP. DATE	4-1-63	CONST.	4-1-63

ENGINEER

FEO - SE Ralph

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	0	1
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

[redacted]

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 73925

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CGH-977	Small Particle Technology Laboratory - 325 Building	62-k

AUTHORIZED FUNDS			
$ 40,000	DESIGN $ 2,000 / CONST. $ 38,000	AEC $ -- / GE $ 40,000	COST & COMM TO 5-13-62 $ 4,050 / ESTIMATED TOTAL COST $ 40,000

STARTING DATES	DATE AUTHORIZED	EST'D. COMPL. DATES
DESIGN 4-23-62 / CONST. 7-1-62	DATE AUTHORIZED 3-21-62 / DIR. COMP. DATE 11-1-62	DESIGN 5-25-62 / CONST. 11-1-62

ENGINEER

FEO - DS Jackson

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION (FEO)	0.5	25
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100*	100
TITLE I			
GE-TIT. II	100	100*	100
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides laboratory space for research and development in small particle technology related to the generation, control, and disposal of radioactive wastes.

Directive HW-535, dated March 21, 1962, authorized total project funds in the amount of $40,000.

Design was started on April 23, 1962.

Detailed design is completed and information is being assembled for issuance to J. A. Jones Construction Company.

* Taken from the Project Planning Schedule.

PROJ. NO.	TITLE	FUNDING
CGH-958	Plutonium Fuels Testing and Evaluation Laboratory-308 Bldg.	62-k

AUTHORIZED FUNDS			
$	DESIGN $ / CONST. $	AEC $ / GE $	COST & COMM. TO $ / ESTIMATED TOTAL COST $ 150,000

STARTING DATES	DATE AUTHORIZED	EST'D. COMPL. DATES
DESIGN 8-1-62* / CONST 11-1-62*	DATE AUTHORIZED / DIR. COMP. DATE	DESIGN 11-1-62* / CONST. 3-15-63*

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

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.

The project proposal was withdrawn from the Commission by General Electric Company for modification. The project proposal, with expanded justification, is being routed for signatures.

*Based on AEC authorization by July 1, 1962.

1237835

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73905

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CGH-951	A-C Column Facility - 321 Building	0290

AUTHORIZED FUNDS			COST & COMM. TO 5-13-62	
$ 55,000	DESIGN $ 5,000	AEC $ -0-		$ 11,535
	CONST. $ 50,000	GE $ 55,000	ESTIMATED TOTAL COST	$ 55,000

STARTING DATES	DATE AUTHORIZED	EST'D. COMPL. DATES
DESIGN 1-30-62	1-12-62	DESIGN 4-1-62
CONST. 3-25-62	DIR. COMP. DATE 10-31-62	CONST 10-31-62

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II	0		
CONST.	100	23	23
PF	100	23	23
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.

Relocation of "C" column is 60% complete. Instrument line gutters are installed. Miscellaneous interconnecting piping work is continuing. Work Order has been issued for fabrication of "A" Column.

PROJ. NO.	TITLE	FUNDING
CGH-955	Reactivation of the H-1 Loop - 105-H Building	0490

AUTHORIZED FUNDS			COST & COMM. TO 5-13-62	
$ 10,000	DESIGN $ 10,000	AEC $		$ 10,000
	CONST. $	GE $ 10,000	ESTIMATED TOTAL COST	$ 105,000

STARTING DATES	DATE AUTHORIZED	EST'D. COMPL. DATES
DESIGN 4-15-62	3-29-62	DESIGN 8-30-62
CONST. 7-15-62	DIR. COMP. DATE	CONST 12-15-62

ENGINEER

FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	23*	11
TITLE I			
GE-TIT. II	100	23*	11
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.

AEC Directive No. HW-536, dated March 29, 1962 authorized $10,000 to initiate design and provide new cost estimate for review by the Commission. Design work is in progress.

* Taken from the Project Planning Schedule.

1231286

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73905

DATE 5-31-62

PROJ. NO. CAE-927	TITLE Additions to the 271-CR Building-Waste Treatment Demonstration Facility		FUNDING 61-j
AUTHORIZED FUNDS $ 92,000	DESIGN $ 11,000 / CONST. $ 81,000	AEC $ 76,300 / GE $ 15,700	COST & COMM. TO 5-13-62 $ 14,899 (GE) / ESTIMATED TOTAL COST $ 92,000
STARTING DATES: DESIGN 6-15-61 / CONST. 2-15-62	DATE AUTHORIZED 5-15-61 / DIR. COMP. DATE 7-31-62	EST'D. COMPL. DATES: DESIGN 2-5-62 / CONST. 8-15-62	

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	4	80
COST PLUS FIXED FEE		110
PLANT FORCES		
ARCHITECT-ENGINEER		150
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II			
AE-TIT. II	100	100	100
CONST.	100	40	50
PF			
CPFF	33	100	90
FP	67	10	30

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.

Construction has been stopped by the strike of construction workers on 5-16-62.

PROJ. NO. CAH-936	TITLE Coolant Systems Development Laboratory 1706-KE Building Addition		FUNDING 62-k
AUTHORIZED FUNDS $ 130,000	DESIGN $ 9,000 / CONST. $ 121,000	AEC $ 115,000 / GE $ 15,000	COST & COMM. TO 5-13-62 $ 14,772 / ESTIMATED TOTAL COST $ 130,000
STARTING DATES: DESIGN 9-8-61 / CONST. 5-1-62	DATE AUTHORIZED 4-5-62* / DIR. COMP. DATE 10-31-62	EST'D. COMPL. DATES: DESIGN 1-1-62 / CONST. 11-15-62	

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	3	33
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		166
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING	.4	4

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	10	7
PF			
CPFF			
FP	100	10	7

SCOPE, PURPOSE, STATUS & PROGRESS

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. Current estimate of Title I and II costs - $11,000.

*Original authorization for design was 8-9-61.

Construction has been stopped by the construction strike of 5-16-62.

1231287

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 73905

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CAH-917	Field Service Center - Atmospheric Physics	61-j

AUTHORIZED FUNDS			COST & COMM. TO	
$	DESIGN $	AEC $	$	
	CONST. $	GE $	ESTIMATED TOTAL COST	$ 154,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
DESIGN 7-30-62*		DATE AUTHORIZED	DESIGN 12-1-62*	
CONST 11-15-63*		DIR. COMP. DATE	CONST 10-1-63*	

ENGINEER

FEO - JT Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

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.

General Electric has requested the proposal be returned.

*Based on AEC authorization by 6-15-62.

PROJ. NO.	TITLE	FUNDING
CAH-922	Burst Test Facility for Irradiated Zirconium Tubes	62-k

AUTHORIZED FUNDS			COST & COMM. TO 5-13-62	
$ 29,600	DESIGN $ 29,600	AEC $	$ 29,600	
	CONST. $	GE $ 29,600	ESTIMATED TOTAL COST	$ 289,000

STARTING DATES	DATE AUTHORIZED	EST'D. COMPL. DATES
DESIGN 11-7-61	DATE AUTHORIZED 10-23-61	DESIGN 5-31-62
CONST. 6-15-62	DIR. COMP. DATE To be established at a later date	CONST. 12-1-62

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER - Bovay Engineers	3	230
DESIGN ENGINEERING OPERATION	2	140
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	57	100	100
AE-TIT. II	43	100	100
CONST.	100		
PF			
CPFF			
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 available because of the limited operating history of Zircaloy-2 pressure tubing in reactors.

A project proposal revision requesting construction funds is routing for approval.

Total project cost is estimated to be $289,000, of which $9,000 is the estimated value of an existing prototype containment vessel which will be transferred to the project.

1231208

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73905

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CAH-916	Fuels Recycle Pilot Plant	4-62-d-3

AUTHORIZED FUNDS			
$ 385,000	DESIGN $ 385,000	AEC $ --	COST & COMM. TO 5-13-62 $ 384,900
	CONST. $ -0-	GE $ 385,000	ESTIMATED TOTAL COST $ 5,100,000***

STARTING DATES		EST'D. COMPL. DATES
DESIGN 3-15-61	DATE AUTHORIZED 10-27-61**	DESIGN 8-15-62
CONST. 6-15-62*	DIR. COMP. DATE	CONST. 11-1-64

ENGINEER

FEO - RW Dascenzo

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION	30	
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	80	80
TITLE I	11	100	100
GE-TIT. II	89	78	78
AE-TIT. II	0		
CONST.	100	0	0
PF			
CPFF			
FP			

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.

A revised project proposal is being routed in General Electric Company for approval.

Design is continuing on all phases as scheduled.

A total of 243 drawings have been issued for comment and 51 for approval.

*Estimated construction starting date for removal of burial ground fill.

**Original authorization for initiation of design was February 9, 1961. Oct. 27, 1961 is the authorization date for the last design supplement.

***Including transferred capital property valued at $100,000.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73905

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CAH-888	Biology Laboratory Improvements	60-h-1

AUTHORIZED FUNDS			COST & COMM. TO 5-13-62	ESTIMATED TOTAL COST
$ 420,000	DESIGN $ 44,000 / CONST. $ 376,000	AEC $ 359,500 / GE $ 60,500	$ 53,346 (GE)	$ 420,000

STARTING DATES	DATE AUTHORIZED	DIR. COMP. DATE	EST'D. COMPL. DATES
DESIGN 8-8-60	4-18-61*		DESIGN 3-31-61
CONST. 7-10-61		3-31-62	CONST 7-1-62**

ENGINEER

FEO - JT Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		2580
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	NS	100
TITLE I			
GE-TIT. II	17	NS	100
AE-TIT. II	83	NS	100
CONST.	100	100	88
PF	1	100	100
CPFF	10	NS	0
FP	89	100	97

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides additional space for biological research supporting services, and involves an addition to the 108-F Building.

Wiring of panels and motors is complete. Startup of equipment has begun.

*Original authorization for design was May 3, 1960.

Preliminary air balancing has been completed. Final adjustments cannot be made until vinyl floor covering is installed. The re-order of floor covering for the first floor and for the shortage on the second floor did not arrive on schedule and has held up final air balancing. Fire alarm tie-in has been made.

Allied Engineering work on refinments to fabrication of Radiation Source Handling Facilities is in progress. Completion is not expected until about May 31, 1962.

The AEC has published a Physical Completion Notice with exceptions dated April 4, 1962.

Work has not been resumed since Monday, May 14, 1962 due to the Carpenters Strike.

** Estimated project completion date provided the strike has been settled and Radiation Source Handling Facilities have been received as scheduled.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 73905

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CAH-867	Fuel Element Rupture Test Loop	58-e-15

AUTHORIZED FUNDS				
$1,500,000	DESIGN $ 130,000	AEC $ 820,000	COST & COMM. TO 5-13-62	$ 549,798 (GE)
	CONST. $ 1,370,000	GE $ 680,000	ESTIMATED TOTAL COST	$1,500,000

STARTING DATES				EST'D. COMPL. DATES	
DESIGN	8-1-60	DATE AUTHORIZED	6-24-60*	DESIGN	3-15-61
CONST.	11-2-60	DIR. COMP. DATE	6-30-62	CONST	6-30-62

ENGINEER

TR&AO-MEEO - PC Walkup

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	5	2385
COST PLUS FIXED FEE		
PLANT FORCES	0	1940
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	91	100	100
AE-TIT. II	9	100	100
CONST.	100	98	95
PF	2	100	50
CPFF	57	100	97
FP (1)	10	100	100
(2)	31	95	92

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 and due to CPFF labor strike.

*Initial authorization was on 10-1-59.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS				
$	DESIGN $	AEC $	COST & COMM. TO	$
	CONST. $	GE $	ESTIMATED TOTAL COST	$

STARTING DATES				EST'D. COMPL. DATES	
DESIGN		DATE AUTHORIZED		DESIGN	
CONST.		DIR. COMP. DATE		CONST.	

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

1231291

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73905

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CAH-866	Shielded Analytical Laboratory - 325-B Building	61-a-1

AUTHORIZED FUNDS			COST & COMM. TO 5-13-62	$ 131,125 (GE)
$ 700,000	DESIGN $ 60,000	AEC $ 546,500	ESTIMATED TOTAL COST	$ 655,000
	CONST. $ 640,000	GE $ 153,500		

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 9-5-59	DATE AUTHORIZED 5-31-60*		DESIGN 11-14-60	
CONST. 6-15-61	DIR. COMP. DATE 6-30-62		CONST. 9-30-62	

ENGINEER

FEO - RW Dascenzo

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	9	1705
COST PLUS FIXED FEE		
PLANT FORCES	0	3
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING	1	

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	10	100	100
AE-TIT. II	90	100	100
CONST.	100	94	58
PF	3	1	1
CPFF	2	0	0
FP	95	100	58

SCOPE, PURPOSE, STATUS & PROGRESS

This project will allow greater capacity for analytical work involving today's more highly radioactive solutions and consists of adding a shielded laboratory to the 325 Building.

*Original authorization for preliminary design was August 12, 1959.

All work stopped by the carpenter's strike as of Wednesday morning May 16, 1962.

The cell roof of normal concrete was poured May 14, 1962.

Roof decking installation was started on May 15, 1962 but not completed due to the strike.

The contractor's completion date has been extended from April 15, 1962 to June 24, 1962.

1231292

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73905

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CGH-858	High Level Utility Cell - 327 Building	0290

AUTHORIZED FUNDS			
$ 400,000	DESIGN $ 50,000	AEC $ -	COST & COMM. TO 5-13-62 $ 369,798
	CONST. $ 350,000	GE $ 400,000	ESTIMATED TOTAL COST $ 399,462

STARTING DATES				EST'D. COMPL. DATES	
DESIGN	11-1-59	DATE AUTHORIZED	4-6-61*	DESIGN	2-15-61
CONST.	5-15-61	DIR. COMP. DATE	6-1-62	CONST.	5-15-62

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		1000
PLANT FORCES		
ARCHITECT-ENGINEER		35
DESIGN ENGINEERING OPERATION		930
GE FIELD ENGINEERING		60

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	95	100	100
AE-TIT. II			
Vendor	5	100	100
CONST.	100	100	100
PF			
CPFF	100	100	100
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

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.

Current estimate of Title I and II costs is $62,000. Detailed design started 4-1-60. Procurement and construction authorized 4-6-61.

*Original authorization for design was October 1, 1959.

Number of purchase orders required	12	Value	$205,825
Number of purchase orders placed	12	Value	205,825

Construction completed 5-15-62.

No further reporting will be made.

1231293

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 73905

DATE 5-31-62

PROJ.NO.	TITLE	FUNDING
CGH-857	Physical & Mechanical Properties Testing Cell - 327 Bldg.	0290

AUTHORIZED FUNDS			
$ 460,000	DESIGN $ 45,000	AEC $ -	COST & COMM TO 5-13-62 $ 278,635
	CONST. $ 415,000	GE $ 460,000	ESTIMATED TOTAL COST $ 460,000

STARTING DATES			EST'D. COMPL. DATES
DESIGN 11-2-59	DATE AUTHORIZED 9-22-61*		DESIGN 3-15-61
CONST. 2-12-62	DIR. COMP. DATE 12-15-62		CONST. 12-15-62

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		34
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		760
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	2	2
PF			
CPFF	18	9	9
FP			
Equip.	82	0	0

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facilities for determining physical and mechanical properties of irradiated materials, and involves the installation of a cell in the 327 Building.

Current estimate of Title I and II costs - $55,000. Detailed design started 4-1-60. Procurement and construction authorized 9-22-61.

Basement floor and foundation concrete work is completed. Construction has stopped until cell assembly is delivered.

Number of purchase orders required	19 Value (Est.)	$253,000**
Number of purchase orders placed	19 Value	203,000

*Original authorization for design was October 1, 1959.
**Includes delivery charges, inspection and contingency.

Cell fabrication is running into difficulty because of vendor errors in boring plug openings over tolerance. To rectify this situation a special procedure is being proposed and submitted for our approval. Special effort will be required by all concerned if delivery date of 6-21-62 is met.

1231294

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73905

DATE 5-31-62

PROJ. NO.	TITLE	FUNDING
CAH-822	Pressurized Gas Cooled Facility	4141 Operating

AUTHORIZED FUNDS			
$ 1,170,000	DESIGN $ 43,000	AEC $ 15,000	COST & COMM. TO 5-13-62 $ 1,129,859
	CONST. $ 1,127,000	GE $ 1,155,000	ESTIMATED TOTAL COST $ 1,170,000

STARTING DATES	DATE AUTHORIZED	EST'D. COMPL. DATES
DESIGN 8-19-59	DATE AUTHORIZED 2-2-62*	DESIGN 4-29-60
CONST. 10-17-60	DIR. COMP. DATE 6-30-62	CONST. 6-30-62

ENGINEER

TR&AO-MEEO - DP Schively

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100	98	91
PF	1.4	0	0
CPFF	22.1	99	99
FP	6.6	100	100
Govt.Eq	69.9	100	90

SCOPE, PURPOSE, STATUS & PROGRESS

Piping stress calculations for new heater indicate overstress at anchors. New analysis being made on basis of relocated anchors. Shells to be complete by 6-1-62, assembly with resistance elements the week following.

Bristol-Siddeley is proceeding with window-pad bearings. Test of first machine is now scheduled for May 28.

*Initial authorization date was December 18, 1958.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS			
$	DESIGN $	AEC $	COST & COMM. TO $
	CONST. $	GE $	ESTIMATED TOTAL COST $

STARTING DATES	DATE AUTHORIZED	EST'D. COMPL. DATES
DESIGN	DATE AUTHORIZED	DESIGN
CONST.	DIR. COMP. DATE	CONST.

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

1231295

TEST REACTOR AND AUXILIARIES OPERATION

MAY 1962

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output was 654 MWD for a plant efficiency of 30% and a total experimental time efficiency of 49.9%. Accumulated exposure through May 31, is 7867 MWD. Additional exposure information is as follows:

Maximum UO_2 exposure/element	2440 MWD/TU
Average UO_2 exposure/element	1370 MWD/TU
Maximum Pu-Al exposure/element	76.3 MWD
Average Pu-Al exposure/element	47.0 MWD
Maximum Moxtyl exposure/element	22.8 MWD
Average Moxtyl exposure/element	10.5 MWD

The tenth refueling was performed during the extended outage that began on April 30, 1962. Six UO_2 elements that are instrumented with Cobalt-Zircaloy wire for flux monitoring were charged in ring 1 under PRTR Test 37. Seven new mixed oxide elements were charged in ring 9. The core was composed of 39 Pu-Al elements, 35 UO_2 elements and 11 mixed oxide elements at startup on May 20. Two additional fresh LX Pu-Al elements and one additional fresh mixed oxide element were charged during a short outage on May 25.

Fifteen process tubes, three HX Pu-Al elements, four mixed oxide elements, and two UO_2 elements were inspected.

The status of the various test elements at the end of May is shown below:

PRTR Test	Tube Location	Element Number	Description	Date Charged	Date Discharged	Accumulated MWD
5	1653	1501	UO_2-Tubular	11/3/61	--	69.2
10	1447	1082	UO_2-Hot Swage	11/3/61	--	42.9
10	1647	1067	UO_2-Vipac	11/3/61	--	47.2
13	1253	5092	Pu-Al Instrumented	12/3/61	--	62.9
13	1544	5093	Pu-Al Instrumented	12/3/61	--	54.5
13	1847	5052	Pu-Al Instrumented	1/17/62	4/2/62	30.7
13	1247	5051	Pu-Al Instrumented	1/17/62	2/7/62	13.1
13	1853	5094	Pu-Al Instrumented	12/3/61	--	55.9
13	1556	5095	Pu-Al Instrumented	1/17/62	4/18/62	41.3

1231296

PRTR Test	Tube Location	Element Number	Description	Date Charged	Date Discharged	Accumulated MWD
14	1146	5096	Moxtyl-Swaged	4/2/62	5/9/62	13.7
14	1156	5097	Moxtyl-Swaged	4/2/62	--	17.9
14	1049	5098	Moxtyl-Vipac	5/8/62	--	7.9
14	1051	5099	Moxtyl-Vipac	5/8/62	--	7.9
37	1449	1096	UO_2-Instrumented	5/12/62	--	8.3
37	1649	1097	UO_2-Instrumented	5/12/62	--	9.1
37	1552	1098	UO_2-Instrumented	5/12/62	--	7.9
37	1548	1099	UO_2-Instrumented	5/12/62	--	8.6
37	1651	1100	UO_2-Instrumented	5/12/62	--	8.3
37	1451	1101	UO_2-Instrumented	5/12/62	--	8.7

The purpose of the extended outage (April 30 to May 20) was to perform some of the modification and improvement work noted below. Operation was without incident. No scrams occurred. D_2O and helium losses were 2,120 pounds (4-16 to 5-25 inventories) and 73,500 scf, respectively. A total of 20,900 pounds D_2O equivalent was shipped to SRP for reprocessing.

Equipment Experience

The pump occupying No. 1 position was replaced during the month when high leakage (over one gallon per minute) persisted. The seal exhibited minor damage in the form of a faint flow path across the face. The raised face had worn approximately .002 inch from an original .018 inch. No indications of eminent failure were noted although numerous chips were displaced from the seal inside edge. The pump bearings were rough, which is believed to account for high amperage experience. Bearings were replaced.

Process Tube 1643 was replaced as part of the regular inspection program without incident.

About forty hours were applied this month in rerouting, replacing, removing, etc. and making other improvements in tubing and piping runs in the process cell.

Moderator pump and motor No. 3 which were damaged last March when the motor bearings failed were returned to service. The motor was rewired and the rotor, bearings and one end bell replaced.

Bearings were replaced on reflector pump motor No. 1 because of excessive play.

In an effort to lower the background in Manhole No. 2 for the evaluation of a low level counting system while still holding the present HM chambers on scale, a pure beta source has been installed inside each chamber to replace the external cobalt source. This change has not effected the operation of the existing effluent monitors.

Programmed maintenance required 861 man hours or 16.5% of total available man hours.

Improvement Work Status (significant items)

Work Completed:

Rerouting of D_2O injection to Primary System
Dual valving on helium stack lines
Automatic condensate drain from HX-7
Modification of Primary Pump motor power cables
Installation of Primary Pump bowl venting system
Replacement of the RTD terminal board
Gas bottle storage facility modification
Installation of additional sectionalizing pole-top switch on the 13.8 KV power line
Boiler feedwater auxiliary control valve
Modification of basement access hoist.

Work partially completed:

Safety circuit ground and low voltage detector
Outlet nozzle cap modification (now 67% complete)
Relocation of pressurizer level transmitters
Fueling vehicle hoist modification
Reactor core liquid level instrumentation

Design work completed:

Primary oxygen analyzer modification
Enlarge chemical feed system
Decontamination facility
Shim rod readout modification
Prototype replacement inlet gas bellows
Chain barricade for rotating shield
Permanent mounting of 2nd startup channel
Modification to storage basin crane
Primary pump recording ammeters
Core blanket system piping modifications
Prototype low level effluent monitor
High pressure helium compressor inter-after cooler relief
Flanges for safety relief valves in helium system
Outlet nozzle bracing

Design work partially completed:

Control room ventilation
Additional fuel storage and examination layout
Oil storage building

Boiler feed pump seals
Third exhaust air activity channel
Interlock between charge-discharge machine shroud seat and discharge hoist
Compressed air supply revision
Rupture monitor sample line changes
Fuel transfer system modifications

Process Engineering and Reactor Physics

Fuel substitution tests were performed on four different kinds of fuel elements. Preliminary analysis of the measurements indicate a general agreement with calculation. The uncertainties in the extrapolation of the critical approach data are on the order of $\pm$ 1/2 inch of $\pm$ 0.25 milli-k, which represent uncertainties in the observed changes of $\pm$ (5 to 10)%. Two sets of lutetium foils were irradiated on fresh LX Pu-Al elements in tube 1556. The data are being analyzed by PIRDO to derive neutron spectral information.

The reactor power distribution was determined during the rise to full power with a uniform shim insertion pattern. All operable shims were inserted to 50 inches and held there during heatup and rise to 60 MW. Flow, temperature and ΔT maps were obtained. These data will be used by PIRDO in attempting to formulate a computational model of the PRTR. They are also of interest from an operational viewpoint since they yield the power generation rates of the various types of fuel elements in comparable locations with a uniform flux perturbation.

PRTR Test Number 24 (Temperature Coefficients) final report was issued. Data showed that the moderator coefficient was about the same as reported for the startup tests. The primary coefficient has decreased to about one-half the value reported for the initial startup because of fuel loading changes.

The first graphite oxidation test (PRTR Test Number 40) was performed. The radiation level, 10 hours after the reactor was down, was 6×10^6 r/hr and was decaying with a 13.8 hr half-life period. Since this is not sufficient to provide significant radiation to graphite samples, this test was terminated by sponsor request.

The full steam generator design pressure of 410 psig was successfully reached during the performance of PRTR Test Number 42 (Full Design Operation of Steam Generator). Comparison of the data between this successful attempt in May and the unsuccessful attempt in April is underway.

Other work on PRTR Tests during the month included:

PRTR Test Number 15 (HX-1 Enclosure Temperatures) Performance was completed.

PRTR Test Number 43 (Process Tube Movement). Data was obtained over two startups and one shutdown.

Procedures

Revised operating procedures issued			10
Revised operating standards issued			2
Temporary deviations to operating standards issued			4
Revised process specifications accepted for use			1
Maintenance manuals issued			0
Maintenance procedures issued			0
Drawing as-built status	May	Total	
Approved for as-built	41	562	
Ready for approval		32	
In drafting		92	
Voided		46	
No change required		84	
		816 - 74%	

Personnel Training

Qualification subjects	802 Man Hours
Specifications, Standards, Procedures	146
Maintenance Craftsmen	55
	1 003 Man Hours

Status of Qualified Personnel at Month-End

Qualified Reactor Engineers	10
Qualified Technicians	6
Qualified Technologists	18

Project Items

Second communication system	-	100%
Fuel element examination facility	-	No Change
Water control Laboratory	-	Work stopped by construction strike

Plutonium Recycle Critical Facility

Design Test Status:

Electrical	100%
Instrumentation	95%
Fuel Handling	20%
Moderator System	100%
Thimble Coolant	75%

Work status of items initiated as a result of startup testing is:

PRCF safety circuit trip on PRTR vent of total containment trip - awaiting PRTR test.

Installation of D_2O purification system - partially complete.

Installation of flange in vent line from weir - design complete.
Removal of PRCF cell radiation monitor from PRTR annunciator - design complete.
Provision for single rod drop - design partially complete.

The neutron source has been installed in the source positioner and nuclear instrumentation response testing is underway.

Preparation of operating procedures was completed.

Facility orientation for all PRTR personnel was completed. Training consisted of 70 man hours for PRCF personnel

Fuel Element Rupture Test Facility

Project Status (Project CAH-867)

Overall construction is estimated at 95% complete. CPFF work is 97% complete and the water plant 92%. All construction work halted on May 16, because of the strike.

Design of shielding for piping and heat exchangers is 90% complete.

GAS COOLED POWER REACTOR PROGRAM

Project Status (Project CAH-822)

The project is 91% complete. Approval drawings were received for review from the heater vendor. Bristol-Siddeley continued to experience difficulties with the gas lubricated bearings for the blowers and testing was delayed.

Operation

Fifty percent of the PRTR operating personnel have completed formal classroom training. Maintenance craftsmen received 166 manhours of training.

TECHNICAL SHOPS

Total productive time for the period was 23,807 hours. This includes 18,581 hours performed in the Technical Shops, 3,761 hours assigned to Minor Construction, 853 hours assigned to off-site vendors, and 612 hours to other project shops. Total shop backlog is 21,518 hours, of which 70% is required in the current month with the remainder distributed over a three-month period. Overtime hours worked during the month was 7% (1,701.3) of the total available hours.

Distribution of time was as follows:

	Man-Hours	% of Total
Fuels Preparation Department	5,221	21.93%
Irradiation Processing Department	1,762	7.40
Chemical Processing Department	700	2.94
Hanford Laboratories Operation	16,124	67.73

Requests for emergency service increased requiring a 7.4% overtime ratio compared to 6.4% ratio for the previous period. Factors influencing the overtime rate include the urgency for developing satisfactory rail supports for NPR fuel elements, for the fabrication of standards used in non-destructive testing of NPR components, for graphite molds associated with weapons work, and for the fabrication of fuel element components scheduled to be tested in the MTR and the ETR at Arco, Idaho.

WD Richmond

Manager
Test Reactor and Auxiliaries

WD Richmond:bk

INVENTIONS OR DISCOVERIES

All persons engaged in work that might reasonably be expected to result in inventions or discoveries advise that, to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during the period covered by this report except as listed below. Such persons further advise that, for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

INVENTOR	TITLE OF INVENTION OR DISCOVERY
H. H. Van Tuyl	Improvement of Strontium Sulfate Precipitation Process
F. P. Roberts	Separation of Sodium from Cesium
Jerry J. Cadwell	Pipe Connectors, Fission Product Clamp Connector

H M Parker

Manager, Hanford Laboratories