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HANFORD LABORATORIES OPERATION MONTHLY ACTIVITIES REPORT

APRIL, 1962

MAY 15, 1962

[redacted]

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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HANFORD LABORATORIES OPERATION
MONTHLY ACTIVITIES REPORT
APRIL, 1962

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Compiled by

Operation Managers

May 15, 1962

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

[redacted]

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: April 30, 1962

	At Beginning of Month		At Close of Month		Total
	Exempt	Salaried	Exempt	Salaried	
Chemical R & D	131	123	131	123	254
Reactor & Fuels R & D	176	154	172	156	328
Physics & Instrument R & D	92	59	93	59	152
Biology	36	53	35	54	89
Operations Res. & Syn.	17	4	17	4	21
Radiation Protection	40	89	40	90	130
Finance and Administration	107	99	100	99	199
Programming	17	3	17	3	20
General	3	4	3	4	7
Test Reactor & Auxiliaries	49	182	49	183	232
TOTAL	668	770	657	775	1,432

BUDGETS AND COSTS

April operating costs totaled $2,057,000, a decrease of $263,000 from the previous month; fiscal year-to-date costs are $21,966,000 or 79% of the $27,946,000 control budget. Hanford Laboratories research and development costs for April, 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	$ 49	$ 442	$ 605	73
03 Program	6	23	33	175	19
04 Program	746	949	8 934	11 006	81
05 Program	78	121	794	1 088	73
06 Program	204	237	2 008	2 637	76
	1 080	1 379	12 211	15 511	79
FPD Sponsored	95	115	1 101	1 400	79
IPD Sponsored	83	138	1 101	1 348	82
CPD Sponsored	117	157	1 340	1 636	82
Total	$ 1 375	$ 1 789	$15 753	$19 895	79

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

UO_2 fuel, irradiated in PRTR to a maximum exposure of 2075 MWD/T, continues to show no mechanical distortion or corrosion.

Hydriding of the Zircaloy-2 cladding on a cold-swaged element irradiated in VBWR to an exposure of 3400 MWD/T is believed to have caused the break in the fuel jacket subsequent to discharge.

A new electrode design for cathodic etching of both the fuel tube and end cap prior to magnetic force welding has been successfully demonstrated.

[redacted]

Short sections of inverted-cluster and single-rod fuel element cladding have been successfully fabricated by high voltage electron beam welding.

Remote fabrication studies continue with the successful simultaneous compaction of all three components of a Mark II-C nested tubular fuel element. Cladding interactions were determined to be minimal.

Electron microscopy examination of irradiated UO_2 crystals at 500 C and 980 C revealed little change in surface appearance due to irradiation.

Six irradiation capsules containing cylindrical specimens of UO_2 for thermal conductivity studies have been discharged after four years of irradiation.

Capsules containing six U-U diffusion couples have been discharged from the reactor. Preliminary examination reveals that the specimens are in good condition and that swelling has been nominal.

The operation of in-reactor swelling capsules has been improved by the addition of automatic control equipment which will maintain temperatures during reactor outage periods. The creep capsules now in the reactor are accumulating exposures before the application of load.

The development of a fast flux monitor containing a thin film of depleted UO_2 is under way. At the present time the results look quite encouraging.

The NPR tube-tube prototype fuel element being irradiated in the M3 loop of the ETR has been recharged and irradiation is proceeding satisfactorily. Examination of irradiated fuel closed by production brazing methods reveals the closure to be of high integrity. No cracks or areas of insipient failure were detected.

Five NPR inner tube prototype elements exposured to 3000 MWD/T appear to be in exceptionally good shape. There is no evidence of surface bumping, clad striations or warp. The brazed closure area appears free of excessive corrosion.

Reconsideration of the design of the NPR fuel supports has given rise to several concrete recommendations. Included are additional inspection, stress relief, and the forming of buggy spring supports to proper height after assembly of the elements.

Autoclave tests of stressed and unstressed Zircaloy-2, rolled strips revealed that zirconium hydride precipitates preferentially in a direction parallel to the rolling direction. The application of stress in various directions had no effect on the hydride precipitation.

The weight gains of etched and pre-autoclaved Zircaloy-2 samples exposed in-reactor to a simulated NPR gas atmosphere at 350 C were about twice that of samples exposed to the same atmosphere out of the reactor. Similar results were obtained for a sample pre-oxidized beyond breakaway and exposed to the gas atmosphere at 400 C.

The KER-1, carbon steel loop was decontaminated. There was no change in the central core temperature of a thermocouple slug after decontamination nor was there an increase in the corrosion rate of the carbon steel as measured by dissolved hydrogen content of the coolant.

A random selection of 102 process acceptable, standard 8-inch I & E fuel elements has been autoclaved 110 hours in steam at 325 C and 2000 psi without failure. Voids in the Al-Si were detectable as depressed areas on the aluminum jacket surface, most of which were on the inner hole.

In preparation for studies of the propagation of brittle cracks in pressure tubes, experiments established the proper projectile shape and powder charge to introduce an initial crack twice the wall thickness in length at the inner surface of pressurized tube samples. This length crack failed to propagate in 50 per cent cold worked, Zircaloy-2 tubes pressurized to 90 per cent of burst pressure at 150 and 300 C. At room temperature, the crack propagated the full length of the sample at a pressure of 80 per cent of the ultimate burst pressure. An initial crack length of eight times the wall thickness propagated 60 per cent of the specimen length in an annealed Zircaloy-2 tube pressurized to 90 per cent of its ultimate strength at 150 C.

The first test irradiation of NPR graphite has been completed at the GETR. Samples were irradiated at 700 C to an estimated exposure of 1.9×10^{21} nvt; about 5700 MWD/AT in the NPR. Contraction of the NPR (TSX) graphite was the same as K-Reactor (TSGBF) graphite and the CSF reference standard: 0.025 per cent transverse to the extrusion axis and about three times this amount parallel to the extrusion axis. Samples from the same capsule at 450 C received an exposure of only 0.7×10^{21} nvt ($E > 0.18$ Mev) and contracted 0.03 per cent in the transverse direction.

Room temperature burst tests were performed on specimens from the annealed portion of PRTR pressure tubes irradiated to exposures up to 10^{20} nvt. Irradiation increased the ultimate strength by 30 per cent. Fracture occurred after 14 to 25 per cent increase in diameter. As the exposure increased, the deformation appears to be confined to local bulging of the diameter in the region of fracture.

Irradiations in the ETR at 600 to 1000 C have demonstrated that high density graphite produced by impregnation with furfural alcohol actually shows an increase in the rate of radiation-induced contraction. When the impregnated samples were heated to graphitization temperature (2500 to 2800 C), the increase in the contraction rate was reduced slightly, but the contraction rate is still higher than that of non-impregnated, lower density samples.

Studies to determine the amount of boiling present when fuel elements are not situated in a coaxial position within process tubes showed that the presence of supports on the heated surface allowed the initiation of boiling at greatly lowered heat fluxes.

A hydraulic investigation to determine the thermal hydraulic effects of a process tube leak caused by a failed front nozzle Vanstone flange disclosed that the leak could be as much as 2-1/2 gpm, resulting in an 11 C temperature rise in a fringe zone tube. It was also concluded that the leak would not always be detected by a change in Panellit pressure.

Tests to determine the effectiveness of water mixers placed in process tubes of the present production reactors showed that a mixer consisting of a 180-degree spiral was very effective in producing even coolant temperatures for the case where the water between the ribs was originally hotter than the rest of the coolant.

Evaluation of four proposed plutonium fuel element designs ($MgO-PuO_2$, ZrO_2-PuO_2, stainless steel clad UO_2-PuO_2, and segregated plutonium-UO_2, 19-rod clusters) for irradiation in the PRTR showed that they will not exceed PRTR tube power, heat flux, or initial reactivity limits.

The document HW-61236 SUP4, "PRTR Final Safeguards Analysis, Supplement 4, Gas-Cooled Loop Analysis," was published. Analyses revealed no serious adverse effect on the nuclear safety of the PRTR.

Four UO_2-PuO_2 19-rod clusters, made by swage compaction, are under irradiation in the PRTR and have accumulated an exposure of about 9 MWD/T. Rods for an additional 30 clusters were loaded during the month, with rods for 18 elements processed through autoclaving.

Results of the accelerated fabrication program were even better than anticipated for the vibrational compaction project. Fabrication of fuel elements containing PuO_2 was started and the process debugged so that four to eight fuel rods were produced per shift. Both the quantity and quality of the resultant fuel elements were excellent thus establishing that it is possible to make vibrationally compacted fuel elements in large numbers despite the additional problems associated with handling PuO_2. Over 200 fuel rods were produced during the month.

Some process control problems were encountered in the early fabrication phases of the 400 UO_2-PuO_2 rods which were loaded and swaged during the month. Variations in UO_2 particle size distribution caused a 200 to 250-gram variation in the amount of oxide loaded into the rods.

X-ray diffraction patterns from ten PuC alloys annealed at 610 C for 65 hours have been analyzed. Between 32 and 41 a/o C, the lattice constants decreased in a uniform fashion with composition extending from 4.9718 to 4.9677 A; below 32 a/o C, the cell size of the PuC phase remained constant at 4.9718 A; lines other than PuC on the films were very weak and appear to conform to a mixture of alpha and delta plutonium; between 41 and 52 a/o C, the lattice parameters were constant at 4.980 A and again Pu_2C_3 was observed at 48.2 a/o C.

Plutonium hydride can be converted to the mononitride by heating in anhydrous ammonia, though first attempts yielded a two-phase product, plutonium dioxide plus plutonium nitride. Further experiments will be performed in an attempt to obtain a single-phase plutonium nitride by this means.

2. Physics and Instrument Research and Development

In support of the production reactor program, the first of 24 gamma spectrometers to be built by GE-APED for the NPR fuel Failure Monitor was thoroughly tested, returned to the vendor for modification of the high voltage supply, and received back for final testing.

Delays encountered in process tube replacements at the production reactors due to breakage of tube splitter blades are expected to be reduced through establishment of blade testing procedures; eight of thirty new blades tested were found to have cracks.

[redacted]

Buckling measurements were resumed on a mockup of the C Reactor lattice. These measurements had been suspended while the plutonium-beryllium neutron sources used to drive the exponential pile were examined, re-canned, and calibrated by Mound Laboratories. The current measurements are intended to provide a basis for determining the change in the physics parameters of C Reactor associated with overboring the process tube channels.

Nuclear safety and control in the handling of enriched uranium will be aided by development of three gamma spectroscopy methods for measuring the enrichment of pieces of uniform or varying sizes and shapes; accuracies up to 2% of the enrichment level are feasible.

Critical mass physics experiments on the reflecting properties of 1/2" and 1" thick paraffin were continued. The measurements this month were made with plutonium nitrate solutions having a concentration of 220 g/l. This is about twice the concentration used previously. The reflector savings determined from the experimental data are about the same as obtained with more dilute plutonium nitrate solutions.

Refined analysis of PCTR data has yielded an improved value of the limiting critical concentration of Pu-239 in aqueous solutions. The new value is 8.4 ± 1 grams of Pu-239/l. The new value is still about one g/l higher than calculated values. The previous experimental value was 9.1 g/l.

In the Plutonium Recycle Program, a series of critical approach experiments has been completed using Pu-Al fuel in light water moderated lattices. The fuel contains 1.8 w/o plutonium in 1/2" diameter rods. Values of the critical mass and material buckling were obtained for five different spacings between rods. The results of the study will be useful in furnishing basic data on plutonium as a fuel for light water lattices as well as for making estimates of the nuclear safety in processing PRTR fuel. This work supplements previous measurements made with fuel containing 5 w/o Pu-Al.

Both zero- and one-dimensional analysis methods were used in analyzing the 5 w/o Pu-Al-H_2O approach-to-critical and exponential lattice experiments. The methods gave results in excellent agreement with experiment in some cases and poor in others. The range of discrepancy in terms of the multiplication constant was 0.2% to 5% in ΔK. The studies pointed out the severity of the neutron spectrum determination problem in these small all plutonium-fueled assemblies.

The planned exposures on two low exposure Pu-Al fuel elements were reached, and the elements were removed from the PRTR during April. The six modified UO_2 elements are scheduled to be loaded in the core during the shutdown around the first of May. Additional burnup data were obtained on the low exposure Pu-Al fuel element 5075; the remainder should be forthcoming in May.

Off-site work in connection with the atmospheric diffusion program at Cape Canaveral, Florida, was completed, embracing a total of 76 experiments during a variety of meteorological conditions. In addition to successfully providing the required diffusion predictions in support of advanced missile firings, the data comprise an important addition to the Hanford and Vandenberg data for evaluating topographic and climatic influences on atmospheric dispersion patterns.

Progress was made in investigations of the effect of wind velocity spectra on mean horizontal dispersion in the atmosphere. Comparison of dispersion data from different sites will be facilitated by this work.

Separation of small variations in three parameters in a simulated multi-layer nondestructive test problem was demonstrated using developmental multiple parameter eddy current test equipment. The equipment will next be applied to separation of three or more test parameters in metal specimens in preparation for application to practical testing problems.

Some temporary difficulty has been encountered in the development of miniature signaling dosimeters due to occasional sticking of the fine quartz recharge fiber to the center rod of the modified pencil ion chamber.

3. Chemical Research and Development

Pilot plant studies indicate that the probable cause of the recent Purex 2D and 2E columns' instability and efficiency problems to be due to unidentified impurities in the aqueous feed to the 2D column. The impurities appear to follow the uranium-bearing organic stream leaving the 2D column and subsequently foul the 2E column.

A new Purex 2E column cartridge has been developed, tested and believed capable of alleviating the current problem of the Purex 2E column.

Second cycle Purex solvent quality is found in laboratory tests to be significantly improved by a carbonate-permanganate solvent treatment vice the conventional carbonate treatment.

Cold shakedown runs of the radiant heat spray calciner, pot calciner and associated off-gas treatment equipment which is to be installed subsequently in heavily shielded hot cells were carried out with synthetic Purex wastes. Tests results thus far are quite favorable.

Uranium dioxide deposits in the form of a 2-inch long, 1-inch diameter right cylinders have been prepared electrolytically from KCl-LiCl melts at 500 C. The deposits are composed of radially oriented columellar grains. Typical bulk densities of 10.8 to 10.9 and lithium impurity contents of less than 10 ppm are observed.

The presence of thallium in plutonium-uranium bearing chloride melts is found to effect a three-fold increase in the PuO_2/UO_2 ratio of the electrolytically prepared deposit.

Dipicrylamine, in comparison with other solvents tested, continues to be the most effective extractant for the recovery of fission product cesium from Purex alkaline waste supernate or from neutralized formaldehyde treated waste.

Preliminary studies to define the path of fission product techneitum in the Purex plant show that about 70 to 80 per cent of the in-process inventory leaves the plant with the HAW stream; about 20 per cent follows the uranium product; and the remainder recycles via the 3WB stream.

Dilute nitric acid (ca. 1 M) was found again to be necessary to initiate dissolution of irradiated Pu-Al alloy fuel material. Once initiated, dissolution of the alloy can be carried to completion at a much higher nitric acid concentration.

In operation of the As-76 prototype monitor at the 100-F Area, interference of the As-76 count by a fission radioisotope, iodine-135, was experienced. Feasibility of using this type of equipment for both As-76 and fuel rupture monitoring is being examined.

4. Biology

The first pigs to show Sr-90 toxicity symptoms in the long-term experiment were noted during the month. These had been exposed utero and fed 625 μc Sr-90 daily for 100 days.

Pigs having received 6.4 μc Ra-226/kg body weight are developing nephritis. Control and Sr-90 animals remain normal in that respect.

DTPA appears much less effective in promoting Np excretion than it is for Pu.

A dog having deposited 4 μc $Pu^{239}O_2$ in its lungs died 2 1/2 years later.

Neosynephrine, Pluronics, and DTPA all markedly increase lung clearance of Ce^{144}-Pr^{144} in rats, with DTPA being most effective.

5. Programming

When all of the possible radionuclides which may have potential for use as heat sources are examined from the standpoint of a few practical criteria, the list shrinks to a total of only six isotopes:

strontium-90
cesium-137
promethium-147
thorium-228
uranium-232
plutonium-238

It is concluded that recovery of promethium-147 should be receiving additional emphasis.

The decay of plutonium-241 results in serious reduction of the exposure attainable with Phoenix fuels at low specific power. In consequence, the fuel life as measured in years passes through a maximum and declines as specific power is increased. Calculations for one case indicated a maximum life of about 15 years for a specific power of about 100 MW per ton of plutonium, decreasing to a life of only 5 years at 10 MW/ton.

TECHNICAL AND OTHER SERVICES

On April 7 about 11 a. m., a nuclear excursion occurred in the K-9 vessel of the Solvent Extraction hood in the Recuplex facility at the 234-5 Building. At the time of the event, there were 22 employees in the building including four persons in the Recuplex area. Automatic activation of the criticality alarm system alerted all persons in the building of the event. Evacuation was prompt and effective.

Surveys of all persons who were in Z-plant were made at the 200-W First Aid Station shortly after the evacuation. No personnel were found to be contaminated. On the basis of surveys made for neutron activation of Na-23 in the body

and statements made by the involved employees, four employees were transferred to the Kadlec Hospital for further tests and observation.

All of the 24 employees in the Z-plant facility were wearing their film badge dosimeters as prescribed. Analysis of these dosimeters, analysis of blood samples and examinations in the Whole Body Counter provided early preliminary estimates of the radiation doses received. For the four men in the room where the excursion occurred, their preliminary dose estimates were known four hours after the excursion. For the other 20 persons in the Z-plant facility, their doses were known six hours after the event. The four men who were were in the room where the excursion occurred received radiation doses of about 110 rems, 43 rems, 19 rems, and 1.4 rems, respectively. All other persons in the Z-plant facility received a dose of < 1 rem due to the excursion except for one employee who was in a nearby office and received 1.7 rems.

The Hanford Laboratories home phone crash alarm system was activated shortly after the incident. Within one hour after the excursion, initial staffing of the Emergency Control Center and emergency supporting Laboratories' facilities were completed. A major portion of the Laboratories' staff in radiation protection, nuclear physics, and analytical laboratories functions were assigned to duties in support of the stabilization of the Recuplex facility during April. Throughout the recovery period normal Hanford radiation protection control limits were in force.

Five senior members of the Laboratories' staff served on the Recuplex Advisory Council or Investigating Committee.

Iodine-131 emitted from the Purex stack during the period April 2 to April 12 totaled about 44 curies. The high emission is believed to have resulted from dissolving "green" metal. Winds of varying speeds and directions dispersed the contamination so that none was detected except in the immediate vicinity of the 200 East Area.

Average fallout concentrations at various localities in the Pacific Northwest ranged from 2.4 to 13 $\mu\mu c/m^3$ during the month of April. Sporadic increases noted during the first week of April appeared to be the first evidence of the spring peaks expected from the USSR testing last fall.

Three minor cases of plutonium deposition, each less than one per cent of the maximum permissible body burden, were confirmed by bioassay analyses during the month. The total number of plutonium deposition cases that have occurred at Hanford is 288, of which 208 are currently employed.

Mathematical expressions have been developed which express the self-interaction coefficients of annulus-shaped containers with interior radial baffles. Such containers are used for storing fissile materials, and their self-interaction coefficients are important in criticality studies.

A formal solution has been obtained to the problem of determining the steady-state non-viscous flow pattern of a fluid in a cylindrical tank which has been equipped with an axially located circulating device.

Sets of experimental data are being collected to test the appropriateness of the tentative format for the creep raw data portion of the mechanical properties raw data file.

A statistical evaluation of plutonium analyses on eight types of plutonium oxide pellets was completed.

SUPPORTING FUNCTIONS

PRTR output was 1044 MWD for a plant efficiency of 49.7% and a total experimental time efficiency of 51.7%. Accumulated exposure through April 30 is 7261 MWD. Additional exposure information is as follows:

Maximum UO_2 exposure/element	2240 MWD/TU
Average UO_2 exposure/element	1518 MWD/TU
Maximum Pu-Al exposure/element	73.7 MWD
Average Pu-Al exposure/element	44.5 MWD
Maximum Moxtyl exposure/element	12.6 MWD
Average Moxtyl exposure/element	10.2 MWD

The ninth refueling was performed on April 2. Three new Pu-Al elements containing high exposure plutonium and three new elements containing a mixture of PuO_2 and UO_2 (Moxtyl) were charged in place of UO_2 elements. Additionally, an LX Pu-Al element and an additional Moxtyl element were charged in mid-month. The performances of the HX and the Moxtyl elements have been satisfactory, with no unusual operating problems.

A preliminary study was made of the hazards associated with the transition from the 3-zone UO_2-Pu-Al loading to a 2-zone UO_2-Moxtyl loading. It was concluded that the proposed change would be safer with respect to an increase in the negative temperature coefficient and would not change the delayed neutron fraction significantly.

[redacted]

Work continued on the project exceptions for the Plutonium Recycle Critical Facility. Safety rod cadmium plating was tested and found to be satisfactory. The rods were installed and drop tested satisfactorily. The Log-N channels triple trip unit was returned from the vendor and tested satisfactorily. Preliminary planning was started on the H_2O moderated second generation experimental program. Current efforts include scope, budget, and scheduling definitions.

Over-all construction for the Fuel Element Rupture Test Facility is estimated at 94% complete. CPFF work is 97% complete and the water plant is 88% complete. Minor design changes were initiated to alleviate potentially hazardous conditions. Design for shielding of the piping and heat exchangers is approximately 60% complete.

The project for the gas cooled loop is 91% complete. The vendor started fabrication of the heater. The blower vendor indicated satisfactory progress toward mid-May testing of the blowers. Formal training of all PRTR operating personnel was initiated April 19. Some work continued on procedure preparation and manuals, etc.

Authorized funds for 11 active projects amount to $2,721,600. The total estimated cost of these projects is $7,730,000 of which $1,274,000 has been spent through April 30, 1962. Project proposals covering a Low Level Radiochemistry Building and Geological and Hydrological Wells - FY 1962 were submitted to the AEC in April.

PRTR operating costs in April included heavy water charges of $21,459 to cover losses of $18,794 and scrap of $2,665. This month's loss represents a significant increase compared with March loss of $7,650. It is expected that scrap valued at $210,000 will be shipped to Savannah River in the latter part of May.

Approximately 2,300 hours of overtime were worked by Hanford Laboratories exempt (excluding administrative) employees in connection with the 200-W criticality incident. Costs aggregating $15,729 have been billed to CPD for technical assistance provided by Hanford Laboratories. Costs for Radiation Protection services and administrative personnel are not included in the above amount.

Recruitment - Four Ph.D. applicants visited HAPO for employment interviews. Four offers were extended; one acceptance and three rejections were received.

This month 50 program offers and 10 direct placement offers were extended; offers accepted included 27 program and four direct placement; rejections came from 54 program offerees and five direct placement prospects.

Technical Graduate Program - Six Technical Graduates were placed on permanent assignment; one new member was added to the roll and one man terminated. Current program members total 38.

W. H. Rea

for Manager
Hanford Laboratories

HM Parker:WHR:st

REACTOR AND FUELS RESEARCH AND DEVELOPMENT OPERATION

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 2000 PROGRAM

1. METALLURGY PROGRAM

Corrosion Studies

Effect of Heat Treatment on Hydrogen Pickup. The effect of heat treating Zircaloy-2, low nickel Zircaloy-2, a heat of ultra fine grained, low nickel, Zircaloy-2, Zircaloy-4 and crystal bar zirconium on hydrogen pickup during corrosion in 400 C, 1500 psi steam is being investigated. Each alloy is being tested in the following heat treated conditions: β quench, $\alpha + \beta$ quench, α anneal, and as-cold rolled.

Pre-transition percent corrosion product hydrogen pickup for Zr-2 is about 10 percent for samples quenched from the β or $\alpha + \beta$ phases and about 20 percent for samples in the as-rolled or α annealed condition. Moreover, there is no increase in hydrogen pickup after transition for β and $\alpha + \beta$ quenched samples, unlike the as-rolled and α annealed samples for which hydrogen pickup increased to 40-50 percent. The hydrogen pickup for low nickel Zr-2 and Zr-4 are similarly affected by heat treatment, though the difference noted prior to transition are less significant.

Directional Precipitation of Zirconium Hydride. Investigations to determine factors affecting the orientation of zirconium hydride platelets in Zircaloy-2 have been conducted. Zircaloy-2 strips plastically deformed 66 percent reduction in thickness by cold rolling were autoclaved in 450 C steam. A 15,000 psi tensile stress was applied in one of three directions, parallel, normal, and diagonally, to the direction of plastic deformation. All samples were autoclaved for a sufficient length of time to introduce approximately 150 ppm of corrosion product hydrogen into the Zircaloy-2.

Metallographic cross-sections of the hydrided samples demonstrate that an oriented precipitation occurs parallel to the direction of rolling. The applied stress appeared to have no effect on the directional precipitation of zirconium hydride. Unstressed samples exposed simultaneously, were found to have the same pattern of hydride precipitation as the stressed samples.

Corrosion of Beryllium Brazed End Closures and Copper Bonded End Closures. Additional corrosion results on beryllium brazed end closures and copper bonded end closures have been obtained. Corrosion for both the welded and unwelded beryllium brazed closure exposed to 360 C water was approximately 0.3 mil following three months of exposure as compared to 0.2 mil for the first 1-1/2 months of exposure. The penetration on the samples exposed to 400 C steam is approximately 0.4 mil following 2-1/2 months of exposure as compared to 0.25 mil for the first 1-1/2-month exposure.

Initial corrosion results on the copper bonded and welded end closures have shown good corrosion resistance in both 400 C steam and 360 C water. Uniform penetrations of 0.05 mil in 360 C water and 0.06 mil in 400 C steam were measured following one month of exposure. However, small portions of the copper contaminated welded area showed local penetrations of 0.65 mil in both test media.

Slug Rupture Test. The first batch of 102 standard 8-inch I & E fuel elements have been autoclaved for 110 hours in 325 C, 2000 psi water without rupture. These elements represented a random selection of recent, acceptable production fuel elements. Voids in the Al-Si were detectable as depressions in the can surface after autoclaving. The fuel elements have been returned to FPD for further measurements and correlation of bond test results with Al-Si voids.

Non-Isothermal Loop. Initial loop tests with coupons have been operated successfully at 340 C and 3000 psi for ten days. However, the failure of one heating element apparently introduced magnesium oxide (and high pH) into the loop water. An attempt will be made to replace the damaged heating element.

Additional valves and welded fittings have been ordered so that upon modification, the loop may be operated isothermally, by isolating the heat exchanger units.

Metallurgy Studies

Zirconium Components for Coextrusion. To determine the effects of both grain size and extrusion coefficient on interface roughness, several metals will be coextruded along with uranium in a composite billet assembly. At the same time a direct comparison between the interface roughness of alpha extruded and beta heat treated-alpha extruded uranium will be made.

Slots were machined along the length on the outer surface of a uranium billet for an N-outer extrusion. The billet was then cut in half and one half beta heat treated. The two halves were butted together and strips of copper, brass, mild steel and Zircaloy were aligned in the slots. The Zircaloy strip was immersion plated with copper prior to assembly. The assembly was canned for extrusion in the regular manner. The extrusion coefficients of the metal strips bracket those of uranium and Zircaloy-2 going from low to high in this order: copper, uranium, brass, Zircaloy-2 and mild steel. The grain diameters of the strips ranged from 10 to 80 microns. After extrusion the interfaces between both the uranium and Zircaloy and the test metals will be analyzed.

The significance of a "memory" effect in Zircaloy coextrusion components from a prior forming operation was recently observed. Zircaloy inner clad inner fuel components fabricated by a rolling process were coextruded. The resulting inner clad had four equally spaced ribs with heights over twice the clad thickness protruding into the uranium core.

Metallic Fuel Development

Fuel Irradiations. The NPR tube-in-tube fuel element which is being irradiated in the GEH-M3 Loop in the ETR has been recharged and irradiation is proceeding satisfactorily. The inner tube now being irradiated has "suitcase handle" rather than "buggy spring" supports. At the completion of this irradiation period, the fuel element will be discharged and the test terminated. The average total exposure on the outer tube at that time should be approximately 1200 MWD/T.

Radiometallurgical examination is continuing on the production brazed irradiation test from the MTR, GEH-4-63 and 64. Both end closures on GEH-4-64 have been sectioned to determine their integrity. Both closures appear to be in good condition with no cracks or indications of any incipient failure in the brazed region. Cracking occurred in the uranium near the end of the tube located in the lower flux region, but none of the cracks extended either to the cladding or to the brazed end closure.

The variable braze thickness irradiation test, GEH-4-68, 69 and 70, is nearing completion of tis sixth and last cycle of irradiation in the MTR. The total accumulated exposure at the end of the irradiation should be approximately 1360 MWD/T.

Five NIE fuel elements (NPR inner tube prototypes) have been examined and weighed in the basin for swelling determinations after attaining 3000 MWD/T exposure. The superficial appearance of the elements is good, and there is no evidence of surface bumping, clad striations, or excessive warp. The Zr-Be brazed closures appear free of excessive corrosion. Average fuel swelling is 1.7 ± 0.5 v/o. The elements will be shipped to the Radiometallurgy facility for additional evaluation.

Four 17-inch NIE fuel elements, which were used as heaters in the thermocoupled crud test charge, have been evaluated in the basin following discharge. The closures were white, apparently the result of operating in coolant somewhat hotter than usual. Fuel swelling for these 750 MWD/T exposure elements was 0.3 ± 0.1 v/o.

Two NPR fuel assemblies operated as a throw-away charge in Loop KER-3 were discharged after attaining an exposure of 420 MWD/T. Determinations made on the two inner and two outer fuel tubes showed no measurable fuel swelling.

Examination of a coextruded Zircaloy-2 clad fuel element which ruptured at about 1700 MWD/T in a KW Reactor production channel has not revealed the cause of failure. However, examination of the cap indicates that a buildup of corrosion products between the cap and the fuel is the most probable mechanism by which the cap was dislodged from the fuel element.

Early this month, KER Loop 1, which contained a thermocouple probe, was decontaminated. Following the outage, the probe operated normally for two days, indicating that no crud film had been deposited on it. The thermocouple probe then failed, terminating the experiment. Total exposure on the thermocouple probe at the time of failure was 900 MWD/T. Basin examination showed that the end cap (at the thermocouple end) was broken from the rest of the fuel rod. The end cap has been examined visually in Radiometallurgy. Longitudinal cracks were observed in the cladding attached to the end cap and radial cracks were seen in the outer surface of the end cap. The cause of the failure has not yet been determined.

A 36-inch long KER-size tube-in-tube fuel element which was irradiated in the GEH-G7 Loop at the ETR has been examined by Radiometallurgy. This fuel element had a natural uranium-2 w/o core. Metallography was completed on sections at the midlength of both the inner and the outer tubes, and burnup and

density measurements were made at the midlength of both tubes and at every three inches from the midlength to one end of the inner tube. The metallography sample at the midlength of the inner tube showed the presence of what appears to be a spheroidized second phase. The calculated temperatures were high enough (575 C) for spheroidization to have occurred. This second phase was not seen in the metallography sample from the outer tube where temperatures were lower. A density decrease of approximately two percent occurred in the inner tube at a burnup of 1800 MWD/T. The swelling data indicate that the observed fuel expansion followed that predicted by the perfect gas laws except that the effective pressure restraining the fission gas expansion was approximately twice the 2000 psi loop pressure. The additional apparent pressure is due, in part, to restraint offered by the cladding and in part to the surface tension forces associated with the gas pores.

Four cladding studies capsules were discharged from DR Reactor at an exposure of 1500 MWD/T and are scheduled for examination by the Radiometallurgy Operation.

Examination of one Zircaloy-2 clad uranium rod recently irradiated in the MTR showed a cladding defect similar to the necking observed on several other Zircaloy-2 clad rods. However, a transverse cross section shows that a crack has propagated from the inner surface of the cladding to the surface, with little plastic deformation at the fracture except where it terminates at the clad surface. The fracture did not propagate into the uranium.

Fabrication of components for a second irradiation test of fuel rods with non-uniform thickness of Zircaloy-2 cladding continued. Preliminary assembly and welding of the capsule bodies is complete. Eighteen Zircaloy-2 clad rods were coextruded with intentional striations of controlled depths at the fuel cladding interface. These rods have been heat treated and are now being cut for irradiation samples.

Fuel Deformation Studies. To investigate the relative swelling of uniformly enriched and "dual" uranium fuel elements operating at equivalent power generation and surface temperatures, three capsule-type experiments are being assembled for irradiation in the MTR. Assembly and measurement of the fuel samples are complete. To measure the non-uniform straining of the cladding during irradiation, a fine grid is being etched on the cladding surface.

Several irradiated 0.593-inch diameter cladding specimens were tested to failure at room temperature by application of a pure hoop stress with an expanding mandrel inside the cylindrical specimens. Very little plastic deformation could be observed in a visual examination of the failed specimens. Detailed examination and measurement was started.

Fluted Fuel Development. The design of a single tube, fluted element was completed. The outer surface of the element will have eight flutes about 0.1 inch from valley to crest. The tangential arcs which form the flutes have a radii ratio of about one-tenth. This results in a shape that resembles an octagonal prism with rounded corners and slightly concave sides. The central bore is cylindrical. The nominal dimensions of the element are 2.327 inches OD and 1.286 inches ID. The element will be fabricated by coextruding 0.95 percent enriched uranium and Zircaloy-2 cladding. The design of the extrusion die for this purpose was completed.

Fuel Closure Development. In an effort to find less expensive and better brazing materials for the NPR fuel closures, several pure metals are being studied for their ability to form a bond or a lower melting brazing alloy by diffusion with the Zircaloy end cap. One of these metals is pure copper. Copper cups have been made to fit Zircaloy caps and then assembled into the chemically-milled end of a Zircaloy-clad fuel element. The end of the element is then heated in vacuum to 1050 C for 30 seconds, using induction heating. Fifteen psi mechanical force is applied to the end cap and the chamber is back-filled to one-half atmosphere of argon to collapse any voids in the braze. Metallographic examination shows that this method is capable of producing sound bonds between the Zircaloy cap and both the uranium and the Zircaloy cladding. Further studies will determine the strength and corrosion resistance of the bond.

Hot Head 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 end of the fuel element.

A modification of the welding projection has been made on the end cap. Requests have been issued to make a quantity of these caps, and it is anticipated that the machine shop will supply these within several days.

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.

A series of exploratory welds were made using assemblies consisting of two Zircaloy caps which had the same size and shape as the fuel element end cap. One disc contained projection welds while the other disc had flat surfaces. Sn and Cu foil preforms were used as interface materials. Subsequent to projection welding, heat treatment was accomplished in the welding machine by varying the number of heat impulses while holding other parameters constant at arbitrary values.

Cursory metallographic examination indicated that the Sn samples and most of the copper samples were unsatisfactory. These results were believed to have occurred because of excessive heat generation. Results also indicated that the Cu preforms were too thick. One welded sample indicated that sufficient heat can be generated with the resistance welding equipment to make a complete welded joint across the entire interface of the caps. Although this demonstration may indicate that discs and bars with similar cross sections may be joined, it may not be applicable for fuel elements because of the amount of heat generated and because of accompanying noticeable deformation.

Previous experimental work has led to the conclusion that the most promising approach to the problem of producing consistently sound bonds at the closures of NPR elements by the "self-brazing" process is to encase the element in a ceramic lined restraining die, while supplying heat and axial load by means of the heavy-duty Sciaky spot welder. A principal difficulty is the lining of the restraining die. This lining must be hard, smooth, essentially water-free, and capable of retaining its electrical non-conductance at a temperature of 700 C. It must also adhere firmly to the interior wall of the die but allow the upset end of the element to be pressed out of the die without seizing. Several proprietary ceramic coatings and other "home-made" lutes have been explored. One mixture that has most of the desired properties consists of equal parts of Alundum cement and Portland cement mixed with Sauereisen thinner to the consistency of texture paint. A smooth mandrel of proper diameter coated with Aquadag is inserted in the loosely fitting bore of the die, which is plugged at one end and charged with a slight excess of the lute material. The mandrel serves as a piston, forcing the lute up into the annulus. After the lute has set, the mandrel is carefully pressed out of the die, leaving an Aquadag-lubrciated

ceramic liner with smooth walls. Finally, the lined tube is dried and baked to harden the ceramic coating.

Copper Welding. Although the porosity encountered in welding copper with nitrogen as a shielding gas has not been eliminated, it has been markedly reduced. The major reduction is caused by the use of silicon copper. Slight further improvement is made by the combination of gas flow, torch position, travel speed, and welding current. Welds on silicon copper show some porosity. It does not appear to be enough to cause leakage into a welded and evacuated coextrusion billet, but this remains to be determined. Superior penetration is evident.

NPR Fuel Support Development. A number of problems involving fabrication, height control and fatigue resistance of supports for the NPR inner fuel tube are under investigation.

Cracking of the supports at the bends during fabrication is a major problem. On the basis of test runs, the following procedure offers a solution to this problem:

1) Stress relief anneal the partially formed supports prior to forming to finished dimensions.

2) Give the heat treated supports a 100 percent XyGlo inspection.

3) Form the supports to finished height.

The prescribed interference fit between the inner tube supports and the inner surface of the outer fuel tube is difficult to achieve. Test runs on a proposed solution to this problem are encouraging. The supports are installed so their height is less than the width of the annulus between the inner and outer fuel tubes. The autoclaved inner fuel tube is then assembled inside the outer tube and an interference fit between the supports and the outer tube is produced with a special tool that increases the height of the supports by shortening the length of the crowned section.

Failure of inner tube supports in the ETR and high speed movies of the behavior of the inner fuel tube in an ex-reactor loop has caused concern over the possibility of fatigue failure of the supports in the N-Reactor. The suitcase handle type support has not revealed this susceptibility to fatigue failure, presumably because its spring constant is five or six times greater than

that of the buggy spring support. A fuel tube assembled by sizing up the supports in place as described previously had a spring constant seven times greater than that produced by the old method of assembly. This spring constant is even larger than that of a suitcase handle type support, yet all the advantages of the buggy spring type are retained.

Supports - Dummies, KER Loops 3 and 4. Present supports on the carbon steel dummies for KER Loops 3 and 4 are hard steel ball bearings, plug welded in place. The balls are spaced four around at each end of a 16-inch long dummy. These supports are unsatisfactory for use in the loops because of their poor wear characteristics, fretting potential due to possible chattering, and their inability to span discontinuities in the process channel. A soft low carbon steel support similar in shape to the N outer fuel tube support is being fabricated for use on the dummies. The N outer fuel support dies were modified to fabricate this support. It is made from 16 gage (0.061 inch) sheet steel and resistance welded to the dummies. It is planned to have sufficient information (strength characteristics, chemical composition, and wear properties) available by the next reactor outage so dummies bearing this support configuration can be charged.

Uranium Billet Heat Treatment. Forty-three NOE, twenty-four NIE, and eight half-section NOE uranium coextrusion billets were beta heat treated as a final fabrication step before coextrusion. The treatment consists of a simple beta phase heating and quench. Coextrusion performance of NOE billets from the alpha annealed and single beta heat treated N process ingot was found to be generally the same as billets prepared from triple beta heat treated ingots. The second ingot was single beta heat treated, without the prior alpha soak, and has been scheduled for primary extrusion.

Cleaning and Pickling. A study to compare the diffusion of metallic lubricants, copper versus low carbon steel, into Zircaloy-2 cladding during coextrusion was initiated. The purpose of this study is to evaluate the diffusion of copper and low carbon steel (ASTM-A-83-60T) during extrusion at 750 C and compare the findings with diffusion at the standard 600 C coextrusion temperature. The procedure being followed is to take short sections of 0.6-inch diameter Zr-2 rods, produced by extrusion in a copper or steel container at a reduction ratio of 13.6 to 1, and remove the metallic lubricant by various chemical means. The cleaned sections are then etched in standard $HF-HNO_3$ etch solution for 60 seconds. The

etch solution is analyzed by colorimetric means to determine the concentration of either copper or iron. Successive layers of Zircaloy are removed until the layer removed has the standard composition of Zircaloy-2. Preliminary results show no appreciable diffusion of copper occurs at 600 C. The diffusion of copper at 750 C is approximately 130 ppm for the first mil of surface. The means by which the metallic lubricant is chemically removed has a tremendous influence upon the copper concentration of the Zircaloy surface. If only the standard hot HNO_3 strip bath (60 percent HNO_3 at 60 C to 70 C) is used, "average" copper concentrations at 200,000 ppm or greater were obtained from etching the first half mil of surface. However, if the standard hot HNO_3 strip bath is followed by a 10-minute dip in room temperature HCl, and then by hot HNO_3 for one to two minutes the low copper concentration reported above can be achieved.

To date, only limited investigations have been conducted on the iron clad material. Cursory evaluation shows the iron diffusion at 750 C to be small and not to penetrate more than one to one and one-half mils.

Zirconium Alloy Studies. Initial corrosion data were obtained for the series of zirconium alloys of two oxygen levels and varying tin content. Examination after 14 days in 750 F - 1500 psi steam showed all the alloys except the low oxygen pure zirconium to have adherent black oxide films and weight gains of approximately 22 mg/dm^2. All the alloys after 10 days in 680 F water had black adherent oxide films. The tests are being continued to approximately 130 days of exposure.

Die Lubrication. It has been found that the addition of a thin film of lithium stearate, applied over the asphalt coating, improves the lubricating properties of the asphaltum to a considerable extent. In drawing Zr-2 wire it has been observed that the application of a lithium stearate film allows for a 300 percent increase in the drawing speed of the wire.

An asphalt surface is not normally wetted by water, but when a stearate coating is applied, a thin film of water will adhere to the surface. The die coolant water spray produces such a film on the wire as it is drawn into the die. The cooling effect of this water within the die acts to increase the film strength of the asphalt. The stearate also reduces the coefficient of friction between the carbaloy die and the wire.

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2. REACTOR PROGRAM

Corrosion and Coolant Systems Development

Hydriding Corrosion Capsules. The weight gains for Zircaloy-2 and Zircaloy-4 samples exposed in-reactor for 167 days to a simulated NPR gas atmosphere at 400 C were higher by a factor of 2 to 3, than comparable samples exposed to the same atmosphere out-of-reactor. Three pretreatments were employed: etched, autoclaved 3 days in 400 C steam and autoclaved 56 days in 400 C steam. These pretreatments caused no difference in relative weight gains of in-reactor and out-of-reactor samples. Samples exposed in-reactor at 300 and 350 C are in the process of being removed from the capsule and weighed. All samples will be analyzed for hydrogen content.

Electrical Resistance of ZrO_2 Films. The rate of change of electrical resistance of ZrO_2 films on Zircaloy-2 were measured at 350 and 450 C in water vapor and vacuum. The resistance rise when oxidizing in water vapor was higher at 350 C (2 meg ohms) than at 450 C (200 K ohms). The resistance dropped below 100 ohms in vacuum for both temperatures. The rates of rise and fall have not been reproducible which may be indicative of a non-reproducible electrical contact resistance.

Another hydriding experiment was conducted at 350 C where electrical resistance of the ZrO_2 film and hydrogen absorption were measured simultaneously. The electrical resistance dropped below 100 ohms and 2-1/2 days later accelerated hydriding occurred agreeing with earlier results at 450 C that hydriding occurs when the electrical resistance of the oxide is low.

NPR Pump Sleeve Test. An NPR pump sleeve fabricated from 17-4 PH, H-900 stainless steel has been exposed to simulated NPR conditions for one month; 120 C, 250 psi, pH 9(LiOH), low oxygen. A fluorescent dye penetrant test after this period did not show any cracks caused by a corrosion process. The pump sleeve will be further examined at one month intervals.

Thermocouple Slug. The KER-1 Loop was decontaminated during the outage of March 30, using two treatments with Phos-1 on both the in-reactor and ex-reactor portions. On April 4, the thermocouple slug ruptured after four days of operation at pH 10 (LiOH) following the decontamination. The delayed neutron monitor rose instantaneously to full scale. The clad of the thermocouple slug was found to have ruptured completely around the circumference near

the fuel end-cap junction. Visual examination of the Zr-2 clad thermocouple slug and heater elements showed no crud deposition. No significant change was noticed in the temperature drop between the center of the slug and the water before and after decontamination.

Dynamic Corrosion of Uranium. Four runs (two at 100 C and two at 130 C in water) were made in TF-9 during the month to determine the weight loss of a special bare uranium assembly. After corrosion rates have been established, a similar element will be used in H-Reactor to provide a controlled rate of activity release for testing of rupture detection instruments. The fuel assembly consists of thin wafers of uranium to give a total surface area of 2000 square centimeters. The two five-hour runs at 100 C produced a weight gain of about 3 grams. The two five-hour runs at 130 C produced a weight loss of 76 and 116 grams, the difference being that the oxide was thick enough to slough off at the beginning of the second run. Several more runs will be made at lower temperatures.

Hydrogen Generation and Analysis Studies. A total of seven samples of KER-1 coolant were analyzed for hydrogen content during the month. Before the system was decontaminated, the water contained 1.5 to 2.5 cc/l (STP) and after decontamination, an average of 5 cc/l (STP). Although these results show a definite increase in dissolved hydrogen concentration in the coolant following decontamination, the concentrations are still well below the 8 to 10 cc/l (STP) measured immediately after the fuel charge was inserted in the loop. Analysis of these results indicate that corrosion of the carbon steel piping in the loop was not greatly accelerated by the decontamination process used.

A colorimetric procedure for measuring dissolved hydrogen concentration is being evaluated. The procedure involves mixing a coolant sample with a solution of methylene blue and palladous chloride. The palladium activates the hydrogen which in turn reduces the methylene blue to a colorless complex. This procedure is affected by the presence of other oxidizing or reducing agents and would be useful only during equilibrium operating periods. Data obtained to date are too preliminary to determine whether or not the procedure is accurate (or specific) enough for our needs.

Single-Pass Decontaminant for NPR. The search for a reagent for single-pass decontamination of NPR resulted in the choice of ammonium citrate after evaluating bisulfate and sulfamic cleaners with the citrate. Laboratory results were good. An average DF

of 100 was obtained with a corrosion of carbon steel rate of 0.04 mil per hour at 85 C and 95 C.

Nozzle Snap Rings. Corrosion tests of nozzle snap rings fabricated from Inconel and Inconel-X are continuing in 57-60 C, filtered Columbia River water. No change in appearance has occurred during the past month.

Samples of Inconel and Inconel-X stressed to the yield strength were removed from boiling 42 percent magnesium chloride after five weeks of exposure. No stress corrosion cracking was found. Stressed 304 stainless steel controls failed within one week. A few pits were noted on the Inconel while the Inconel-X appeared in excellent condition.

Corrosion Tests in pH 10 Water. A corrosion test in TF-4 at NPR primary system inlet temperature in pH 10 (LiOH) deionized water is in progress. After 1200 hours of exposure, corrosion rates of 0.2 mil/year on A212 carbon steel and 0.016 mil/year on 304 stainless steel. No weight changes have been observed on Zr-2. The carbon steel corrosion rate is comparable to that obtained in 300 C pH 10 water.

Corrosion Inhibitor. Screening tests are performed to evaluate the effectiveness of several additives in inhibiting the corrosion of aluminum and steel in room temperature static water. The first group of aluminum and carbon steel samples have been discharged from beaker tests of sixteen different inhibitor combinations. Corrosion of carbon steel samples continues to spread across the metal surfaces in all inhibited solutions, but at different rates in different solutions. Tubercules are forming on some aluminum samples in some solutions, and aluminum samples are beginning to tarnish in other solutions. After seven weeks of exposure the five most promising mixtures are: (1) 100 ppm sodium nitrite; (2) 100 ppm sodium chrome glucosate; (3) 50 ppm sodium dichromate plus 50 ppm Calquartz 1645 (sodium silicate); (4) 50 ppm sodium nitrite plus 50 ppm sodium silicate (Na_2O - 3.25 SiO_2); and (5) 100 ppm sodium hexametaphosphate (calgon).

Visual examination of these samples is not complete. However, total weight losses confirm the observations that sodium nitrite and sodium hexametaphosphate are promising inhibitors for both aluminum and carbon steel, the mixture of sodium nitrite plus sodium silicate is a promising inhibitor for aluminum, and sodium chrome glucosate is a promising inhibitor for carbon steel.

Structural Materials Development

Burst Tests of NPR Pressure Tubes. Burst tests were performed on three sections of NPR pressure tubes. The samples were selected from tubes that had received 35 percent cold work (% CW) in the final forming step. The tubes were autoclaved at 425 C for three days prior to being sectioned for samples. The average burst pressure was 11,600 psi at 300 C compared to 10,000 psi for the 18% CW tubes. The circumferential elongation varied from 11 to 26% for 35% CW and 15 to 33% for 18% CW. The maximum local elongation measured on a grid pattern was 45% for the 35% CW tubes.

Burst Test of Irradiated Pressure Tubes. A prototype unit for burst testing irradiated pressure tubes at elevated temperature is under test. Two steel tubes were tested with circumferential grooves to induce a circumferential break. The sudden formation of steam projects the specimen into an aluminum pad set in the top of the furance. The shock of the impact is absorbed by the plastic deformation of the aluminum thus preventing shock loading of the containment vessel. Electrical resistance strain gages were used to monitor the strains induced in the wall of the containment vessel. Maximum strains caused by the release of pressure and the projectile striking the top of the furnace were equivalent to the strains caused by a static pressure of 215 psi, well below design conditions.

Non-Metallic Materials

Graphite Burnout Monitoring. Burnout rates calculated from measurements on small monitoring samples irradiated in channel 2577 in DR Reactor from 1-7-62 to 3-31-62 show the familiar burnout profile. Peak rates measured on two samples located 96 and 100 inches in from the front of the graphite stack were 34.4 and 107 percent per 1000 operating days, respectively.

A computer program for calculating graphite burnout rates based on static equilibrium data was developed by G. E. Zima (see HW-71737). This program was used to calculate burnout profiles which were compared with experimental findings at KE and D Reactors. The basic equation involved is:

$$R = K(A \quad \Delta C1/\Delta X)^m$$

where R is the burnout rate in wt % per 1000 operating days per cfm of inert gas flow, K and m are constants and $\Delta C1/\Delta X$ is the

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change in the carbon load of the gas over a given zone length. The carbon load is the ratio of the carbon content of the gas (as weight of carbon) to the weight of inert gas. A is a factor relating to the geometry of the system.

The calculated burnout profiles show peaks forward of the stack center line. By varying the value of A and estimating the flow in the burnout channel, agreement with observed peak burnout can be obtained. However, thus far the calculated peaks appear broader (at KE and D) than the observed peaks; also the maximum at D Reactor is closer to the stack center line than is calculated. Furthermore, in the area downstream from the maximum graphite temperature the program predicts high carbon deposition rates, which are not observed. A value of $m = 2$ appears to give a better correlation than $m = 1$, a result similar to that reported by Zima in comparing laboratory burnout rates with the program.

Radiation Damage - NPR Reflector Graphite. The properties of NPR reflector graphite from the Great Lakes Carbon Corporation (Grade GLC-R1) and from the National Carbon Company (Grade AGOT-LS) were measured and reported in HW-66996. The first irradiation data at approximately 500 C have now been obtained on these graphites after an exposure of 6400 MWD/AT in KW Reactor. As shown in the table the dimensional changes were virtually the same for these grades.

Dimensional Changes After 6400 MWD/AT at 500 C

Length Change, %

Grade	Transverse	Parallel
GLC-R1	- 0.041	- 0.099
AGOT-LS	- 0.037	- 0.109

NPR Core Graphite Irradiations. Preliminary post-irradiation results have been obtained for the graphite samples from the first capsule, H-4-1, in the series of long-term irradiations of NPR graphite in the General Electric Test Reactor. The samples were irradiated at temperatures in the range 450 to 700 C. Maximum exposure is estimated to be 1.8×10^{21} nvt ($E > 0.18$ Mev), which is equivalent to approximately 14,000 MWD/AT in C Reactor and 5700 MWD/AT in the NPR. Exposures are based on calculated PDQ average fluxes with the distribution based on flux monitors from the previous EGCR graphite irradiation

experiments in a different position in the GETR. Verification of exposures from flux monitors at each sample position is in progress.

The contraction of samples irradiated at 700 C at the peak exposure of 1.8×10^{21} nvt (E > 0.18 Mev) was approximately 0.025 percent for NPR (TSX) graphite, K-Reactor (TSGBF) graphite and the TSF reference standards samples cut transverse to the extrusion axis. Samples cut parallel to the extrusion axis contracted at a rate two to three times the transverse samples. The contraction rate of transverse samples irradiated in the same capsule at 450 C to an exposure of 0.7×10^{21} nvt (E > 0.18 Mev) was about twice the rate at 700 C.

The second capsule in the NPR graphite irradiation series, H-5-1, has successfully completed the third cycle of reactor operation with all nine thermocouples functioning properly. The third capsule in the series, H-6-1, has completed the first cycle of operation. All nine thermocouples are operating satisfactorily. The plastic materials in the seal pot have shown no evidence of failure. Apparently the shielding provided by the lead in the annular space between the leadout tubes has been satisfactory.

Thermal Hydraulic Studies

Thermal Hydraulic Characteristics of I&E Fuel Elements in a K-Reactor Process Tube. Laboratory experiments were performed to extend the knowledge of hydraulic demand data for I&E fuel elements in K Reactor. These data, which express the relationship between pressure drop and flow at a constant tube power, are useful in establishing the flow trip requirements of the process tube pressure gages. These data are an extension of previous work performed for a different sized fuel and at slightly lower power levels.

The test section simulated a 38-piece charge of K-IV-N fuel elements in a K Reactor 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 a cosine heat generation along the length of the test section.

Experimental procedures consisted of first establishing an equilibrium flow rate which resulted in an outlet temperature of 130 C at a given tube power. Then, while the heat generation was held constant, readings of pressures, temperatures and flow

were obtained as the flow was slowly reduced in a stepwise manner. Each run was terminated at boiling burnout conditions as detected by a large temperature surge at some point on the heater rod.

Data were obtained for runs at 500, 1000, 1500, 2000, and 2300 kw. A preliminary analysis of the data indicated a favorable comparison with the data obtained previously and reported in HW-64029. One interesting occurrence, which was also observed in the previous tests, was a flow fluctuation between the hole and annular flow passages under boiling conditions during the 500 and 1000 kw runs. This has tentatively been attributed to the following process. Either the hole or annular flow passage experiences an unstable flow condition soon after boiling starts in the test section. This causes a flow reduction in the unstable flow passage with a corresponding flow increase in the other passage. Film boiling conditions are then established in the unstable flow passage with a corresponding flow increase in the other passage. Film boiling conditions are then established in the unstable flow passage and with the accompanying decrease in the heat transfer coefficient some of the heat transfer is shifted to the other flow passage. With a sufficient shift in heat transfer, the flow conditions are no longer unstable in the stricken passage and flow conditions return to normal. This in turn increases the heat transfer coefficient and when the heat transfer shifts back to normal, unstable flow conditions are again set up in the same flow passage. This process repeats itself at a frequency dependent upon the thermal conductivity, specific heat, and mass of the heated rod. During this time the total flow rate remains nearly constant. The phenomenon is being studied further to determine its effect on reactor safety.

Visual Studies of Boiling on Hanford Type Fuel Elements. Studies were continued to define the heat transfer conditions when fuel elements are not situated in a coaxial position within a process tube. The test section in this study consisted of a 13.6-inch long electrically heated rod, 1.304 inches OD, placed within a 1.504 ID glass tube such that the distance between the rod and glass wall was 0.030 inch. For selected runs metal strips were fastened to the rod in a manner to simulate the support pieces used on "self-supported" fuel elements.

Tests were made at two flow rates with and without the metal strips attached. The conditions for these runs are listed in the following table:

Flow Rate gpm	Outlet Temp. °F below Boiling Temp.	Surface Conditions	Max. Heat Flux Attained B/hr-ft^2	Comments
20	121	No supports	700,000	No boiling
20	127	Supports present	552,000	Extensive boiling with film boiling in places
40	90	Supports present	703,000	Moderate boiling
40	163	Supports present	905,000	Moderate boiling with film boiling in places

High speed motion pictures were taken of each of these conditions.

It was concluded from these runs that the presence of the supports greatly influences the amount of boiling that occurs when the fuel is displaced toward the wall of the process tube.

Process Tube Flow Following a Failure of a Vanstone Flange. Laboratory tests and reactor experience have shown that the use of bumpered and self-supported fuel elements require larger charge-discharge forces than the old style fuel pieces used in ribbed process tubes. Repeated usage of large forces may weaken the process tube at the front Vanstone flange with possible subsequent failure during reactor operation. This would result in reduction of coolant flow past the fuel as water leaks through the thin annulus between the gun barrel and the process tube into the graphite stack.

An experiment was performed in the hydraulics laboratory to determine the pressure-flow relationship for the leak through a failed front nozzle Vanstone flange. Using the resultant data, calculations were performed to determine the change in flow past the fuel charge following such a failure. The results show that the most severe tube flow reduction would occur in fringe zone tubes operating near critical flow in the inlet orifices. Even with this condition the flow reduction for a typical BDF fringe tube would be less than 2-1/2 gpm. With an initial outlet temperature of 130 C, the leak would cause the outlet temperature to rise to 141 C. The reduction in flow past the fuel of a central zone tube would be less than 1 gpm. Thus, a front

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Vanstone failure would not reduce the coolant flow past the fuel enough to cause flow instability.

The results also show that a Panellit pressure trip could not be assured on failure of a front Vanstone. A Panellit pressure change of only about 20 psi would occur on a fringe tube and a change of only about 50 psi would occur on a central tube. The above results which were obtained for BDF type reactors would also apply to C and K Reactors and, in fact, the flow reduction past the fuel would be slightly less for C than K Reactors.

Flow Mixing Studies. Tests were continued in the hydraulics laboratory to determine the effectiveness of water mixers placed in process tubes of the present production reactors. The tests are conducted by introducing two water streams of different temperatures upstream of a water mixer and measuring the water temperature distribution around the annulus and in the hole of the I&E fuel downstream of the mixer. In the latest tests the inlet water temperature between the ribs was adjusted (by steam injection) to be about 7 to 10 F hotter than the remainder of the channel. This allowed the specific determination of the effectiveness of the mixers in reducing the high between-rib-temperatures such as are occurring at the K Reactors.

The data indicate that a simple 180 degree spiral does quite well in mixing the fluid streams. Although all of the temperatures downstream of the mixer agreed within ± 1 F, there does seem to be a slight preferential displacement of the between-rib fluid to the top of the annulus.

A split spiral combination (half left hand, half right hand) mixer with the spirals diverging from the rib area showed about equal mixing effectiveness as the simple spiral. When it was oriented to converge at the rib area, it was not as effective. Thus, the split spiral has the disadvantage that it must be oriented within the tube.

Tests are being re-run to make sure this preferential displacement is repeatable rather than being random scatter of the data.

Boiling Burnout Data. Considerable boiling burnout data have been obtained during the past year using a 12-foot long, 0.44-inch ID test section in first a horizontal position and then a vertical position. A comparison of the data for these two positions has been made.

The comparison at low coolant mass velocities (0.5×10^6 and 1.0×10^6 lb/hr-sq ft) showed generally higher burnout heat fluxes for the vertical case; up to a factor of two greater than for the horizontal case at comparable coolant enthalpies. Local steam qualities at burnout ranged from 25% to 60% at these low mass velocities. At mass velocities from 2×10^6 to 7×10^6 lb/hr-sq ft, with local coolant conditions ranging from about 20 F subcooled to steam qualities slightly above 30%, results for the horizontal and vertical cases showed excellent agreement. Further, tests with inlet temperatures near saturation and with mass velocities of 5.0×10^6 to 7.0×10^6 lb/hr-sq ft produced upstream burnout for both orientations, also with good agreement between burnout heat fluxes at comparable local enthalpies. Thus, at mass velocities of 2.0×10^6 lb/hr-sq ft and higher, there is no apparent orientation effect with the size test section used. At lower mass velocities, stratification may have affected the burnout heat flux for the horizontal case.

A cursory examination was also made on the effects of pressure since boiling burnout experiments using 0.44-inch ID tubular test sections have been conducted at 1500 psig, 1000 psig, and 100 psig. Results for these three pressures were compared to determine the applicability of 1500 psig data, which comprises by far the greater part, to other pressures. For mass velocities up to 2×10^6 lb/hr-sq ft, the range over which data at 100 psig were obtained, decreasing pressures produced higher burnout heat fluxes at comparable steam qualities. However, when the data were plotted as burnout heat flux versus local enthalpy, results for three pressures generally agreed quite well.

At mass velocities above 2×10^6 lb/hr-sq ft, and up to 7×10^6 lb/hr-sq ft, no data were available at 100 psig. A comparison of data taken at 1000 psig and 1500 psig in this flow range showed that at comparable enthalpies the burnout heat flux was lower at 1000 psig than at 1500 psig; the difference increasing with increasing mass velocity. At the upper end of this flow range, burnout heat fluxes at 1000 psig were approximately one half of those at 1500 psig at the same coolant enthalpy. However, in this region comparable burnout heat fluxes were obtained at comparable steam qualities at the two pressures, a fact which may be coincidental.

The data at the three pressures indicate that there is a mass velocity dependent effect of pressure on burnout heat flux. However, the data examined so far are insufficient to establish any definite relationship. Further investigations, both analytical and experimental, will be made to define and explain this behavior.

[redacted] [redacted]

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 Hydraulics Studies

Consequences of Failure of the PRTR Rupture Loop Pressure Control Valve. The PRTR rupture test facility has essentially a once-through coolant system. The flow is from a storage tank through a positive displacement pump, through the test section, through a pressure control valve which reduces the pressure to near atmospheric, and then either directly into the storage tank, or through ion exchange units into the storage tanks. The loop is operated at about 2000 psi and 600 F in the test section. Calculations were made to assess the possibility of rupturing the ion exchange unit or the storage tank if the pressure reducing valve were to fail to the full open position. The calculations are not complete, but extrapolations of the results obtained so far indicate little likelihood of rupturing the storage tank. The extrapolation does not give so clear an indication for the case where the flow is through the ion exchange units. In this case it appears quite possible that bursting pressures would be reached. Calculations were also made which show that if an orifice with a pressure drop of 200 to 300 psi at normal flow were placed upstream of the pressure control valve, neither the storage tank nor the ion exchange units would burst if the control valve failed open.

Hazards Analysis

PRTR Gas-Cooled Loop. The document HW-61236 SUP4, "PRTR Final Safeguards Analysis, Supplement 4, Gas-Cooled Lop Analysis," was published, Analyses revealed no serious adverse effect of the gas loop on the nuclear safety of the PRTR. The radiological consequences of a reactor accident are not changed by the presence of the loop.

PRTR Hazards Studies. An evaluation of four plutonium fuel element designs to be tested in the PRTR was completed. These elements include $MgO-PuO_2$, ZrO_2-PuO_2, stainless steel clad UO_2-PuO_2 and segregated plutonium-UO_2 19-rod clusters. Neutron flux distribution calculations in conjunction with heat transfer considerations indicate that these fuel elements will not exceed PRTR tube power and heat flux limits. Also, the reactivity worth of these elements was found to be under the maximum specified for standard spike fuel.

Additional calculations were completed to obtain kinetic constants for use in analog studies of PRTR excursions with a mixed crystal loading. A value of 0.5×10^{-3} seconds was obtained for the prompt neutron lifetime of this loading. This is a reduction by about 20 percent of the value for the spike loaded core. Since the fraction of fissions occurring in plutonium and uranium should not be much different than at present, the effective delayed neutron fraction for the reactor with a mixed crystal loading should be relatively unchanged.

Plutonium Fuels Development

UO_2-PuO_2 Fuel for PRTR. The first four UO_2-PuO_2 19-rod clusters made by swage compaction are under irradiation in the PRTR and have accumulated an exposure of about 9 MWD/T. Rods for an additional 30 clusters have been loaded during the month and, of these, rods for 18 elements have been processed through autoclaving. Eight to ten elements are scheduled for charging into PRTR in early May.

The results of the accelerated fabrication program put into effect for the month of April were even better than anticipated for the vibrational compaction project. Fabrication of fuel elements containing PuO_2 was started, and the process was debugged so that it produced from four to eight fuel rods per shift during the month. Both the quantity and quality of the resultant fuel elements were excellent, thus establishing that it is possible to make vibrationally compacted fuel elements in large numbers despite the additional problems associated with handling PuO_2. The procedures devised for controlling contamination and for insuring uniformity of the PuO_2 in the fuel elements worked very well and over 200 fuel rods were produced.

Some process control problems were encountered in the early fabrication phases of the four hundred UO_2-PuO_2 rods which were loaded and swaged during the month. Variations in UO_2 particle

size distribution caused a 200- to 250-gram variation in the amount of oxide loaded into the rods. Cleaner size fraction control was achieved by increased screening efficiency and a different blend was selected which repeatedly produced a powder load in the weight range desired for our particular swaging requirements. In addition, the new blend reached the desired tap density without tamping which reduced over-all loading time.

Some layering of the fine UO_2-PuO_2 and the coarse UO_2 has been observed in these rods due to differential settling rates of the size fractions involved, although longer range uniformity is good. Mixing funnels built on a maze or multi-splitting principle were used on several rods to improve increment mixing and to slow the drop of the larger particles. Autoradiography indicated improved core uniformity, but additional improvements are desirable for more severe fuel applications. The effect of the amplitude of 60-cycle vibrations on intermixing of adjacent increments in the tube during loading is also being studied.

Modifications were made on the swage which permit a greater degree of control over the positioning of the die throw wedges. A motor driven system which replaced a mechanical lock pin device greatly reduces the time required to change wedge settings and gives the operator more flexibility in his selection.

Special Fuel Elements for PRTR Irradiation. Zircaloy clad 19-rod clusters containing MgO-PuO_2 and ZrO_2-PuO_2 fuel material are being fabricated for irradiation testing in the PRTR. Experimental work on the MgO-PuO_2 element is further advanced than the ZrO_2-PuO_2 element, and it will be ready for irradiation first. The elements are being fabricated by swage compaction using swageable end caps. Fused pure MgO and fused stabilized ZrO_2 (5 w/o CaO added) are the inert diluents. As-calcined -325 mesh PuO_2 containing 11.61 percent Pu-240 will be blended with -325 mesh MgO. This mixture of fines and the coarse fraction material will be incrementally loaded into the tubes before swaging. Each element will contain 371 grams of PuO_2 which is a fuel composition of MgO - 2.1 w/o PuO_2 and ZrO_2-1.3 w/o PuO_2. Several experiments have been performed to determine the optimum particle size fractions, using available material, which produce the highest tap and swaged densities. Swaged densities in the range of 87 to 90 percent of theoretical can be easily attained and the final density is relatively insensitive to the size fraction composition of the MgO within certain limits. A swaged density of 88-89 percent is obtained with as-received MgO which has five percent fines added. This

type of material will be used for the $MgO-PuO_2$ element. The radial power distributions calculated for these elements are essentially the same as for an Al-Pu spike fuel element and enough high exposure plutonium enrichment has been added to match the initial reactivity of a standard low exposure Al-Pu spike element. Under these conditions, then, these elements should generate about eleven percent more power than a standard low exposure Al-Pu spike element operating in the same neutron flux.

A process is being developed for preparing a PRTR 19-rod cluster in which the "old" plutonium is segregated from the "new" plutonium formed in the UO_2. A typical fuel rod will be composed of a Zr-2 outer tube (0.565-inch OD, 0.030-inch wall), an Al - 7.5 w/o Pu - 2 w/o Ni alloy inner tube (0.495-inch OD, 0.030-inch wall), and a natural UO_2 core (80-86 percent of theoretical density). A completed fuel rod will contain 14 grams of plutonium and the UO_2 will be densified by use of the vibrational compaction technique. The tentative process for producing the Al-Pu-Ni alloy tube is as follows: cast extrusion billet, hot extrude tube, draw with mandrel, straighten, remove mandrel, draw without mandrel, straighten, cut to length, and degrease. Drawing is required because the extrusion press does not have a mandrel manipulator so that the extruded tube has a tapered wall thickness along its length. Tests have been conducted with aluminum billets and a tube has been extruded. UO_2 compaction experiments were performed with different loading and clamping techniques and with various UO_2 size fractions. As anticipated, the manner and degree of coupling between the inner and outer tubes affects the ultimate density of the UO_2 core at the compaction step.

Phoenix Experiment. The irradiation and reactivity measurements on the high exposure aluminum-plutonium samples are continuing in the MTR and ARMF. The current status of the irradiations is as follows: the sample which initially contained 6.25 w/o Pu-240 (GEH-21-1) has been irradiated for its fourth cycle and ARMF measurements have been made; the sample containing plutonium with 16.33 w/o Pu-240 (GEH-21-3) is now being irradiated for its fourth cycle; and the sample containing plutonium with initially 27.17 w/o Pu-240 (GEH-21-19) has completed two cycles of irradiation.

The irradiation samples will soon attain their originally predicted goal exposure of five MTR cycles. The ARMF data are being

evaluated to determine whether the originally estimated exposure is adequate or if the samples should be given additional irradiation.

Irradiation Testing. Two UO_2-PuO_2 capsules were returned to the MTR for additional irradiation. One of the high density oxide capsules (GEH-14-85) contains UO_2 - 2.57 a/o PuO_2 and is scheduled for insertion. The second contains UO_2 - 4.13 a/o PuO_2 and is tentatively scheduled for insertion during Cycle 174. Current average exposure for both is about 160×10^{18} fissions/cm^3 or 5500 MWD/T of UO_2-PuO_2.

Microhardness data were obtained on some irradiated UO_2-PuO_2 capsules. All specimens showed a density increase in the central region. The most marked changes occurred in the low density specimens. Post-irradiation measurements show near-theoretical densities in these pieces.

The irradiated Zircaloy-clad capsule (GEH-14-27) containing an Al - 2.1 w/o Pu - 2.0 w/o Ni alloy core and fabricated by injection casting is being examined. Diameter measurements on this specimen and on the earlier companion piece (GEH-14-28) reveal little change. No marked change in external appearance was observed.

Uranium Fuels Development

PRTR Fuel Element Performance. Periodic examinations of PRTR UO_2 fuel elements continue to show no mechanical distortion and no corrosion of the fuel elements. Maximum exposures reached thus far are: Mark I cold swaged UO_2, 2075 MWD/T_U; Mark I hot swaged UO_2, 561 MWD/T_U; Mark I vibrationally compacted UO_2, 642 MWD/T_U; Mark II-C vibrationally compacted UO_2 (nested tubular element), 883 MWD/T_U.

Physics Test Elements. Inspection and testing of components for six vibrationally compacted UO_2 Mark-I fuel elements were completed. The fuel elements have cobalt-zirconium wire flux monitor-spacing members and will be used for in-reactor instrument tests. Detailed gamma-ray attenuation measurements of 40 of the fuel rods were completed to permit correlations of fuel density with neutron flux profile.

Characterization of Fused UO_2. Particle size distribution measurements on -200 mesh (-74 micron) fused UO_2 indicate that 90 w/o of this mesh fraction is composed of 20 micron (-600 mesh)

particles. Formerly it was thought that this -200 mesh fused UO_2 consisted of a broad spectrum of particle sizes from -74 microns downward. Measurements by a new direct counting technique (Coulter Counter) and Fisher sub-sieve analyzer upon -200 mesh powders prepared by crushing or milling indicate that fused UO_2 is characteristically reduced to particle size range of 10-30 microns (-600 mesh). Particle size distribution measurements are planned upon other types of material, such as electrodeposited and high energy impact formed UO_2.

Thermocouple Test. The thermocoupled, 19-rod PRTR fuel element was disassembled for sectioning and correlation of the microstructure of the core material with temperatures measured during irradiation. The axial power generation will be determined by burnup analyses of samples taken at short intervals along the length of two fuel rods.

Fuel Evaluation. A nine-rod cluster fuel element, fabricated by swaging, which was successfully irradiated in the VBWR to an exposure of 3400 MWD/Tu at a peak surface heat flux of approximately 280,000 BTU/hr-ft^2, was returned to HAPO for examination. Four fuel rods in the cluster were broken off just below the end cap weld joint, subsequent to discharge of the cluster from the VBWR. Preliminary metallographic examination has revealed hydriding of the Zircaloy-2 cladding in the region of the break.

Cathodic Etching. A new electrode design was employed in the investigation of cathodic etching prior to completing fuel element end closures by magnetic force welding. This electrode provided the maneuverability required to etch both the rod cladding and the end cap. Remotizing studies are continuing.

Cladding Studies. Sections of inverted-cluster and single-rod fuel element cladding fabricated by high voltage electron beam welding were returned from the vendor. Evaluation of the welds and over-all fuel element design is in progress; preliminary data indicate that several changes are desirable for the end closures and the internal design of the inverted cluster element. High voltage electron beam welding of the single-rod test sections was considered successful; no changes in the design appear to be necessary.

Remote Fabrication Studies. Simultaneous compaction of fuel into all three components of a Mark II-C nested tubular fuel element is being investigated. To determine cladding interactions under

the most vigorous vibrational compaction conditions, a fully assembled Mark II-C (nested tubular) element was suspended from an overhead steel beam and the full power of a 5000-lb-thrust-output vibrator applied to the beam. The energy transmission efficiency of this system is not known, but accelerometer measurements made on the fuel element indicated that accelerations up to 95 g were achieved. The only cladding component to fail was a one-eighth-inch pin which held the center rod in place. The element will be completely disassembled and inspected for other cladding damage. In addition to the maximum power tests an endurance test was made during which the element was vibrated for ten minutes at frequencies varied continuously from 150 to 300 cps at an output power level producing an acceleration of 50 g.

X-rays of the fuel in the cladding components after testing showed an increase in compaction efficiency over that achieved in vibrating the components separately, particularly in the outer tube. This is significant in that no follower rod could be used during these tests. Follower rod pressure was previously deemed necessary to achieve the maximum density. If this situation can be duplicated and expanded, it will significantly simplify remote control compaction operations.

Materials Development

PRTR Aluminum Corrosion. Aluminum corrosion in a water vapor argon mixture was examined at 260 and 315 C. The atmosphere consisted of approximately 58% A, 41% H_2O, 0.6% H_2, and 0.3% O_2. In one experiment a minute leak contaminated the atmosphere with carbon tetrachloride and its decomposition products such as CO, CO_2, HCl, etc. The weight gains of the alloys at 315 C for 1600 hours were less than 0.8 mg/dm^2 in the following decreasing order: 6063-T6, 6061-T6, X-8001, and a special alloy 2.1% Fe, 0.6% Ni balance aluminum. The weight gains at 260 C for 3640 hours were less than 1.5 mg/dm^2 in the following decreasing order: 6063-T6, X-8001, 6061-T6, and the special alloy. The chloride contaminated test at 315 C for 2740 hours showed much higher weight gains for the 6061-T6, 6063-T6 with intergranular attack and blistering. The X-8001 and the special alloy were relatively unattacked by comparison and had only a slight increase in weight gain over the same alloys in the uncontaminated 315 C test at the same exposure.

PRTR Rupture Loop Deionization System. The makeup deionization system was regenerated twice and exhausted once during the past

month. Regeneration of the cation and anion exchangers was accomplished without difficulty in each case. Performance of these units during exhaustion met both exchange capacity and effluent purity specifications even though the system was processing Yakima River water (175 ppm total solids) rather than the Columbia River water (80-100 ppm total solids) that it was designed to process. As expected, the effluent purity was higher after the second regeneration (5-10 micromho/cm) than after the first (20-30 micromho/cm).

Initial regeneration of the mixed bed unit was hindered by problems encountered in maintaining constant flow rates in the acid and caustic regenerant streams. Performance studies of the second exhaustion cycle are now in progress.

Corrosion of Zircaloy-2 under Heat Transfer Conditions. Testing of Zr-2 cladding was initiated at 280,000 Btu/hr-ft^2 and 580 F in high purity water adjusted to a pH of 10.0 with LiOH. A thermocouple was inserted in the Zr-2 to monitor temperature increases as corrosion occurred. The small (0.025-inch) thermocouple suffered a mechanical failure and the test has been in progress without measuring the surface temperature directly. Heat transfer calculations estimate the Zr-2 surface temperature at 637 F; this temperature is just below the point at which nucleate boiling occurs. The test was designed to provide information on Zr-2 corrosion under conditions which might cause LiOH concentration on fuel elements. Examination of the specimen after 277 hours revealed no unusual effects; the surface of the Zr-2 was covered with a shiny, adherent black ZrO_2 film. After a total of 746 hours, the shiny black film was covered with a very loose thin film of a reddish deposit in the heat transfer region. No estimate of thickness is possible except that the crud layer was less than 0.1 mil thick. The sample was recharged for additional exposure.

Pre- and Post-Irradiation Evaluation of PRTR Pressure Tubes. A total of 18 replacement process tubes have passed all tests and are ready for use in-reactor.

Creep rupture tests are being performed on the annealed portion of the Zr-2 PRTR process tubes. The following table is a summary of the results to date:

Creep Rupture of Zr-2, PRTR Pressure Tubes at 550 F

Stress psi	Time hr.	Total Strain percent	Secondary Creep Rate
43,700	1(a)	60	--
40,900	772(a)	43	1.5×10^{-5}
39,600	2164	3.48	9×10^{-7}
36,200	2161	1.28	7×10^{-7}

(a) Tube ruptured.

Three pressure tubes have been removed from the PRTR for detailed tests and examination. Room temperature burst tests were performed on sections of these tubes with different exposure histories. The following table summarizes the results to date:

Room Temperature Burst Tests
on Irradiated Zr-2, PRTR Pressure Tubes

Tube Location	Exposure nvt - E>1 Mev	Burst Pressure	Ultimate Hoop Stress
Control	None	8535	97,700
1154	10^{15} (a)	7450	86,200
1756	10^{16} (a)	9450	106,200
1659	10^{17} (a)	8475	97,000
1154	5×10^{19} (b)	7600	86,700
1659	1.5×10^{20}	9700	111,000

(a) These samples were cut from the region of the lower fuel support contact marks in the tubes. A steep flux gradient exists in this region and the exposure is estimated for the point of initial failure only.

(b) Failed with a short crack oriented 45 degrees to the tube axis. Failures with similar appearance have occurred in unirradiated samples at defects in the tube wall.

From observations made to date, the effect of irradiation on the appearance of the fracture was nil. The deformation prior to fracture appears to be confined to a local bulging for irradiated tubes whereas for unirradiated tubes the diameter increases along the entire gage length of the tube prior to the formation of a

local bulge at the point of fracture. The sample from channel 1154 that failed at a small crack is of special note in that the crack having propagated through the wall of the tube was then arrested even though the tube was pressurized to about 90 percent of its ultimate strength.

Measurements of the depth of contact corrosion marks in the tube removed from channel 1659 were 1 mil and 5 mils where the lower fuel supports had contacted the tube and 5.5 mils where the wire wrap from an individual rod contacted the tube.

Mark III Monitoring Equipment. Some underwater viewing experiments using the TV-Omniscope system, were performed in the 314 Building PRTR process tube mockup. An appreciable amount of illumination will be needed. The heat from the illuminating lamps produced small bubbles which tended to collect on the glass surface of the objective lense. These bubbles can be swept from the objective lense if the water velocity past the objective lense is sufficiently high. More tests are needed to establish the minimum relative water velocity. Galvanic corrosion occurred in the lamp sockets and more corrosion resistant lamp socket material will eventually be needed for underwater use.

The TV camera-eyepiece section of the borescope has been returned to the vendor to increase the TV camera image size. A 37 percent increase in image size is required to produce a full-sized picture on the TV monitor.

A rubber ring to wipe off contaminated water upon probe retraction has been designed and fabricated. Testing to obtain the most effective configuration is continuing.

A system for coiling the TV and instrument cables has been designed and tested. This system was found to be necessary to protect the electrical cables and avoid interference with the necessary vertical traverse of the PRTR process tube inspection probe.

A granular quartz-filled silicone-resin molding was obtained for the gas gap body. Machining problems with this molding compound have been encountered by the fabricator of the gas gap instrument. Resolution of these difficulties is under way at present.

Brittle Fracture Studies. Brittle fracture studies were continued. By varying the weight of the projectile, its position in the gun barrel, and the strength of the powder charge, it is

possible to gauge the impact so that a crack will be formed on the inside surface of a given type of tubing. The proper combinations were determined experimentally for each type of tubing and vary with wall thickness, diameter, and degree of cold work of each material. On a given type of tubing a lighter impact is required to form a crack at elevated temperatures than at room temperature. Pressurization of the specimen does not appreciably alter the parameters required for crack-formation.

Using these newly developed parameters confirmation was obtained of a previously reported series of tests on 65 percent cold worked KER tubing. In the original tests, when the crack did not propagate, a hole had been sheared through the tube wall with no evidence that a crack had actually been formed. Using the new test parameters, a crack was definitely formed on the inside surface of tubes where no further propagation occurred.

A similar series was run on 50 percent cold worked, Zr-2, KER tubing at room temperature, 150 C, and 300 C. Pressures equal to 90 percent of ultimate burst pressure failed to propagate a crack at the two elevated temperatures although an initial crack sufficient to form a leak was formed by the impact of the projectile. At room temperature the crack did not propagate at 75 percent of the ultimate burst pressure but propagated the full length of the specimen at 80 percent of the ultimate burst pressure.

A test was run on a section of annealed KER tubing with a hydrided spot 1.2 inches long (8 times wall thickness). The tube was pressurized to 90 percent of ultimate at 150 C. The projectile formed an initial crack the length of the hydrided spot. This crack then propagated to 58 percent of the full length of the specimen before being arrested in the annealed material. A test under identical conditions except with a hydrided spot only two wall thicknesses in length produced identical length of crack.

2. PLUTONIUM UTILIZATION STUDIES

Plutonium Oxide

The following data were obtained on the lattice expansion of PuO_2 as a function of temperature.

Temp., C	a_o	$\Delta a/a_o \times 10^3$
23	5.3958 ± 0.0002	
280	5.4080 ± 0.0002	2.224
500	5.4186 ± 0.0005	4.225
604	5.4227 ± 0.0004	4.948
812	5.434 ± 0.001	7.134
902	5.439 ± 0.002	7.969

The lattice expansion data when plotted against dilatometer results are in quite good agreement. At higher temperatures thermal diffuse scattering results in broadened lines and it is apparent that higher temperature data will have to result from the lower angle spectra at the expense of accuracy.

Samples of UO_2-PuO_2 which were melted in helium were analyzed chemically to obtain additional data on plutonium losses. The following illustrates the losses found and are consistent with previous data.

Initial Composition	Percent PuO_2 After Melting
UO_2 - 25 w/o PuO_2	20.3
UO_2 - 40 w/o PuO_2	36.2
UO_2 - 75 w/o PuO_2	62.9
UO_2 - 90 w/o PuO_2	71.7
UO_2 - 95 w/o PuO_2	78.7

Initial experiments have been run to determine precise O/Pu ratios coulombmetrically via the Pu^{+3}/Pu^{+4} couple. First results are encouraging and additional runs are planned.

Plutonium Carbide

The x-ray diffraction patterns from the ten PuC alloys annealed at 610 C for 65 hours have been analyzed. Between 32 and 41 a/o C the lattice constants decreased in a uniform fashion with composition extending from 4.9718 to 4.9677A. Below 32 a/o C the cell size of the PuC phase remained constant at 4.9718A. Lines other than PuC on the films were very weak and appear to conform to a mixture of alpha and delta plutonium. Because of their low intensity, a large degree of certainty cannot be placed on this identification. Between 41 and 52 a/o C the lattice parameters were constant at 4.980A and again Pu_2C_3 was observed at 48.2 a/o C.

A change in lattice parameter of a 48.2 a/o C sample with time has been noted. The sample, which was arc melted and annealed between 400-550 C, gave a cell size of 4.9748A immediately after the anneal. After storage for about a month, this value increased to 4.9774A. No explanation is readily apparent. Other specimens are presently being re-examined.

Continuing work on the reaction of dry hydrogen with PuC has shown that the hyperstoichiometric PuC does not suffer the same attack as the hypostoichiometric exhibits. A sample of 53.0 a/o C has been exposed for 17 days with no sign of attack. As a first test of the effects of heat treatment on the resistance to attack, a 46.0 a/o sample which was made by resintering fused PuC is presently being exposed.

A drop casting hearth for the arc melter was designed and built. Using graphite molds of a "hot top" design it produces sound rods about 5/16-inch diameter by 1-1/4-inch long. These will be used for dilatometer and resistivity measurements.

Compatibility studies have shown that a BeO coating on graphite is effective in inhibiting the reaction between graphite and PuC for short times at 1550 C as long as the coating remains integral. Wherever the coating is broken, though, the carbide sticks badly. ZrO_2 shows some promise but is generally less satisfactory than the BeO. Several other coatings have been tried with various degrees of success. A slip was made with TaC in ethyl alcohol and graphite was coated with an apparently tight layer. Upon heating, though, the layer often spalls. Where the layer remains unbroken, good protection is afforded. Thin coats which are subsequently reacted with the graphite substrate are presently being tried. Another approach under study is the evaporation of Ta metal onto graphite and conversion to TaC by high temperature reaction.

Plutonium Nitride

X-ray diffraction analyses of the arc melted plutonium nitride generally involve the Debye-Sherrer powder method. Powder is obtained by filing the arc melted button in a nitrogen atmosphere and screening out the -325 mesh fraction. Evidence now indicates that this fraction is not representative of the material as a whole. X-ray diffractometer charts often show strong alpha plutonium lines from the polished button, whereas -325 mesh powder filed from the same button gives only a trace or no indication of free plutonium on a Debye-Sherrer film. It is thought

that during filing the brittle phase, plutonium nitride, is preferentially comminuted. Screening then leaves the more ductile plutonium in the coarser plus 325 mesh fraction. Melting experiments using the arc melted material will be deferred until a specimen free of residual plutonium is obtained.

Photomicrographs were taken of an arc melted plutonium nitride button containing residual plutonium. Chromic oxide and tetarphosphoric acid were both found to be suitable etchants, with chromic oxide perhaps the better of the two. The plutonium nitride is in the form of dendrites in a plutonium matrix. It appears that approximately 80 percent of the plutonium combined with the nitrogen to form plutonium nitride. Chemical analyses of this same specimen for plutonium support the 80 percent conversion estimate. The Nessler ammonium procedure was used to analyze quantitatively for nitrogen; however, it gave low values.

Plutonium hydride can be converted to the mononitride by heating in anhydrous ammonia, though first attempts yielded a two-phase product, plutonium dioxide plus plutonium nitride. Further experiments will be performed in an attempt to obtain a single-phase plutonium nitride by this means.

3. UO_2 FUELS RESEARCH

Electron Microscopy

Examination of an irradiated (5000 MWD/T) UO_2 crystal continued, using reflection electron microscopy techniques. The crystal was surface-heated using an arc-image furnace to about 500 C above ambient temperature; in a second experiment the specimen was bulk-heated using the standard wire-wound microscope furnace to 980 C. Both fracture and polished surfaces were examined during the heating and cooling cycles of about 40 minutes each. Only minor changes in the surfaces were observed, similar to those reported previously on non-irradiated crystals. It appears likely that higher temperatures, longer irradiations, and/or higher magnification may be required to observe irradiation-induced surface effects.

Greatly increased electron beam densities have been achieved from an electron gun being developed for a specimen heating attachment for reflection electron microscopy by coating a carburized filament with LaB_6.

Etching Procedure for UO_2 Single Crystals

Difficulty in interpreting ceramographic results of UO_2 single crystals is complicated by the variations in microstructure resulting from different etching techniques. A standard etching procedure should be established to compare ceramographic data from different specimens. Preliminary studies suggest a one minute etch with 10 percent H_2SO_4 -90 percent H_2O_2 at room temperature. The etching characteristics are affected by the crystal orientation.

UO_2 Thermal Conductivity

The thermal conductivity program in cooperation with BMI has been resumed. Measurements on the non-irradiated UO_2 single crystal were extended to 1200 C, and calculation of results is in progress. The crystal was returned to HAPO for low level irradiation prior to further measurements. The crystal has survived the physical handling and thermal cycling necessary for the measurements, with only minor physical damage apparent (surface discoloration, edge chips, etc.)

Refurbishing equipment at BMI for measurements on the final sintered UO_2 rod in the original irradiated series is in progress. A pulsed-energy-input method of determining thermal diffusivity being evaluated at BMI appears promising for small ceramic samples. BMI personnel plan some exploratory measurements on UO_2 at elevated temperatures in connection with the HLO-BMI cooperative investigation of UO_2 thermal conductivity.

The last in a series of six irradiation capsules, containing cylindrical specimens of UO_2 for thermal and electrical conductivity and ceramographic studies, was discharged after four years of irradiation. The specimens have an exposure twice that of previous specimens.

Precompaction and Hot Isostatic Pressing of UO_2

Swaging followed by hot isostatic pressing offers a promising method for compacting inexpensive fuel materials to uniform, high density fuel elements. Mixtures of fused UO_2 and micronized UO_2 were swaged in 0.007-inch thick stainless steel cladding, and then hot isostatically pressed to densities greater than 95 percent TD.

4. BASIC SWELLING PROGRAM

Irradiation Program

The irradiation of four general swelling capsules is continuing. These capsules are being irradiated in tandem, two to each test hole. The control temperature is 625 C on two of the capsules and 575 C on a third. These temperatures are held constant by means of auxiliary electrical heat regardless of reactor operating conditions. The fourth capsule operates at slightly over 400 C only during reactor operation. An additional capsule is complete and another capsule is approximately sixty percent complete. These two capsules will also be irradiated in tandem. Two capsules previously irradiated are in Radiometallurgy for disassembly and specimen recovery. Redesign and development of components for the metallographic swelling capsule is continuing. The inability to weld onto the sheath of the 1/16-inch sheathed heaters to be used for this capsule has presented some sealing problems. It has been necessary to resort to high temperature silver brazing in order to obtain satisfactory seals. The specimen holder has been designed and is being fabricated. Assembly of a prototype capsule is proceeding.

Post-Irradiation Examination

As indicated in previous reports, uranium irradiated to two burnup levels, 0.29 a/o and 0.41 a/o, respectively, at approximately 350 C under conditions of restraint swells considerably during vacuum annealing at temperatures above 600 C. The higher burnup specimens, 0.41 a/o, undergo volume increases which are far greater than those of the lower burnup specimens, 0.29 a/o. The respective volume increases associated with the annealing at 600 C for one hour are: 16.3 percent and 5.0 percent. The annealing treatment caused extensive change to the as-irradiated pore size distributions. In contrast, specimen 4-4 which was irradiated at 575 C to a burnup of 0.28 a/o under conditions of no restraint showed no change in pore size distribution after a 100-hour post-irradiation anneal at 575 C. The pore volume fraction and pore void density exhibited no change. Since this particular specimen (4-4) had an extremely large number of very small pores, micrographs at 35,000x were used in addition to the standard micrographs at 15,000x. A well defined peak in the pore volume fraction distribution curve was revealed in the data obtained from the 35,000x micrographs, but the resolution was not sufficient at lower magnification to observe this peak. Since no change occurred, it is concluded that the pore configuration

attained in unrestrained uranium specimens irradiated in the high alpha range (approximately 600 C) is thermally stable at temperatures at or below the irradiation temperature. Similar comparisons are currently being made on unrestrained specimens (irradiated to particular burnups at prescribed temperatures) before and after annealing.

Two capsules, GEH-14-334 and 336, each containing six coaxial U-U diffusion couples, have been opened after irradiation and the specimens have been reclaimed. Sampling techniques, which have been developed, will permit eventual determination of the U-235 burnup in the enriched layer and the depleted core of all specimens; the exposure gradient along the length of the capsules will be established.

The specimens generally appear to be in excellent condition. The enriched U-235 shell, bonded metallurgically to the Zircaloy-2 clad, appears dark under macroscopic viewing. This coloration is due to the bumping and growth of the enriched shell during the low temperature ($<$300 C) irradiations. The enriched shells in specimens from capsule 336, which received twice the exposure of specimens in capsule 334, have grown the most. The amount of growth observed, however, is considerably less than that observed in previously described copper clad (non-bonded) specimens, GEH-14-281 and 282. Presumably, the Zircaloy-2 clad which is bonded to the enriched U-235 shell restrains the growth of this shell. A sample from each of the two capsules has been processed for metallography. The Zircaloy-2 U-235 interface has remained intact. No porosity has been detected in the enriched layer. An additional sample from each of the two capsules is being annealed at 650 C for 100 hours to precipitate gas pores in the enriched layer. If diffusion of gas atoms into the core depleted in U-235 does not occur to a measurable extent, similar specimens will be annealed at higher temperatures. Gas atom diffusion into the Zircaloy-2 clad will also be investigated.

5. IRRADIATION DAMAGE TO REACTOR METALS

Alloy Selection

Efforts to procure sufficient quantities of the alloys selected for study under this program have continued. Additional contacts, both written and personal, have been made to procure heat analysis data and fabrication histories of past heats of alloys. Samples from five different heats of some alloys have been requested in plate form. From this material differences in mechanical

properties will be determined, as well as differences in chemical composition. Recent data indicate marked differences exist between heats in high temperature tensile and stress to rupture strength in Hastelloy X-280 alloys. Reasons for these differences have not been determined.

In-Reactor Measurement of Mechanical Properties

Two creep capsules operating in the reactor were used to check temperature and pressure control instrumentation during the month. No creep data were obtained. Some instruments required recertification for accuracy since they had been in continual use on creep tests since installation.

A new temperature control system has been developed and is now being checked in the laboratory to provide automatic control during reactor outages. The present temperature control system for the creep capsules must be adjusted manually at every reactor shutdown and startup to maintain a constant temperature. It is desirable to automatically control the temperature as an unscheduled shutdown would give a twenty to thirty degree excursion if the manual adjustments were not made. A new system, overcoming the requirement of manual adjustments, has been developed. The new control system requires only one controller with additional amplifiers to control both the heating and cooling in a capsule. The new system uses one controller to control one zone of the furnace from one thermocouple. All other thermocouples are then compared to the thermocouple on control to obtain a signal if a temperature difference exists. This output difference, or signal, from the control thermocouple and the compared thermocouple is amplified and fed to the furnace zone of the compared thermocouple with a phase shift to reduce the signal from the two thermocouples to zero. In this way, all temperatures are maintained the same as the one thermocouple on control. Laboratory tests with one standard controller and solid state, high power, amplifiers have shown that control of temperature and gradients to within plus or minus one degree is possible. The control system also operates the flow valves to cool the capsule when cooling is required. Again, this control of cooling is also automatic and does not require manual adjustment during reactor outages.

A standby data printing system has been received and checked out. The standby equipment was ordered when the data printer, on two occasions, burned out the printing solenoids during operation. No cause for burning the solenoids could be found. However,

during checkout of the standby printer and matrix processor, a possible cause for the original difficulty may have been found. The checkout revealed improper serial operation of relays in the circuits. The poor operation of the serializing relays was caused from shipping vibration. When the relays were properly adjusted, no further difficulty was experienced and the system operated perfectly for six weeks. The standby system was then installed in the data logging system at the reactor and the old system at the reactor was brought to the laboratory. The malfunctioning printer is now being checked for proper serial operation of the relays and if performance is satisfactory a digital voltmeter will be added to the system. The voltmeter will provide the capability of recording additional variables with the strain and time. The recording of additional variables, including stress, temperature, electrical power and coolant flow, will aid in interpreting the data, analyzing the creep curve and evaluating the performance of the capsule.

Two headers from the lot of capsules were identified as having Inconel pins instead of chromel and alumel pins. The incorrect pins were found during checking of the capsule components as modifications for internal water cooling were being made. The headers are used in the capsule to separate the high pressure section from the low pressure shroud tubes. The insulated electrical connections, water tubes, pressure tubes, reference tubes and rotating shafts all go through this header. The proper alloy should be used in the pins for thermocouple connections to eliminate possible errors if a temperature gradient should exist on a dissimilar metal combination. The headers were reworked and proper pins installed in all electrical and thermocouple locations.

A new statistical model for creep is being developed which may explain the low values for the activation energy for creep measured on the in-reactor specimen at 30,000 psi stress and 300 C. The model is based on a statistical random walk of vacancies to a dislocation line, resulting in a statistical dislocation climb. This model may be applicable to other metals and explain the anomalies reported in their activation energy spectrums not explained by thermodynamics.

Irradiation Effects on Structural Materials

The purpose of this program is to investigate the combined effects of radiation and reactor environment on the mechanical

properties of structural materials. Special attention will be given to the determination of mechanical property changes produced in metals by irradiation at elevated temperatures in contact with water.

The remainder of the 88 specimen quadrants scheduled for irradiation in the ETR, G-7 hot water loop were assembled and shipped to the ETR. These quadrants will keep the loop full through cycle 54 at which time specimens prepared from standard material lots procured on the program will be inserted. Assembly of control quadrants for the ex-reactor loop was completed and Zircaloy-2 specimens corresponding to ETR cycles 39 and 40 were discharged during the month. The oxide film on these specimens was uniform, and there were no signs of pitting.

Upon completion of ETR cycle 44, 36 tensile and 48 bend test specimens of Zircaloy-2 were discharged. These specimens received estimated exposures ranging from 5×10^{19} to 10^{21} nvt (>1 Mev), and were replaced with 72 Zircaloy-2 notch-tensile specimens. In order to facilitate calculation of notch-tensile parameters, a Fortran program for the 7090 computer was written.

Twenty-nine Zircaloy-2 tensile specimens representing the 0, 20, and 40 percent levels of cold work and irradiated at 280 C to approximately 3×10^{19} nvt (>1 Mev) were tested at room temperature. Crosshead speeds were varied from 0.005 to 2 in/min. The data obtained were compared with data for similar specimens irradiated at ETR bulk water temperatures (about 80 C) to 0.84×10^{19} nvt (>1 Mev). Comparing tensile data for the two irradiation temperatures revealed only minor differences in either strength or ductility. If the factor of 3.5 difference in total exposure between the two sets of irradiations is found to significantly affect properties, then some recovery due to 280 C operation would be indicated. The strain-rate sensitivity and directional effects for the specimens irradiated at 280 C were similar to those for unirradiated specimens corresponding to equal levels of cold work.

Damage Mechanisms in Iron

The purpose of this program of study is to establish the nature of interactions between defects present in iron prior to irradiation and defects produced by irradiation. Controlled pre-irradiation defects will consist of interstitial carbon and nitrogen, as well as substitutional titanium in various proportions.

Iron of a purity of 99.99 percent and 99.95 percent has been ordered. Impurity defects will be introduced in this metal. Required laboratory equipment is being assembled and tested. Foil specimens of high purity iron, 0.003-inch thick and 1/2-inch diameter (Johnson and Mathey stock) are being heat treated for transmission electron microscope studies. After characterization, these foils will be irradiated and changes will be ascertained.

A problem in neutron damage studies is the accurate designation of integrated fast flux (energies >1 Mev). Ti and Ni wires are commonly used to determine fast flux. In order to calculate integrated neutron flux from either of the above monitors, the shape of the neutron spectrum must be assumed. An attempt is being made to evaluate the possibility of measuring fast integrated neutron flux by utilizing thin evaporated films containing atoms which undergo fast fission. Evaporated films of UO_2, in which the uranium U-235 content had been depleted by a factor of twenty, were prepared with thicknesses in the range of 24 to 132 A (calculated). These films and wire monitors, contained in a cadmium capsule, are currently being irradiated. After irradiation the films will be examined by transmission electron microscopy and fission fragment tracks will be counted and measured. The data will then be analyzed to ascertain whether this technique can supply information on integrated flux for particular neutron energy ranges.

6. GAS-GRAPHITE STUDIES

EGCR Graphite Irradiations

The fourth capsule in the series of long term EGCR graphite irradiations, H-3-4, has completed the second cycle of irradiation in the General Electric Test Reactor. The maximum exposure received by samples which have been irradiated in all four H-3 capsules is estimated to be 9.5×10^{21} (E >0.18 Mev), which is equal to approximately 65,000 MWD/AT in the EGCR.

During the second irradiation cycle on the H-3-4 capsule difficulties were encountered in keeping the maximum sample temperatures below 850 C. The samples being exposed at the maximum temperature had been previously exposed to 8.7×10^{21} nvt (E >0.18 Mev), and it was imperative that the samples follow the established temperature pattern. The desired temperature pattern was successfully attained through the combination of three

actions: the capsule was removed and inspected; the reactor core loading was rearranged to shift the flux; and the reactor was run at less than full power for several days. The cause of the temperature peaking is a matter of speculation; it may have been due to a slight shifting of the samples inside the capsule during the previous shutdown. Such a shift could have changed the heat transfer characteristics.

Dimensional Stability of Fuel-Support-Sleeve Graphite. A graphite which has a somewhat lower degree of preferred orientation than conventional nuclear graphites will be used for the EGCR fuel element support sleeves. Several variations of this graphite, produced by Speer Carbon Company and generally known as Type 901, have been irradiated at about 600 C to determine dimensional stability. Samples extruded through a 3 by 8-inch die and given single and double impregnations were compared with a sample from the same mix extruded through a 4-inch round die and given a single impregnation. Samples of the same type but produced from a new source of raw materials and extruded with 3 x 8-inch die were included along with standard samples of CSF graphites.

After an exposure of 3120 MWD/AT at approximately 600 C, the following relative behavior was observed:

1. The effect of the re-impregnation did not significantly alter the magnitude of average contraction in either the transverse or parallel directions. In the singly impregnated sample, however, a variation in transverse samples was noted. Samples parallel to the 3-inch dimension of the die contracted 1/3 less than those from the 8-inch direction. This difference was apparently masked in the re-impregnated sample.

2. The radiation-induced contraction of samples from the 4-inch round extrusion was nearly isotropic, whereas graphites extruded with the rectangular die behaved much like conventional nuclear graphites.

3. The new raw material reduced contraction in both the transverse and parallel directions.

Altering the pore structure by impregnation has apparently changed the anisotropy of the radiation-induced dimensional changes. This explanation implies that the pores were elongated

and oriented in a preferred direction. Irradiation of these samples will be continued to provide information on the rate of contraction.

Molten Metal Impregnation of Graphite

Filling the graphite pores with metal is useful in the metallographic examination of the structure of various carbons and graphites. It also provides a means of measuring the density profile of graphite and thereby of examining the depth of oxidation of moderator bars under various oxidizing conditions.

Methods have been developed to impregnate the pores of graphite with such molten metals as Zn, Sn, Bi, Pb and Ag. Graphite of dimensions 1/2 by 1/2 by 1 inch have been successfully impregnated. The equipment consists of a high pressure autoclave, an electrical resistance furnace, a vacuum pump, a cylinder of helium and accessory tubing and valves.

After metallographic examination of the impregnated graphite samples, it was concluded that bismuth would be the best choice for an impregnant. Bismuth enters and fills the pores as small as 1μ diameter at the center of the sample; however, in some instances pores of larger diameter are not completely filled. These may be connected by very small diameter pipes and require much longer outgassing and filling time than was allowed.

Irradiation of Graphite Containing B_4C

Three irradiations have been concluded on graphite containing B_4C, one in the Snout II facility at approximately 30 C to an exposure of 7.5 MWD/AT and two in a Hanford hot test hole at approximately 575 C to exposures of 150 and 300 MWD/AT. No large difference in dimensional changes between nuclear grade graphite and graphite containing up to 7 wt % boron was observed in these irradiations. However, some deleterious effects of pitting and corrosion of the B_4C samples were observed which casts some doubt as to the stability of B_4C in a slightly oxidizing atmosphere. It is now known that the reaction, $B_4C + CO_2$ yielding B_2O_3 occurs at a measurable rate under the conditions of these tests. The reaction is also enhanced with the presence of water vapor, although the presence of water vapor is not suspected in these irradiations. A material which had cemented several samples together and formed a glazed coating on the flux monitor holders was confirmed as B_2O_3 by analysis.

A new irradiation technique is being explored in which it is planned to irradiate the samples in a vacuum. Preliminary estimates of heating by the reaction B^{10} (n,α) Li^7 are being made, and the effects of sample configuration on self-shielding are being studied to facilitate experiment design.

EGCR Graphite Oxidation Study

A study of the oxidation characteristics of the EGCR moderator has been conducted at Hanford to provide a basis to estimate potential combustion hazards in the event of a depressurization accident. Results of this study were recently discussed with representatives of the AEC, ORNL and Allis Chalmers Company to enable machine calculations to be made on the hazards analysis. Of particular interest were several factors to be applied to the basic oxidation rate for use in combustion calculations. Each is summarized briefly.

1. Gamma radiation: Oxidation experiments have been conducted in the temperature range of interest (>500 C) in the absence of radiation, and in the presence of two levels of gamma flux intensity, 2.5×10^5 and 1×10^6 r/hr. A decrease in the activation energy was observed from 50 kcal/mole for radiation intensities of 0 and 2.5×10^5 r/hr to 30 kcal/mole for a radiation intensity of 1×10^6 r/hr. Recently the gamma flux following an EGCR most credible accident was estimated to be 1×10^7 r/hr. In order to establish the oxidation rate at this higher intensity an oxidation study will be conducted in the core of the PRTR on a high priority basis.

2. Depth of oxidation: An analytical expression was derived and verified experimentally for determining the depth of oxidation in EGCR graphite after a given amount of oxidation. This expression will enable calculation of the heat generating mass which is essential in a combustion analysis.

3. Prior neutron damage: Oxidation experiments have been conducted on samples which have been irradiated to a neutron dose approximately equivalent to 50 percent of the expected EGCR moderator life. A rate enhancement of less than a factor of two has been observed.

4. Oxidation inhibitor: On the basis of the maximum credible accident now postulated for the EGCR, a reliable oxidation inhibitor cannot yet be recommended that would effectively

extinguish a fire or prevent one from starting. It was concluded that a research program in this field is vital to the Gas Cooled Reactor Program and the AEC has asked that such a study be conducted at Hanford.

5. Surface impurities: Surface impurities may influence oxidation; however, the type and amount of these impurities which would deposit in a combustion zone is difficult to estimate. The oxidation rate of discharged fuel sleeves will be measured to detect the presence of any catalytic effects of impurities in the EGCR coolant.

6. Gas purification of fuel sleeves: Based on the oxidation rate of gas-purified graphites, it is estimated that the oxidation rate of fuel sleeves could be reduced by a factor of 1/2 by gas purification.

7. Moisture: No catalytic effect of moisture on the oxidation rate of graphite in air has been detected in Hanford experiments.

8. Interactions: The most likely interaction of the above factors is one involving gamma radiation and water vapor, which could result in complex gas-gas reactions.

7. GRAPHITE RADIATION DAMAGE STUDIES

Effect of Impregnation on High Temperature Contraction

The first irradiation has been completed on a series of samples which are intended to determine the effect of impregnation with furfuryl alcohol on high-temperature radiation-induced contraction. Data from the irradiation in the range 600 to 1000 C in hot capsules in the ETR are found in the table. Each capsule contained four samples: unimpregnated; impregnated and baked 50 900 C; impregnated and graphitized at 2650; impregnated and graphitized at 2900 C. Since each capsule was subjected to a slightly different neutron flux and, due to variations in gamma flux, operated at slightly different temperatures, comparisons should be made only among samples in a single capsule. However, the results clearly indicate that impregnation decreases the dimensional stability of both TSX and HLM-85, an HTGR candidate graphite, under high temperature irradiation. Samples have been returned to the ETR for additional exposure.

Contraction of Impregnated Graphites

Graphite	Estimated Integrated flux, nvt (E>0.18 Mev)	Length Change, % Non-impregnated	Impregnated 900 C	Impregnated 2650 C	Impregnated 2900 C
TSX	9.9×10^{20}	-0.027	-0.149	-0.069	-0.065
TSX	10.6	-0.165	-0.205	-0.222	-0.178
TSX	6.2	-0.029_5	-0.070	-0.037	-0.032_5
HLM	7.0	-0.080	-0.094	-0.100	-0.110
HLM	6.7	-0.069_5	-0.083_5	-0.081_5	-0.076

Flux Intensity Test

The graphite irradiation capsule, GKH-13-8, designed to study the effect of flux intensity on property changes at a controlled sample temperature, is operating satisfactorily in the second cycle in the ETR. The four positions having heaters are being controlled at 650 C. The heater in Position 1 began operating erratically near the end of the last cycle and is now functioning intermittently. Thus far the sample temperature has been maintained in the range 625 and 650 C. Three of the sample positions are heated only by gamma heating, and the thermocouples in these positions continue to function properly.

8. ALUMINUM CORROSION AND ALLOY DEVELOPMENT

In-Reactor Testing of Aluminum Alloys

The H-1 Loop was operated 10 days at 200 C as a part of the startup procedure. Upon completing this run, the loop was charged with six coupon holders in high, low, and zero flux, containing coupons of X-8001 and A288 aluminum, nickel-plated aluminum, Zircaloy-2, carbon steel, and stainless steel. A few of the Zircaloy-2 and stainless steel coupons contained silver foils under the surface to determine whether crud deposition is increased by an internal beta source. The test is being run with deionized water (pH 6-7). The desired temperature is 300 C; however, it may not be possible to attain a temperature this high.

Design has been started to modify H-1 to permit charging of fuel elements.

The operating procedure for C-1 Loop is about 75 percent complete. The production test to install an improved in-reactor test section is routing for approvals. The coupon holders, designed for 1-inch x 5/8-inch coupon, are being fabricated.

Dynamic Corrosion Behavior of a New Aluminum Alloy

Dynamic corrosion tests on samples of the KY aluminum alloy (1.8 Fe, 1.2 Ni, High Purity Aluminum Base) have completed three months (2130 hours) at 330 C, 2500 psi, with 9.4 gal/hr refreshment and 25 fps linear flow rate. Preliminary data at 10, 30 and 90 days indicate a linear corrosion rate for X-8001, while the rate for the KY alloy appears to diminish. Coupons placed in the low flow section of the autoclave show penetrations of 1.8 mils for X-8001 compared to 0.57 mil for the new alloy at 1900 hours. These tests confirm improved high temperature (>320 C) corrosion behavior of the KY alloy relative to X-8001 under low flow conditions previously reported by Bowen (HW-68253). Preliminary data also indicate improved behavior of the KY alloy under dynamic conditions.

9. AEC-AECL PROGRAM

Twenty transverse notches were spark machined on the outside surface of a 0.680-inch ID by 0.035-inch wall Zircaloy tube. These notches were all 30 mils long and varied in depth from 1.4 to 17.5 mils in approximately one mil increments. With the crystal aligned for the transverse test the ultrasonic response of each of these notches was measured. A ten megacycle 1/2-inch by 1/4-inch flat rectangular lithium sulfate crystal was used at a 21 degree entry angle. Similar results were obtained at a 17 degree entry angle.

Analysis of the resulting curve of ultrasonic response versus notch depth reveals that the response vs depth is nonlinear, the initial slope of the curve rises sharply to depths up to 4 mils, then decreases to a much smaller slope for grooves up to 17.5 mils deep.

10. REACTOR STUDIES PROGRAM

Advanced Reactor Concept Studies

Several design parameters were selected for the initial reference design study of a plutonium-fueled power reactor for spacecraft applications. The reactor to be studied will be a 30 Mw

thermal (~5 Mwe) fast reactor. Fuel will be plutonium oxide with tantalum cladding. The coolant will be boiling rubidium, operating at 1000 C outlet temperature and 150 psia. Heat will be extracted by a turbogenerator and rejected at a radiation temperature of 670 C.

Choice of fuel for this study was arbitrarily limited to plutonium oxide or carbide because of lack of data on other plutonium compounds which are potentially of interest. The oxide was selected because of its higher melting point and compatibility with the tantalum jacket material.

A preliminary comparison of metal coolants indicated that rubidium would be preferable to potassium or sodium at this operating temperature primarily because of its lower specific volume, particularly in the radiator. Nevertheless, the specific volume is high enough to present significant problems in reactor plant and piping arrangement and core design. The accompanying higher operating pressure may also present structural design problems, but these have not yet been studied.

Study of thermal hydraulic considerations in the use of boiling metals was initiated during the month to develop a basis for estimating potential rubidium vapor quality at the reactor exit. Moisture separation schemes are also being studied.

A parametric study of rubidium cooled, plutonium fueled fast reactor cores was initiated in cooperation with the Applied Physics Operation. Additional neutron cross section data for rubidium, beryllium, U-234, U-236 and nitrogen were generated from the GAM-1 data tape to supplement the 16 group set being used for this study.

D. RADIATION EFFECTS ON METALS - 5000 PROGRAM

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

Molybdenum foils of two carbon levels, 15 ppm and 450 ppm carbon, were quenched from 2000 C, thinned, and examined by electron transmission microscopy. Numerous small dislocation loops, 50 to 500 angstroms in diameter, were observed. Zones free of loops

surrounded carbide particles; similar denuded zones were also observed at grain boundaries. The carbides and boundaries are excellent sinks for vacancies and account for the absence of loops in their immediate vicinity.

A capsule containing four molybdenum single crystal bend test specimens has been discharged from the reactor at an exposure of 10^{18} nvt. Flux monitors are being analyzed to insure that the goal exposure was reached.

An investigation of field ion microscopy as a technique for radiation damage studies is in progress. Molybdenum single crystal emitters containing carbon as an impurity are being prepared for this study. Preliminary evaluation of the technique will be made with unirradiated molybdenum before irradiating pre-shaped emitters.

The crystallographic orientations of the axis of molybdenum single-crystal tensile specimens, 112 in all, have been determined. The range of orientations is not as great as was desired, most rod axes lying on or near the plane containing [100] and [110] (the 001 zone). At least three of the four predicted modes of deformation can be investigated, however.

Twenty-four single-crystal specimens have been prepared with 110 faces for x-ray measurements. Preliminary results on four low-carbon crystals indicate a lattice parameter of 3.1472 A ± 0.0001 A. Measurements on high-carbon and medium-carbon crystals are incomplete.

A fixture has been designed which permits back-reflection Laue x-ray photographs of strained single-crystal tensile specimens. This fixture will maintain a load on the specimen during x-ray examination of any given deformation stage.

An irradiation proposal is being prepared to cover 18 capsules which contain the following specimens: six capsules containing 16 single crystal tensile specimens and six x-ray specimens per capsule; six capsules containing four polycrystalline tensile specimens per capsule; three capsules containing two polycrystalline stored energy and two length change specimens per capsule; and three capsules containing nine annealed polycrystalline resistivity specimens per capsule. Specimens in these 18 capsules will be exposed to a high fast flux for three exposure levels.

Five capsules containing a total of 27 annealed polycrystalline resistivity specimens are being irradiated to three exposures in a moderate fast flux environment. Cold worked polycrystalline

resistivity specimens are being irradiated in the same environment so that a comparison of the two may be made.

E. CUSTOMER WORK

1. RADIOMETALLURGY EXAMINATIONS

The radiography source of Finished Products Operation, CPD, was removed from the shielding and replaced with an 80 curie Co-60 source. (RM 349) Two x-ray diffraction samples from the inside and outside surfaces of a Purex waste tank were run; both x-ray diffraction patterns exhibited the same pattern. It was thought that any damage to the tank would show up on the sample from the inside of the tank. (RM 366)

The relative preferred orientation of an unfailed irradiated dingot fuel element was obtained by x-ray. These data were compared with the previous orientation data obtained on failed elements and the comparison showed the failed fuel elements have a greater degree of preferred orientation than the non-failed elements. (RM 437) Examination of two I&E enriched production elements with ruptures in the spire revealed the ruptures were caused by severe groove pitting that penetrated the internal aluminum cladding. (RM 444)

Visual and metallographic examination of a process tube failure from 3070-F revealed that the tube had failed by a fatigue mechanism.

2. EQUIPMENT PROJECTS

CGH-857, Physical and Mechanical Properties Testing Cell. The minor modifications, necessitated by manipulator interference, to the High Level Utility Cell have been forwarded to the vendor and the scheduled delivery date for the cell castings remains June 15, 1962.

CGH-858, High Level Utility Cell. Installation of the milling machine was completed. The decladder has arrived on the site and is being installed in the cell. Cell facilities are about 90 percent complete and construction forces are scheduled to complete their work within two weeks. Pending any unforeseen difficulties work on the project should be completed before June 1, 1962.

Micro Sampling Equipment. The micro drilling plug containing a 10X viewing scope and precision drilling spindle was completed by the Technical Shop. Fabrication of a sample collecting equipment is in progress.

Measuring Equipment. The remote micrometer with 0 to 2-inch capacity was received and placed in operation. The micrometer features a stage with vertical motion to center round specimens at their diameters. The micrometer head is located on the exterior of the cell and accuracy has proven to be within 1 mil.

Burst Test Equipment. The newly designed equipment consists of a steel mandrel with a Neoprene sleeve vulcanized to the metal mandrel, sealing being effected by the same principle as a tubeless tire. Tests have been completed successfully on six specimens of Zr-2 cladding approximately 5/8-inch in diameter. Slight modification would be required to use the same type of equipment at elevated temperatures and on larger tubes.

Remote Macro Camera. A summary paper titled "Improved Macro Camera for Hot Cell Application" was submitted for review by the ANS Hot Laboratory Division Program Committee for the 10th Hot Laboratory and Equipment Conference.

3. METALLOGRAPHIC LABORATORY

Examination of NPR carbon steel connector tubing at areas pinpointed by non-destructive testing has shown surface defects from 2 to 3 mils deep and occasionally as deep as 10 mils. These defects are inclusions of a foreign material, primarily iron oxide, which have become imbedded in the surface during mechanical reduction of the tubing. During bend testing these defects cause partial fracturing of the tube wall to a depth exceeding 100 mils. The steel beneath this thin outer layer appears to be of good quality.

Surfaces of copper, Zircaloy-2, uranium and aluminum have been cleaned by Coextruded Products Operation using their cathodic cleaning apparatus. The cathodic etching (cleaning) apparatus is in the developmental stage and work is in progress to determine optimum etching conditions for cleaning metal components prior to coextrusion. Argon etching pressure has been varied in the current test. Each of the four metals was etched under five different argon pressure conditions. Following cleaning, the metals were submitted to be replicated and subsequently examined in the electron microscope. Examination of the replicas

from areas representing the five etching conditions on a single metal reveals no significant changes have been induced by varying the argon pressure.

4. N-REACTOR CHARGING MACHINE

Modifications. The hydraulic motor which powered the vertical lift drive assembly developed a serious shaft seal leak. A new shaft seal was installed and operation of the motor is satisfactory.

The valves which control the hydraulic connector cylinders caused considerable shock to the charging machine when the valves were de-energized to allow the hydraulic connectors to retract. With the assistance of a Vicker's service man, this problem was investigated. It was recommended that the valves be changed to a double solenoid, three-position valve. A modified spool and a time delay relay will be used also with these new valves to correct the shock condition.

The charging machine transfer arms, when moved to positions 2 or 3, oscillated when they arrived at the desired position. This problem was caused partially by the electrical system tending to hunt and partially by the drive system which had considerable backlash. New drive shafts (torque tube) and solid-sleeve type couplings were installed to reduce the backlash and shaft angular deflection. Although this helped, it did not solve the problem. Plans for further modification of this system are under way.

The prototype test assembly for removal of pistons from empty magazines was fabricated and installed. The assembly was tested and works very well.

The new trolley wheels for the "C" elevator magazine support were fabricated, installed, and tested, but did not operate satisfactorily. An alternate design will be tested.

Testing. Nozzle-Fuel Foot Test No. 1 was completed and an informal report issued. The severe galling which was experienced between autoclaved components has prompted a request for further investigation into the matter.

All of the laboratory work has been completed on Design Test No. 8, Idler Roller Test.

All preparatory work is complete for Design Test No. 20, Process Tube Charging.

5. FISSION PRODUCT TRANSIENT SAMPLES FOR PHILLIPS

Fabrication of 72 fission product transient samples for Phillips Petroleum Company is continuing. Coextrusion of a number of these tubular elements clad with high purity aluminum containing Al - 3.2 w/o Pu alloy cores has repeatedly shown "dogboning" or core thickening at the trailing ends. Several core end configurations have been used and it appears that the minimum "dogbone" attainable with these alloys results in a cladding thickness 0.004-inch to 0.005-inch less than specifications require. Additions of 2 w/o Si to the high purity aluminum cladding material appears to have eliminated some of the "dogboning". One extrusion of this type resulted in a cladding thickness within specifications.

F. W. Albaugh

Manager, Reactor and Fuels Research and Development

FW Albaugh:kb

PHYSICS AND INSTRUMENT RESEARCH AND DEVELOPMENT OPERATION

MONTHLY REPORT

APRIL 1962

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

N-Reactor Program

The exponential mockup of N-Reactor has been rebuilt with modifications as outlined last month. Vertical traverses have been measured in the pile with enriched, wet fuel with six control rods. This measurement was taken without control rod extensions in the base region. This result will be compared with the measurement with extensions in place to see if there is any reduction of control rod transients near the base.

An experiment has been made to determine the thermal neutron leakage into a 1/2" BF_3 tube that has been covered with a cadmium sleeve which does not enclose the connector end. When the connector end was covered with cadmium, the counting rate was reduced by about 2%. Such corrections must be applied before true cadmium ratios are obtained.

Optimization of Retubed Lattices

The Pu-Be sources have been returned from Mound Laboratories where they were examined, recanned, and calibrated. Measurements have been resumed for the CII-N fuel in the water-cooled, C-reactor lattice.

PCTR Measurements for Large Diameter Fuels

A Quarterly Report was written entitled "Lattice Parameter Measurements for a 1.66-inch Uranium Rod in a 6.5-inch Graphite Lattice."

Data Correlation and Analysis

The derivation of effective cross sections for reactor lattices by fitting one-group P_3 calculations to copper flux traverses was continued to increase the range of lattices included in the analysis. Values of f for the analysis of exponential experiments can be calculated by using cross sections derived from this analysis. Several multigroup calculations were made in an attempt to derive effective neutron temperatures from the

analysis. Four cases gave results which could be represented by $T_n/T_m = 1 + 0.204\Delta$, where $\Delta = \frac{4\Sigma_a}{\xi\Sigma_s}$. Other cases gave poor results because of uncertainties in interpreting cadmium ratio measurements.

Double-Focusing Beta-Ray Spectrometer

A Lu-176m source was obtained for the double-focusing beta-ray spectrometer from an irradiation of Lu_2O_3. This was made possible by using graphite as the source mount. The purpose of preparing the source in this way was to prepare a stronger source than had been possible previously. The source also was counted with a 256 channel analyzer in order to determine the amount of Na^{24} contamination which had been troublesome previously.

The momentum distribution of the high energy beta particles which were emitted by the source were measured with the beta spectrometer. The distribution is being analyzed with the programs FERMI, FERMI-AUX and LLESQ.

The Modified Heavy Gas Model and Neutron Spectra

Investigation of the effects of preserving the logarithmic decrement rather than the average energy loss per collision in modifying the gas model has been completed for graphite. The results differ significantly from those obtained directly from the code SPECTRUM. Hence, in the thermal region preserving the logarithmic decrement produces an unsatisfactory theory. This suggests a need to investigate the role played by the logarithmic decrement in slowing down theory; such an investigation is now being considered.

Computational Programming Services

A number of simplifications in input have been added to TRIP004, the reactor kinetics code. The program now gives users the option of omitting certain data, for which it then supplies standard values. An improved method of choosing the next parameter is conditionally operative. This is expected to greatly speed convergence on cases which are sensitive to small changes in the parameter. On a preliminary test with a loose convergence criterion, the new method gave convergence in four iterations where 39 iterations had not satisfied the same convergence criterion with the old method. The TRIP document is ready for rough draft typing.

Reactor Lattice Physics has requested a program, BVT0C0, combining the two exponential data processing codes VT0CL and C0FIT2. BVT0C0 first processes vertical traverse data as VT0CL now does. It then varies the vertical

extrapolation (λ_z), using a specified set of points, until a minimum dB^2 (measure of the quality of the least squares fit) is found. The B_{11} and λ_z from this "best fit" are then used by the C0FIT2 method to determine λ_a and λ_b (extrapolation in the horizontal directions) from two horizontal traverses. The λ_a and λ_b are then re-entered into the VT0CL portion of the program, and a final fit is made with the "best" values of extrapolation in all three directions. In planning BVT0C0, primary consideration was given to compatibility with present data decks, rather than program efficiency; present VT0CL and C0FIT2 data may be submitted to BVT0C0 intact and without alteration. Nevertheless, a significant saving in machine time is expected and elapsed time (from experiment to final results) will be cut by a factor ranging from five to ten. BVT0C0 is in debug.

Instrumentation

The prototype unit for twenty-four transistorized gamma spectrometers to be fabricated by GE-APED for the NPR Fuel Failure Monitor was extensively tested and then returned to the vendor for modification of the high voltage supply. Modifications were completed and the spectrometer was just received back for final testing. A meeting was held with GE-APED, CE&UO, and Kaiser Engineers representatives to discuss other possible changes in the twenty-four final units, with a proposed change to solid-state relays throughout and to a different method of readout.

Drafting of the formal final report on the experimental Fast and Slow Scanning-Type Fuel Failure Monitor continued. Sections completed included the ratio-trip circuitry, comparison of ratio-trip to difference-trip circuits, slip-ring performance, and general mechanical design of the scanner.

The IPD Appropriation Request for experimental fuel element rupture instrumentation for the PRTR Rupture Test Loop was approved. Final design and specification of equipment are now beginning.

A new multi-channel pulse-height analyzer was used to further study gamma scintillation spectroscopy methods for determining the enrichment of uranium samples. Three different schemes have been developed for different accuracy requirements and sample geometries. If all samples are identical in size and shape (e.g., FPD fuel cores), a single channel measurement of the U-235 185-kev gamma will allow measurement of the enrichment to better than 5% of its value. The errors are primarily due to small variations in sample density and placement. If the samples are variable in size, but of the order of 0.25" thick by 0.5" or larger in diameter, the enrichment can be measured to within 10% of its value with a two-channel system. The second channel measures the Compton continuum from U-238 daughters at a higher energy, about 250 kev. This is used to normalize the count from the

first channel. If the samples are of similar size, the accuracy increases to about 2%. If the samples are more varied, or high accuracy is required, accuracies of the order of 2% of the enrichment can be obtained with a three-channel system. The third channel measures the slope of the U-238 continuum, which changes with sample geometry, to permit a correction for the normalization. The responses of all three methods do not vary linearly with enrichment for values above 10%, but all three could be calibrated to the stated accuracies for values up to full enrichment. Equipment is being designed for an FPD fuel core enrichment monitor and for an instrument to measure the enrichment of samples of varying sizes and shapes in the HLO 306 Building metallurgy labs. Both are expected to be helpful aids in nuclear safety control.

Systems Studies

Technical consultations regarding NPR instrumentation continued in support of the NPR Project Section. A recommended electrical network to make the 109 Building instrument outputs compatible with the central data logger inputs was accepted by both vendors concerned. Changes were recommended in the requirements for a power supply for the flow monitor in order to obtain responsive bidding. A functional review of the primary coolant pressure control system was completed and reported in a memorandum. Further attention was given to problems in isolating the flow logger inputs from the flow monitor signal lines.

Work began on the preparation of preliminary specifications for the NPR Simulator analog computer. On the basis of present scoping, the computer will consist of a double console with 325 amplifiers and associated nonlinear equipment, a single console with 100 amplifiers and associated nonlinear equipment, and all necessary auxiliary equipment, e.g., recorders, large screen scope, and special test equipment. The double console will be used in the plant simulator and the single console will be used for detailed studies on sub-systems. The specifications will be based on current state-of-the-art equipment. The computer will be capable of fast solution repetition and semiautomatic problem setup. A review of the needed control room simulation equipment has been started in light of recent design changes and increased knowledge of expected operating procedures.

Further experimental data were obtained for the reactor control study. The effects of control rod movements on the signals from four in-core neutron flux monitors were measured at 105-KW. Additional tests with other control rods are planned. Attempts will be made to correlate these data to obtain estimates of spatial reactor transfer functions.

An eleven region reactor kinetics simulation has been developed and checked on the analog computer. The transient response is excellent and shows errors less than 0.25% of full scale. This simulation is being programmed on the EASE and GEDA computers for refined studies of reactor instrumentation as related to nuclear safety and control. A method was found for eliminating eleven amplifiers from the simulation; these will provide spares which will reduce the downtime due to component failure during the problem runs.

SEPARATIONS

Experiments with Plutonium Solutions

Critical mass experiments were temporarily suspended during the month following the criticality incident at Recuplex on April 7. Three of the neutron detection instruments normally used in the critical experiments were removed to the 234-5 Building for neutron flux monitoring during the subsequent operations of securing this building.

Several critical mass measurements were completed during the first portion of the month prior to the criticality excursion at Recuplex, and the subsequent removal of these instruments. Measurements were made with the 14-inch sphere bare, and with reflectors of 1/2-inch and one inch of paraffin, including several experiments in which the sphere was only partially covered with paraffin. The experiments with these paraffin reflectors provided additional data from which to obtain more consistent values for the reflector savings of paraffin. The data resulting from these measurements are presented in the following table. (Table I)

Calculations for the reflector savings of the paraffin and the stainless steel shell of the criticality vessel are presented below.

TABLE I

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

(Measured Sphere Volume 23.22 Liters; Wall Thickness, 0.044-Inch)

Experiment Number	Date	Reflector	Pu(g/ℓ)	Acid Molarity	Sp.Gr.	H_2O (g/ℓ)	Total NO_3 (g/ℓ)	H/Pu (Atomic Ratio)	Critical Volume (liters)	Critical Mass (Kg Pu)
1141080	3/29/62	Unreflected	221	4.35	1.503	734	553	92.9	29.5	6.52
1141081	3/30/62	½" Paraffin	220	4.41	1.502	734	552	93.4	24.8	5.46
1141082	3/30/62	½" Paraffin, 5" diameter section moved ½" back from vessel	220	4.41	1.502	734	552	93.4	24.9	5.48
1141083	4/4/62	1" Paraffin	221	4.46	1.503	729	548	88.8	22.0 +.05 -.07	5.06
1141084	4/5/62	1" Paraffin, 10" diameter section removed from top of vessel	219	4.45	1.506	731	501*	93.5	23.2 +.12 -.17	5.08
1141085	4/9/62	1" Paraffin, 10" diameter section removed from side of vessel	219	4.45	1.506	731	501*	93.5	23.1 +.08 -.14	5.06

* Chemical analysis subject to change.

REFLECTOR SAVINGS

Reflector	Pu Concentration g/ℓ	Critical Volume (liters)	Equivalent Radius (cm)	Reflector Savings (cm)
0.044" SS	221	29.5	19.17	0.13
Corrected to "Bare" Unit*	221	30.1	19.30	
0.044" SS + ½" paraffin	220	24.8	18.10	1.20
0.044" SS + 1" paraffin	221	22.0	17.07	2.23

* Correction factor for the thin stainless steel shell of the vessel, 14 ml/mil of SS, previously determined.

In the next series of experiments a water reflector will be used. The removable reflector tank has been installed in preparation for the experiments with the 14-inch sphere fully water reflected. The neutron detection instruments which were used at the 234-5 Building, have now been reinstalled at the laboratory in preparation for these measurements.

Experiments with Plutonium Oxide - Plastic Mixtures

The fabrication of the split half machine for use in criticality experiments with PuO_2-plastic mixtures is about 90 percent complete. Preliminary testing of the electrical part of the machine has begun. The electrical design for wiring changes needed in the 209-E Building for the installation of this device is also about 90 percent completed.

Limiting Critical Concentration of U^{235} in Aqueous Solutions by Monte Carlo Techniques

The value obtained from Monte Carlo calculations for the limiting critical concentration of 93% enriched UO_2F_2 solution is 13.7 g U/ℓ. This is to be compared with the experimental value of 12.97 ± .05 g U/ℓ (12.1 g U^{235}/ℓ) determined from PCTR experiments and critical experiments at ORNL, res-

pectively.[1,2]

The value of k_∞ was computed for concentrations of 12.97 g U/ℓ and 13.5 g U/ℓ with the HISMC Monte Carlo Code. A total of four thousand neutron histories were traced for each concentration. The results of these calculations are summarized in the following table.

MONTE CARLO CALCULATION OF k_∞
FOR ENRICHED UO_2F_2 SOLUTION

Run No.	12.97 g U/ℓ			13.50 g U/ℓ		
	Fissions	Absorptions	k_∞	Fissions	Absorptions	k_∞
1	396.51	1018.23	.9463	427.45	1017.38	1.0210
2	401.11	994.10	.9805	403.25	974.17	1.0059
3	394.11	973.42	.9843	398.14	1015.42	0.9528
4	413.18	1011.58	.9925	407.14	989.31	1.0000
	$k_\infty = 0.981 \pm .010$			$k_\infty = .995 \pm .012$		

Limiting Critical Concentrations of Pu^{239} in an Aqueous Solution

Measurements were previously made in the PCTR to determine the limiting critical concentration of Pu^{239} in an aqueous solution.[3] In preparing

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(1) Masterson, R. H., Neeley, V. I. and Powell, T. J., "Limited Critical Concentrations for a Plutonium Nitrate Solution and for a Uranium-235 Solution," Nuclear Physics Research Quarterly Report for January, February, and March, 1962, HW-73116.

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

(3) Masterson, R. H., V. I. Neeley, and T. J. Powell, "Limited Critical Concentrations for a Plutonium Nitrate Solution and for a Uranium-235 Solution," Nuclear Physics Research Quarterly Report for January, February, March, 1962, HW-73116.

a report covering this work, the data were re-analyzed (and corrected) to yield a value of 8.4 ± 1 gm Pu^{239}/ℓ (the previous value given was 9.1 gm/ℓ). The new value is more consistent with computed values, being only about 1 gm/ℓ above the theoretic values.

Consulting Services on Nuclear Safety - Criticality Hazards

Participation on Hazards Review Group for Recuplex Criticality Incident

Consulting services on the 234-5 Building criticality excursion were provided to CPD during April 7-26, 1962. The excursion took place at 10:59 a.m., April 7, in the K-9 tank located in the SE hood of Recuplex (Room 221). From an analysis of the neutron decay with time, while the unit was in the subcritical state, it is estimated that K-9 ultimately became subcritical, and remained so thereafter, at ~12:30 a.m. on Monday of April 9. Hazards reviews were made of all actions taken to re-enter the Recuplex area, to determine a kill procedure, and ultimately to assure that criticality would not recur.

Neutron Flux Measurements in Recuplex Facility Following Criticality Incident

In connection with the recent criticality excursion at Recuplex, Critical Mass Physics personnel set up three complete neutron monitoring channels at the 234-5 Building. The electronics and associated recorders were located in Corridor 912 near the entrance to Corridor 3.

The "Safety Snooper" positioned the first two counters on the floor behind the SE hood at about 8 ft. from vessel K-9 (the criticality vessel) on April 13. These counters were perhaps the only units in the area sensitive enough, and at the same time strategically located, to obtain any useful information on vessel K-9 while in the subcritical state.

The third channel was installed a week later on a movable cart and was used to monitor the neutron flux adjacent to, and in the vicinity of, the RB hood.

These counters were used to monitor all operations such as 1) personnel entry into Room 221 (a "neutron instrumented" approach to vessel K-9 was used), (2) closing valve 312 which would have stopped air sparge if this were occurring, (3) shutting off the vacuum to K-9, (4) draining the solution from K-9, (5) procedure for poisoning the large K and L tanks with cadmium, (6) sampling procedure for the various tanks.

A small decrease in the neutron flux could be observed (a few percent in some cases) as cadmium was added to the large K and L tanks.

During the draining of vessel K-9 it was possible to obtain an estimate of the critical volume from the change in neutron flux. Since the draining process was more or less continuous, but somewhat intermittent, it was not possible to accurately correlate the observed flux values with solution volume in K-9. Also, there is some question concerning the hold up in the line from K-9 to the receiver vessel.

From the neutron flux measurements during the draining process, the critical volume in K-9 is estimated to be $\sim 43 \pm 2$ liters. The analytical results of the solution for Pu concentration were 35 g Pu/ℓ; with 120 g NO_3/ℓ. The above volume is in qualitative agreement with the estimated critical volume of ~ 42 liters, as predicted from theory for the above solution in K-9.

Subcritical Interactions

The interaction probability function has been analytically integrated for cases of (1) equal parallel slabs, (2) equal perpendicular slabs, and partially integrated for (3) equal parallel cylinders. Evaluation of the analytic functions requires far less machine time than straight numerical integration (in fact, for single cases the calculation can be done easily by hand), and the answers are more accurate, since the number of mesh points is no longer a variable. Comparison with the eigenvalues of last month's report shows two trends. First, the calculated k_{eff} are about 1-1.5% closer to the measured value. Secondly, for a series of calculations at different height and separation for the same system, the eigenvalue is nearly constant. Evidently the increasing or decreasing trends in k_{eff} shown by numerical integration are a function of the number of mesh points used.

In summary, a system has been found which predicts well the interactions of more than two components, or components of unequal thickness, but equal interacting area. A complete write-up of the method and the results are to be included in the Physics Research Quarterly Report.

Transport Theory Analysis of Pu-Metal Solutions

The difficulties reported last month have been traced to a mislabeled table of hydrogen cross section data in LAMS-2543. The table is labeled "Anisotropic Hydrogen Cross Sections," but in reality it is the isotropic cross sections with a small correction for second moment scatter.

The simple systems of bare α and δ-phase Pu and fully reflected δ-phase were run successfully, indicating that the actual problem of Pu metal surrounded by a Pu solution should give good results also.

Instrumentation and Systems Studies

Simulation studies of a separations "C" column continued to evaluate several tentative mathematical models developed by Chemical Research and Development. Some special techniques were developed for use with this study. The simulation was operated one hundred times faster than previously, thus generating a complete solution in one second. This made it possible to use the repetitive mode of computer operation and to display the solution on the large screen linear oscilloscope. The experimentally determined concentration curves for the twenty runs were set up on an arbitrary function generator. This was summed with the computed concentration curve to form an error signal. The error signal was squared and integrated to produce the error square integral. This integral is a very sensitive measure of the "fit" between the two curves. The X-Y recorder was set up with an automatic pen dropping circuit, thus making it possible to make a point plot. The chart paper was marked with three axes; a vertical axis for the value of the error square integral, a horizontal axis for the value of one of the variables, and a diagonal axis for the value of another variable. If the variable driving the horizontal axis is swept slowly through a given range, complete solutions can be generated for many values of this variable. For each solution, one value of the error square integral is determined and plotted as a point. If the reference points for the variable along the diagonal axis are changed as the variable itself changes, then the final result is a chart representing cross sections of an error surface. Eighteen of the runs based on one of the mathematical models were made. Two of the runs did not match the model. These runs will be made with a different mathematical model. This work is still in progress.

Consultation was provided on the design and location of cabling for a C-column data logging system. Specifications were compiled for the several types of cable needed to carry some thirty different signals, and a tentative low-noise grounding configuration was developed.

Instrument maintenance problems at the Critical Mass Laboratory have increased. New procedural checks of the criticality alarm system revealed that three neutron systems had inoperative BF_3 neutron counter tubes. It was determined that the tubes had been overvoltaged 100% through a fault in the design of the high voltage power supply. The circuit has been redesigned to limit the output voltage.

The temperature control system on the water reflector tank on the critical assembly at the Critical Mass Lab is being installed. The system will maintain a constant water temperature during critical experiments.

Circuit development continued for the control rod drive assembly for use at the Critical Mass Laboratory. All circuitry except the selsyn position indicator performed correctly. To obtain the desired position accuracy of 0.001 inch, it was required that the transmitter selsyn rotate 280 times for one revolution of the cable drum. During rod free fall, the selsyn inertia, magnified by the gear reduction, caused pin and set-screw failures in the gear train. After discussion of requirements, the position accuracy was changed to 0.01 inch, and this permitted changes in the gear train to nearly, but not completely, eliminate the problem. Finally, it was agreed to de-energize the transmitter selsyn during free-fall; however, this requires the operator to manually position the control room indicator after each scram. This was agreed to and will be used. As an alternate, the selsyn could have been clutch-coupled to the motor shaft; however, clutch slippage would reflect as phase shift with the position indicator then being incorrect. This method was discarded. Following the foregoing, the rod drop time under control was measured at 1.1 seconds in comparison to free fall time of about 0.4 second. All final required mechanical components have been ordered.

A number of neutron and gamma monitoring systems were installed in the 234-5 Building following the recuplex criticality incident. Three neutron monitors (a LiI(Eu) scintillation crystal, a fission counter, and a boron-lined proportional counter) were installed for neutron multiplication measurements. Six other neutron channels using double-moderated BF_3 tubes and scintillation detectors were installed to obtain neutron flux and dose-rate data. One of these was equipped with triggered multisweep oscilloscopes to record possible fast bursts over five decades of intensity. Two gamma channels were installed for dose-rate monitoring. All the temporary instrumentation operated without failure from the time of installation until removal one to two weeks later.

NEUTRON CROSS SECTION PROGRAM

Quasi-Elastic Neutron Scattering from Water

The processing of the quasi-elastic scattering data for neutrons of energies of 0.1, 0.15, and 0.25 ev was completed. The analysis of the diffusive broadening of the quasi-elastic peak and its dependence on momentum transfer was also completed. The diffusive broadening which was observed is not inconsistent with the simple diffusion model for the dynamics of the water with which the neutron interacts. However, the coefficient of self-

diffusion for water which is inferred from these results is approximately 50 percent larger than values obtained by other means, including the diffusive boradening measured elsewhere using very low energy neutrons. The significance of this result is not explained at the present time. An abstract of this work was prepared and submitted for the Symposium on the Inelastic Scattering of Neutrons to be held by the International Atomic Energy Agency in September 1962.

Inelastic Scattering of Neutrons from Water

A preliminary analysis has been completed of the results of inelastic scattering of neutrons of initial energies of 0.15, 0.2, 0.3, and 0.4 ev from room temperature water. The results were formulated in terms of the Egelstaff Scattering Law function $S(\alpha,\beta)$ where β and α are the reduced energy and momentum changes, respectively. These results cover the approximate ranges $2 \leq \beta \leq 10$ and $0.5 \leq \alpha \leq 15$. The results have been compared with the most recent results obtained by the Harwell-Chalk River cooperative measurements program. The Hanford results agree quite well with the Chalk River results in the region of overlap which is approximately for $\beta \leq 6$. The Hanford results are of generally much greater precision, however. The $S(\alpha,\beta)$ values have been used to derive the generalized frequency distribution function $p(\beta)$ of water. The $p(\beta)$ function which is derived shows the effects of the higher energy molecular states of water. Work is in progress to determine more accurate corrections for finite sample thickness and variation in spectrometer efficiency for a more complete analysis of the data. In addition, a few experimental runs for initial neutron energies of 0.4 ev are being repeated to resolve discrepancies noted in the preliminary analysis. These measurements have been interrupted because the spectrometer shielding was dismantled to permit reactor maintenance. An abstract on the inelastic scattering results was prepared and submitted for the International Atomic Energy Agency Symposium on the Inelastic Scattering of Neutrons to be held in September 1962.

Fast Neutron Cross Sections

The analysis of the total cross section data taken last December has been completed. The effects of zero-time instability were corrected for successfully in five out of the six samples which were investigated. The results of these measurements have been prepared for a quarterly report publication.

Radiographic testing was performed on 22 samples for total cross section measurements. Voids were observed in the samples of Li, Na, K, S, and Bi. The flaws in the Na and K samples are probably too small to be significant.

The preparation of a computer program for the reduction and analysis of total cross section data has been started.

Instrumentation

Assembly of the 1024-channel time-of-flight analyzer is 75% completed. This analyzer will be used with a rotating crystal neutron beam chopper for slow neutron energy spectra measurements in experiments with the neutron diffraction spectrometer. The analyzer can be split into two 512-channel or four 256-channel sections. Transfer of data from one section of memory to another, as well as subtraction or addition of any two sections, will be possible. Data stored in memory can be displayed on an oscilloscope or recorded on a parallel printer.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE

Status of PRTR Fuel Irradiation Experiments

Low Exposure Plutonium-Aluminum Elements (PRTR Test 13)

Element 5052 in location 1247 was discharged to the basin on April 2, 1962, having accumulated 30.7 MWD exposure. Element 5095 in location 1556 was discharged to the basin on April 18, 1962, with an exposure of about 41.3 MWD. The first element was discharged February 2, 1962, after 13.1 MWD exposure.

Mark I UO_2 Elements (PRTR Test 37)

Two elements are assembled and all six will be ready for charging during the extended shutdown scheduled for the first two weeks in May. Wrapping wires have been engraved with serial numbers and rod wraps are marked with an "X" to identify points where the wires cross the outer surface of the rods. The wire wrap around the element is similarly marked at each crossing of the major gusset direction. The test proposal, HW-73166 RD, was issued March 29, 1962.

Gamma Scanning Facility

Interference in B cell with the proposed supercritical loop installation is being investigated.

Radiochemical and Mass Spectrometer Work

All thirteen samples from element 5075 have been analyzed for plutonium content by coulometric titration, for Cs-137 gamma activity, for Sr-90 beta activity, and alpha counting extracted plutonium is in process. Three samples have been analyzed for plutonium-isotopic composition. Analysis of the results for consistency and interpretation has begun.

Foil Irradiations (PRTR Test 44)

A test proposal concerning the flux monitoring foil irradiations in PRTR fuel elements was submitted to PRTR Technical Planning Operation on April 13, 1962.

Preparations were made for attaching bare and cadmium covered lutetium foils to two unirradiated Pu-Al elements. These foils will be irradiated at minimum power in the PRTR for 30 minutes during the next startup. Calculations which will enable more accurate corrections to be made to the lutetium data have been requested from APO.

A collimator has been set up and tested for use in counting Co-Al flux monitors. These monitors are mounted on the fuel elements being irradiated in the PRTR. The collimator was tested using Co-Al monitors which were obtained in an experiment conducted in the MTR at Idaho Falls.

Other experiments were conducted with a 256 channel analyzer in preparation for counting Zr-Co flux monitors.

PRTR "Phoenix" Fuel Experiment

Further work on the possible PRTR Phoenix experiment has been carried out. The power change in the UO_2 regions of the central irradiation cell was analyzed, and it was found that the power increase is quite large--in one region up to six times the power with natural UO_2, and in the other region up to twelve times. This pertains to the case of 30% U-235 retaining the present absorption cross section in the process tube. It seems that the U-235 enrichment must be limited to less than ten percent, and that, instead, the Pu enrichment of the test sample can be increased, thereby, maintaining the necessary Westcott r value without exceeding tube power limits in the cell. Presently, calculations are being carried out for 5 w/o Pu-Al in the central region.

Critical Mass Studies for 1.8 w/o Pu-Al Fuel in H_2O

Experiments to determine nuclear parameters for 1.8 w/o Pu-Al fuel are being conducted in the tank in the TTR reactor room. The experiments consist of approach to critical and exponential measurements with lattice spacing of 0.75, 0.80, 0.85, 0.90, and 0.95 inches. The information will be used to revise the specifications for fuel element storage in the PRCF cell and to check the analytical techniques for H_2O moderated reactors. In turn these techniques will be used to furnish information about the H_2O moderated lattices planned for the PRCF.

The approaches to critical have all been completed. Table I gives the critical loadings which were indicated by extrapolation of the inverse multiplication curves. The highest multiplications occurred for loadings of 95% of the critical masses. The results for the 0.80 inch lattice spacing have not been analyzed by the least squares method. There may be a small change when this analysis is made.

TABLE I

Triangular Lattice Spacing (in. Center-to-Center)	H/Pu	Critical No. of Rods	Critical Weight (Kg Pu)	Total B^2* ($10^{-6}cm^{-2}$)
0.75	638	580 ± 2	4.13	6335
0.80	819	509	3.62	6339
0.85	1013	494 ± 2	3.52	5954
0.90	1218	516 ± 2	3.67	5340
0.95	1435	578 ± 2	4.12	4582

* The B^2 numbers include 609 x $10^{-6}cm^{-2}$ for a vertical buckling and assume an extrapolation length of 7.7 cm since the loadings are totally reflected in all dimensions.

The Pu-Al rods are 44 inches long, 0.500 inches in diameter and contain 1.82 w/o Pu. An average composition for the Pu by weight is 93.93% Pu^{239}, 5.62% Pu^{240}, 0.442% Pu^{241} and 0.014% Pu^{242}. An average composition for the aluminum by weight is 0.65% Si, 1.65% Ni, 0.47% Fe, and 97.23% Al. The alloy is clad in 0.030 inch thick Zircaloy-2 with an outside diameter of 0.566".

Work that remains to be done are the exponential experiments on the 0.75 and 0.80 inch lattice. Also, since the fuel rods that have been used in these experiments have different amounts of Pu^{240} content (5 to 6%), an approach to critical using zoned loadings in the 0.85 inch lattice instead of a random loading that was used might be of interest and is being considered.

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

Zero-Dimensional Analysis of Pu-Al-H_2O Assemblies

The objective of this brief study was to examine the applicability of the Westcott cross section formalism to the analysis of some Pu-Al-H_2O assemblies, and to suggest possible prescriptions for survey calculations of reactor classes not directly tractable by means of the Westcott routines. While more refined methods of analysis, of course, exist, the simple zero-dimensional approach is presently widely used in reactor survey calculations.

The Westcott parameters can be used to generate reaction cross sections for well-moderated assemblies and, hence, k_∞ can be determined. For the analysis of finite assemblies, separate leakage calculations must be performed to arrive at a k_{eff} value which can be compared with the experimental value. For this purpose the Deutsch prescriptions(4,5) were used.

The Pu-Al-H_2O experiments, analyzed in this study, are reported in reference (6). Summarizing briefly: 24" long, .506" diameter Pu-Al alloy rods with 5 w/o Pu enrichment, clad with 30 mils of Zircaloy-2 were positioned in a hexagonal pattern in a light water medium. The Pu contained ~5% of Pu-240. The lattice pitch was varied from .75" to 1.80" and neutron multiplication and exponential measurements were conducted.

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(4) Deutsch, R. W., "Computing Three-Group Constants for Neutron Diffusion," Nucleonics 15(1): 47 (1957).

(5) Deutsch, R. W., "Method for Analyzing Low-Enrichment Light-Water Cores," GNEC-133, October 5, 1960.

(6) Neeley, V. I., R. C. Lloyd, and E. D. Clayton, "Neutron Multiplication Measurement with Pu-Al Alloy Rods in Light Water," HW-70944, August 29, 1961.

In analyzing these assemblies a series of "refinements" were considered separately:

(1) Effect of "neutron temperature."
(2) Effect of "Bell correction."
(3) Effect of thermal disadvantage factors.
(4) Effect of reactor leakage on reactor spectrum.
(5) Effect of spectrum departure from the Maxwellian - 1/E shape.

If all those effects are taken into account, the calculated values nearly fall within the experimental error. All the above effects can, of course, not be directly deduced from a zero-dimensional study using Westcott parameters. Hence, for survey calculations, a series of normalizations are required, which improve the calculational accuracy somewhat, without unduly complicating the calculational procedure.

Heterogeneous Pu-H_2O Reactor Physics Calculations

An effort is being made to use three-group diffusion theory and related theoretical models of cross section specifications to compute the behavior of multiplying systems of Pu-Al rods in H_2O. The experimental data used in these analyses are described in the preceding section.

The pertinent result for theoretical comparison is the estimated critical number of rods at each lattice spacing, shown in Table I:

TABLE I

ESTIMATED CRITICAL NUMBER OF RODS

Lattice Spacing (inches)	Experimental No. of Rods	Theoretical Results I	II	III	IV	Calc. k_{eff}*
0.75	356.1	371.6	366	390	350	0.990
0.85	230.2	258.7				0.971
0.90	192.0	230.7				0.954
1.00	170.1	200.5				0.960
1.10	166.5	191.5				0.970
1.20	181.1	196.1				0.984
1.30	215.5	217.9				0.998
1.50	307.8	370.5				0.981

* For the experimental number of rods using method I.

Theoretical values were computed using the SWAP code. All cross sections for materials other than fuel, and the high energy group fuel cross sections are group-averaged values obtained from APEX-515(7). Thermal fuel cross sections are Westcott cross sections with an upper energy limit at 3.681 KT. Macroscopic epithermal (group 2) fuel cross sections are given by:

$$\Sigma_x = No\,(S_4+b_4\,g_x)\,\sigma_{ox}\sqrt{\frac{\pi T_o}{4\,T}}\left(1+\frac{N_o\,\sigma_{peak}}{\Sigma_s+\tau}\right)^{-\frac{1}{2}}$$

where

$\tau = S_o = (1/D)$

D = fuel rod diameter

} theory result I

$$\tau = \frac{S_o\,\Sigma_1}{\Sigma_1 + S_o\,(V_o/V_1)}$$

Σ_1 = total moderator cross section

V_o = rod volume

V_1 = moderator volume in unit cell

} theory result II

N_o = isotopic nuclear density

S_4 = non-1/V correction for the epithermal group

b_4 = group cutoff parameter

g_x = non-1/v correction on Maxwellian

T_o = 293 K

T = neutron temperature

σ_{peak} = cross section value at resonance peak

Σ_s = fuel rod scattering cross sections

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(7) Henderson, W. B. and Mary J. Stanley, "Cross Sections for Reactor Analysis," APEX-515, GE-ANPD, 1957.

The above model assumes the infinite dilution resonance integrals given by Westcott, and an empirical resonance self-shielding prescription. In results I and II of Table I, cross sections are based on an assumed 20 C neutron temperature, group cross section decreased slightly, using the results of Bell[8]. Result III differs from result II in that the neutron temperature (140 C) in the fuel is computed by the method given by Deutsch:

$$\frac{T_{eff}}{T} = \left[1 + 0.75 \frac{\Sigma_a (KT)}{\xi \Sigma_s}\right]^2$$

with T_{eff} and T in degrees K and the homogenized cell values of the cross sections.

Results I, II, and III assume an axial reflector savings of 9.0 cm on each, and in all three energy groups. The core in all cases consists of an homogenized medium, specified by cross sections which are volume- and flux-weighted values computed in a cell flux calculation using SWAP.

Reactor calculations have shown that in the tightly packed lattices the epithermal-to-thermal flux ratio drops sharply in the core near the reflector boundary. This effectively increases the specific multiplication rate in that zone. Calculations show that in the case of the core with lattice spacing of 0.85 inch, removal of an outermost rod perturbs the system -0.9 mk, whereas, the removal of a center rod has an effect of -0.2 mk. Result IV reflects the results of a perturbation calculation to correct result III for the lower spectral ratio near the top and bottom reflectors.

The corrections of results II, III, and IV are expected to be strongest in the more tightly packed lattices, and of nearly no effect in the case of the 1.5 inch lattice.

The Plutonium Recycle Critical Facility

Experiments in which H_2O is used as moderator in the PRCF are scheduled for approximately eight months after the startup with D_2O. Letters which describe some of the modifications which are necessary to the PRCF and which contain a description of the scope of the experimental program have been forwarded to PRCFO and MEEO. These letters will be a basis for scoping the design modifications.

(8) Letter to Editor, George I. Bell (LASL), Nuclear Science and Engineering, Vol. 5, p. 138, 1959.

The detailed procedure for the first startup test was reviewed and revisions suggested to TPO. A chronological outline and tentative time schedule for the startup experiments has been prepared. The total time estimated for the experiments is 50-60 eight-hour days. This is longer than the original estimate of two months. The reason for the difference is that parts of the experiments which are to be done towards the end of the startup include some post-startup experiments. They are scheduled to be done during the startup.

Graphite Lattice Parameters for Low Exposure Pu-Al Fuel

Previous attempts to use Program S[9] for the analysis of plutonium-aluminum fueled lattices were unsuccessful. Inconsistencies in input data have been found and corrected during the past month. A calculation has been made in which non-negative fluxes were obtained. The values of the various cross sections in at least the two thermal groups still need refinement. Also, the anisotropic scattering cross sections that have been calculated by GAM-1[10] are incorrect. This is due to a peculiarity of the code itself. The program integrates the scattering kernel over the neutron current spectrum to obtain the anisotropic scattering cross sections. However, in an infinite, homogeneous system the current vanishes everywhere and the code gives zero for all the anisotropic cross sections. Attempts to circumvent this by keeping the system large but finite have been unsuccessful. Because of this difficulty, it has been decided to proceed with the cell calculation, neglecting anisotropic scattering for the present time.

Ceramic Reactors for Space Applications

The effect of changing from graphite to nickel reflectors for some compact ceramic reactors was briefly examined. The analyses showed that a 2.5 cm nickel reflector substituted for a 10 cm graphite reflector increased the total reactor weight. It was also found that 2.5 cm of graphite is nearly as effective as 2.5 cm of nickel as a reflector. Hence, this substitution of nickel for graphite does not seem promising.

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

(10) Joanou, G. D. and Dudek, J. S., GAM-1: A Consistent P1 Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants, GA-1850. June 28, 1961.

In connection with the ceramic reactor work, another set of 16-group cross sections for use in the HFN code has been prepared and placed on tape. This set is based on data from LAMS-2543 by Hansen and Roach. Thus far, the agreement between calculations using this set and the Yiftah, Okrent, and Moldauer (YOM) set of cross sections is quite poor, and an explanation of the differences is being sought.

Fast-Thermal Reactor Complexes: "Fuel Reuse"

The single pass fuel reuse cycle described in the last report has now been extended to a double cycle. The results are presently being analyzed. (Work on fuel reuse is being carried out jointly with Programming Operation.)

Code Development

CALX

The planned modifications to the MELEAGER burnup code have become so extensive that it was decided to give the new burnup code a new name - CALX. Its companion cross section code, which will combine results from GAM and TEMPEST into a single CALX data tape will be known as SIGMA-3.

Data Processing personnel have modified GAM. The modified GAM optionally writes the results of its slowing-down calculation to a binary tape, for further analysis by SIGMA-3 (or some other program). A similar modification to TEMPEST will be made. Data Processing has also changed GAM to allow a more flexible treatment of resonance self-shielding, though the code for putting the needed additional parameters on the data tape is still being planned.

The multi-group spectrum code for CALX, which is similar to ANNE, is essentially all coded. SIGMA-3 remains to be written.

Transport Theory Development Work

Application of the S_n transport analysis to evaluation of multiple-scattering effects, in support of cross section measurement work on the double-diffraction neutron spectrometer, revealed evidence of numerical instability in the calculation of back-scattered flux near the emergent-beam face of thin foils. Further scrutiny of symptoms and trends indicated that angularly-integrated flux and current met neutron balance requirements satisfactorily, and traced the origin of the instability to a fluctuous distortion of higher harmonics within a mean free path of slab-void interfaces. This subtle effect escaped previous observation perhaps because most nuclear

analysis work is done in cylindrical geometry with major attention focused upon criticality, power distribution, or foil activation rather than upon the fine structure of the neutron angular distribution.

Reformulation of the transport analysis to provide numerical stability control by flexible selection of angular integration sequence, as summarized in Figure 1, has eliminated the instability difficulty. But the reason for the success of this procedure is not yet fully understood. From both physical analogy, and from mathematical consideration of the integral equation formulation of transport theory, it becomes clear that numerical stability is best when the transport integration follows the last-free-flight path for the flux and the next-free-flight path for the adjoint.

Yet, there seems to be more to it than this. For all the integration patterns in Figure 1 meet this basic requirement, and yet each pattern displays a geometry-dependent range of instability. Empirically, it appears that the angular integration must not pass through the transverse direction in rectangular geometry, but must pass through the transverse direction in curvilinear geometry, for best numerical stability.

In summary, however, the improved double-S_n transport analysis (GE-HAPO program S) now provides sufficient flexibility in choice of angular integration sequence to achieve numerical stability in all geometries. As indicated in Figure 1, the recommended "always safe" choices are the double-S_n alternating sequence (pattern A) for rectangular geometry and the double-S_n successive sequence (pattern B) for curvilinear geometry.

A wide range of other angular integration sequences has been incorporated into the improved analysis, to facilitate further stability-optimization study. Exploratory theory-improvement work involving selection of other integration sequences must, of course, be accompanied by close scrutiny for any evidence of numerical instability (non-physical flux distortion persisting for many iterations).

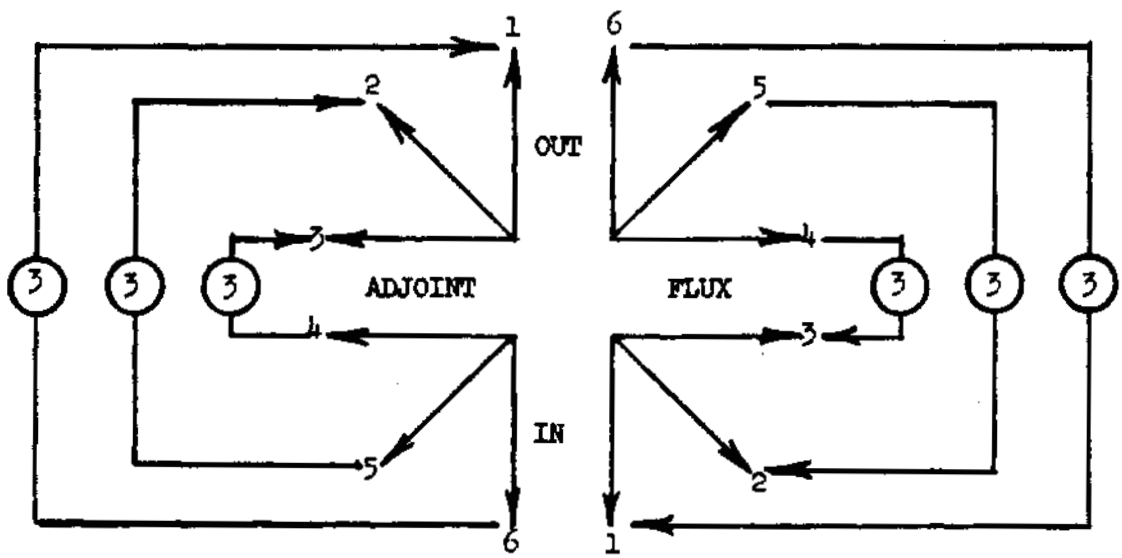
The improved analysis has been programmed for the IBM-7090, is being distributed internally to those doing transport analysis work, and is expected to be suitably documented for external distribution in the next Physics Research Quarterly Report.

RBU

The Monte Carlo flow charts are completed and preliminary investigation shows the logic to be correct. The changes in logic necessary to allow treatment of bound scattering centers on the neutron thermalization process have been completed in the collision routine.

Figure 1: GE-HAPO Program S

Double-S_n Radial-Angular Integration Sequence, with Chain-compounded Nest of Three-Pass Albedo Loops, Shown for Double-S_2 Angular Basis



Geometry Sequence	Maximum-Stability Integration Sequence: Zero Outer Albedo	Maximum-Stability Integration Sequence: Non-zero Outer Albedo
A. Rectangular. Alternating.	1-6-2-5-3-4.	1-6-1-6-1-6- 2-5-2-5-2-5- 3-4-3-4-3-4.
B. Curvilinear. Successive.	1-2-3-4-5-6.	1-2-3-4-3-4-3-4-5- 2-3-4-3-4-3-4-5- 2-3-4-3-4-3-4-5-6- 1-2-3-4-3-4-3-4-5- 2-3-4-3-4-3-4-5- 2-3-4-3-4-3-4-5-6- 1-2-3-4-3-4-3-4-5- 2-3-4-3-4-3-4-5- 2-3-4-3-4-3-4-5-6.

[redacted]

1234450

Results of discussions during a recent meeting at Atomics International have provided several possible refinements in the Monte Carlo techniques used in RBU. These refinements are planned for use in an advanced version of the RBU Monte Carlo to be formulated in the future. Further planning of future investigations using RBU indicates that the analyses of lattice experiments, following the current evaluation of uranium and plutonium solution experiments will provide the necessary information needed to begin the combined Monte Carlo, Diffusion, and Burnup studies associated with the PRP program.

Instrumentation and Systems Studies

Recommendations were made regarding instrument adjustments to improve operation of the PRTR Fuel Failure Monitor. These included: Electronic system tests, calibration with a standard Cs^{137} source, setting of discriminator levels, tests to obtain phototubes with low drift characteristics, and reduction of excessively high count rates.

The experimental prototype transistorized PRTR Liquid Effluent Monitor was tested for the month with an EMI/US-9536-B phototube. Continuous operation, with a radionuclide source at 10^5 counts per minute, was satisfactory with no gain-drift changes observed on the recorder chart. Two of the stable phototubes were tested. The commercial contacting-type meters have not proven satisfactory for this application due to some feedback between the trip relays and the measuring circuit. Investigations were started on two types of commercial magnetic amplifiers for possible use in the alarm-trip circuits. The instrument was designed to monitor and provide alarm functions for gamma-emitting radionuclides in liquid from 10^{-5} gamma μc/cc to 10^{-1} gamma μc/cc.

The feasibility study for acquisition of an experimental process control computer is being revised to agree with the format required for compliance with an administrative directive from the Commission.

Arrangements have been made to install transducers on the PRTR degasser system as required to simultaneously record degasser pressure, level and liquid effluent flow during the next scheduled outage. The transducers have been tested in the laboratory. Recording of these quantities will be used with other available data in an attempt to determine the cause of the erratic variations in measured degasser level and effluent flow.

The equipment required to measure and record PRTR moderator level simultaneously with neutron flux has been assembled and tested. It is planned to install the equipment during the next PRTR outage. The test may not be scheduled until the following outage, however, due to the priority of other

tests. Recordings will be made on magnetic tape of neutron flux level from two sources and of moderator level height from two sources. The recordings will be made at "zero power" level; i.e., in the neighborhood of ten kilowatts. It has been observed that the moderator level varies in a somewhat random manner when power is held at equilibrium--especially at a low level. It is expected that analysis of the contribution of moderator level variations to a flux frequency response curve might improve the accuracy of measurements of the ratio β/ℓ^*, and might even yield information as to the source of the moderator level fluctuations. Most of the electronic equipment necessary for the recording has been acquired. A diagram was prepared showing the intended position of each component in the 309 Building.

The four-transducer, ultrasonic resonance probe assembly for measuring the wall thickness of the PRTR process tubes has been assembled. The probe and its associated electronics are to be incorporated into the present PRTR process tube inspection system following a series of performance and calibration tests in the 314 Building mockup facility. This equipment will measure the tube wall thickness (at 90° intervals) over a 0.055 inch range (0.125 to 0.180 inch) at traverse speeds up to four feet per minute. From bench test results, an over-all system accuracy within ± 0.002 inch is expected. It may be necessary to mount the oscillator near the top of the inspection tower to minimize cable losses.

U^{233} - Thorium Experiments in the PCTR

Specifications for U^{233}-Thorium oxide fuel elements were prepared and sent to Brookhaven. The fuel elements would be used for experiments in the PCTR on both graphite and water lattices, the latter in conjunction with a Brookhaven series of experiments on similar elements. There is still a question concerning the compatibility of the use of the elements by Brookhaven and by Hanford Laboratories.

NEUTRON FLUX MONITORS

Technical editing of the formal report on long-life plutonium and uranium in-core neutron flux detectors of the Phoenix or regenerating types was completed. Reproduction and issuance of the report is delayed pending classification review; some of the calculations may be relatable to planned future experimental tests in a Hanford production reactor.

Concurrent with a change of technical personnel on this program, the analytical results and planned experimental tests have been critically reviewed. The findings to date indicate that: (1) The Phoenix concept at least doubles the lifetime of a detector within a given sensitivity range; (2) Use of fertile and fissile nuclides of plutonium and uranium provides methods for obtaining various levels of sensitivity; (3) The time the

detector sensitivity stays within pre-established limits for usefulness depends upon both inherent sensitivity and flux environment; and (4) Deviations from nominal reactor environment can influence detector sensitivity in an exposure-dependent manner.

Efforts are being made to formulate a more concise description of the interrelated parameters for regenerating detectors to guide the next phase of experimental studies.

The experimental microwave equipment, long delayed in delivery, arrived and was assembled during the month. Work is under way to reduce system noise, increase sensitivity, and calibrate on known absorption lines of NH_3. Experiments are being designed to first examine the feasibility of monitoring neutron flux by means of the $Cl^{35}(n,\gamma)Cl^{36}$ reaction.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

The separation of small variations in three parameters in a simulated multilayer nondestructive test problem has been demonstrated using multiple parameter eddy current testing equipment now under development. False readout of the parameters being observed, that is, apparent interchannel interference, has been reduced to eight percent of full scale or less. This represents the end of the phase of the work in which the objective was to demonstrate that more than two parameters could be separated by the proposed method. In the next phase the separation of three or more parameters in metal test specimens will be demonstrated, and design data will be obtained for applying the new testing technique to a production testing problem.

The present separation results obtained in the three-parameter case are due to improvements in the over-all stability of the equipment and in changes made in the simulated test specimen to increase the difference between signals from the three parameters. As the parameters which are varied become more nearly identical, the separation becomes more difficult, and the interchannel interference increases. Separation error similarly increases as the number of parameters are increased.

Heat Transfer Testing

Further development of heat transfer testing instrumentation has been aimed at detection of small differences in heat transfer due to metallurgical differences in fuel core to cladding bonds. Differences in surface emissivity of the fuel elements is the present factor determining the upper sensitivity limit of the test. A dual radiometer for use in an emissivity-

independent heat transfer testing system has been fabricated. Installation of a 50-kw, 10-kc induction heater for special heat transfer testing applications is complete. Initial tests show it will be desirable to compensate for the distributed inductance of the high frequency leads connecting the generator to the work station.

X-Ray - Stress Analysis

Stress analysis by the conventional X-ray diffraction line shift method can be applied to well annealed, elastically-strained steel with an accuracy of about ± 20%. An attempt to measure stresses in welded steel pipes, using available equipment, has shown that plastic deformation occurred in regions near the welds. Due to this plastic deformation, reasonable accuracy can only be obtained on stress relieved pipes. Even after stress relieving, the average of a number of measurements are required to give a reasonable indication of the engineering stress in the pipes, since some microscopic residual stress remains. The average of six measurements adjacent to the weld in two pipes stress relieved one-half hour at 1325°F, and in one stress relieved one-half hour at 800°F, gave an estimated accuracy of about ± 5000 psi.

Each stress measurement requires a least squares fit of a straight line plot to four data points, and determination of the standard deviation of the slope of the line. Because of the number of measurements and the statistical analysis required, the X-ray line shift method is too cumbersome in its present form for use as a routine production test for welded steel pipes. Its main usefulness would be as a research tool.

There appeared to be a correlation between the stress relief temperature and the 95% confidence interval of the stress measurement. This is reasonable, since the errors in measurement are largely due to micro residual stresses.

Although promising, the results would have to be confirmed by further experiments because of the possible effect of initial conditions and composition variations. Statistical analysis of the data is continuing, but no further experimental work is planned at this time.

A meeting was held with NPR Primary Loop Engineering and Offsite Inspection to discuss possible application of X-ray stress measurement techniques to NPR piping. The new maximum temperature indication method disclosed in HWIR-1498 was described, and its possible use to replace thermocouples now being used to monitor stress relieving temperatures was discussed.

Zirconium Hydride Detection

Dislocation theory indicates that the stress dependence of ultrasonic attenuation in Zircaloy-2 may be different from that in hydrided Zircaloy-2 samples. To obtain more information on work in this field, a literature search was made. Several good references were found describing the sensitivity of ultrasonic attenuation and velocity to elastic and plastic deformation (between 0 and 0.5% strain), and recovery. Most of this work was performed by Alers, Hikata, and Truell, who used the pulse technique described by Roderick and Truell (Journal of Applied Physics, 23, 267 (1952)), at frequencies between 5 and 10 megacycles. These investigators explain their results on the basis of dislocation theory. It was also found that dynamic stress affects the attenuation more than static stress. Investigators at Oak Ridge have made measurements of ultrasonic attenuation in Zircaloy-2, and they found that the attenuation was proportional to the fourth power of frequency, indicating that Rayleigh scattering is the principal mechanism of the attenuation (these measurements were made between 5 and 40 Mc). Most investigators have used salol as the coupling media for their attenuation measurements.

Experiments made with aluminum to check the laboratory method have shown a stress dependence of attenuation, the attenuation being greater for dynamic stress than for static stress of the same value. These attenuation measurements were made with the pulse technique (Roderick and Truell) at 15 Mc, with applied compressive stresses between 0 and 9000 psi. The aluminum sample was made with a cylindrical plug machined on one face, and the quartz transducer was attached to this plug. This was done to eliminate shear stress in the salol, which was used as a couplant between the aluminum and the quartz transducer. Salol coupling is affected by changing stresses, thus giving erroneous attenuation values, unless the transducer is applied to a surface isolated from strain.

Eddy Current Testing of Carbon Steel Reactor Piping

Fourteen samples of 8½-inch-O.D. x 1.0-inch-wall piping of varying heat treatment histories was received. The maximum temperatures of treatment were as follows: 800°F - four samples; 1125°F - four samples; 1325°F - four samples; 1500°F - one sample; and 1800°F - one sample. Preliminary testing of these samples, with a General Electric Metals Comparator and a specially fabricated test probe, indicates a difference can be detected with this technique. However, at this time it is necessary to make several observations around the periphery of each sample and examine the data for statistical differences to obtain separation between adjacent temperatures; i.e., to separate the samples treated at 1125°F from samples treated at 800°F or 1325°F.

Statistical methods are not required to distinguish between 800°F and 1325°F or higher, or between 1125°F and 1500°F or higher. It is hoped that further refinements, both in the test probe and in the test procedures, will increase the sensitivity of this test, to the variable in question, to the point where statistical evaluation is unnecessary.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

A major effort for the current period was the consolidation of data and the preparation of a formal interim report covering progress on the program through March.

Measurements of ultrasonic response amplitude as a function of incident angle continued. The relationships are being investigated at several frequency depth products with rectangular and circular lithium sulphate transducers. Measurements of ultrasonic response losses as a function of metal-path length for the incident angles with peak response also continued.

A tentative alignment procedure has been developed for the Immerscope which does not depend on transducer frequency characteristics. For the previous method of alignment a transducer was arbitrarily selected and a reflected signal was used for adjusting the instrument center frequency and bandwidth. Center frequencies and bandwidths can vary unpredictably among different crystals. For instance, one particular Immerscope was found to be aligned, using a particular 10 Mc crystal to a center frequency of 8.2 Mc at the 10 Mc front panel setting. The bandwidth was found to be 2.4 Mc. The new method uses a calibrated, modulated sine-wave oscillator as the signal source. This method permits accurate alignment at a center frequency of 10 Mc with a predictable bandwidth. The best bandwidth for Lamb wave testing is largely undetermined at this time. However, with the reliable alignment procedure developed, various bandwidths can be selected for study.

Techniques to measure the frequency components contained in the characteristic Immerscope transmitter pulse are being developed. This measurement is very difficult, since pulse sampling by most techniques gives distorted information. The simplest method uses a receiver crystal as a sampling element. Since receiver crystals themselves have oscillating properties, they can add undesirable frequency components to the sampled pulse. An attempt has been made to passively sample the transmitted pulses with a transducer having a much higher resonance frequency than the pulse frequency. For sampling 10 Mc pulses, a 25-Mc quartz receiver is being used. However, the received pulse is accompanied by a portion of the transmitter pulse which is radiatively coupled to the receiver crystal. This transmitter pulse must

be eliminated before the receiver pulse can be frequency analyzed. Methods for gating out the transmitter pulse are being developed.

Analysis of the effects of defect depth continued. Lamb wave modes with phase velocities greater than the longitudinal velocity, V_L, were studied. The results of the studies are not complete; however, it appears that the behavior of modes with phase velocities greater than V_L could be rather erratic under certain circumstances. Such conditions are particularly true of symmetrical modes which have an approximately constant phase velocity over a fairly wide range of frequency thickness product. In one respect this condition would appear to be a desirable testing criterion, since the sample thickness can vary over a greater range while operating at one frequency. However, analytical work has shown that in these situations the particle vibrations on the sample surface change drastically with very small changes in entry angle. A one-degree change in entry angle could cause the surface particle vibration to change from predominantly horizontal to predominantly vertical. The mode propagation thus suffers a very large change in attenuation leading to complete loss of sensitivity for detecting defects. The above effect has not yet been experimentally examined.

Experimental work to measure the actual response over wide ranges of defect depths is continuing. Completed data have not been fully analyzed.

PHYSICAL RESEARCH - 05 PROGRAM

Mechanism of Graphite Damage

Work is continuing on improving the reproducibility of resistivity measurements. Fittings are being designed that will permit replacement of the vacuum pumps on the electron Van de Graaff by an ion-getter pump.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Investigations of the spectrum of the wind velocity and its effect on the mean horizontal dispersion of a smoke plume continued. Values of the standard deviation of the mass distribution, σ_y, were derived directly from the wind direction time series at the seven foot level of the Hanford Tower for nine Green Glow Tests. These calculated values, $\sigma_y(c)$, are tabled below along with the observed values, $\sigma_y(o)$.

[redacted] [redacted]

ARCWISE STANDARD DEVIATION OF MASS DISTRIBUTION (METERS)

Run No.	Standard Deviation (meters)	200	800	1600	3200	12800	25600
5	$\sigma_y(c)$	7	40	95	170	1050	
	$\sigma_y(o)$	12	35	65	160	820	
6	$\sigma_y(c)$	25	88	160	225	265	270
	$\sigma_y(o)$	34	125	200	270	530	1850
7	$\sigma_y(c)$	21	84	125	170		
	$\sigma_y(o)$	37	145	215	300		
8	$\sigma_y(c)$	12	42	120	270		
	$\sigma_y(o)$	14	70	160	245		
9	$\sigma_y(c)$	-	80	150	285	725	
	$\sigma_y(o)$	29	110	150	305	1500	
15	$\sigma_y(c)$	55	210	330	530		
	$\sigma_y(o)$	55	190	300	500		
17	$\sigma_y(c)$	-	95	190	320	200	930
	$\sigma_y(o)$	20	75	105	265	310	825
18	$\sigma_y(c)$	15	55	95	150	180	750
	$\sigma_y(o)$	15	60	145	340	470	2000
26	$\sigma_y(c)$	13	30	65	140	200	
	$\sigma_y(o)$	16	40	70	120	430	

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The model is based on the rather restrictive assumption that the wind vectors at all points in the test grid were identical to that measured at the source point throughout the period of testing. The validity of this assumption is readily checked, since wind data are available at the source, two miles from the source and at 13 miles. It was concluded that the assumption is fair within two to four miles from the source, but very poor at 13 miles' distance. Comparisons of observed and calculated values of σ_y bear out these expectations. Investigation will continue toward development of a less restrictive and more realistic model.

Because of the general arcwise sweeping of the plume, caused by a general trend in wind direction, any relationship between thermal stability and peak exposure values has been masked for the Green Glow and Dry Gulch series. The relationship has been demonstrated for Ocean Breeze tests since trends in wind direction are extremely small relative to the other series. To remove the effect of trend in the Green Glow and Dry Gulch tests, exposures have been integrated along the arc so that the cross-wind integrated exposure normalized to the source strength, $\frac{CIE}{Q}$, when plotted against distance, x, shows a noticeable stratification with respect to measured thermal stability.

Off-site work in connection with the diffusion program at Cape Canaveral, Florida, was completed, embracing a total of 76 experiments during a variety of meteorological conditions. In addition to successfully providing the required diffusion predictions in support of advanced missile firings, the data comprise an important addition to the Hanford and Vandenberg data for evaluating topographic and climatic influences on atmospheric dispersion patterns.

Progress has been made in the development of a multi-pigment tracer technique. Twenty-five different types of fluorescent and phosphorescent pigment have been artificially distributed on more than 100 filters, and the number of particles has been estimated by microscopic examination. The feasibility of using any of these will now be determined by studying their counting characteristics by the Rankin counter.

Dosimetry

A new thin, large-area scintillation crystal was received to be used in bremsstrahlung counting. It is mounted on quartz rather than glass and is three times as thick as the old crystal. Sensitivity to P-32 was 20% higher and the background was 2/3 that of the old crystal.

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Two batches of germane (GeH_4) were made and tested for use in proportional counters for plutonium X-rays. Some counting action was observed. Analysis showed the germane also contained water vapor and oxygen which will have to be removed before good counting action can be expected.

Preparations were started for a trip to Alaska with the shadow shield counter to measure radioactivity in Eskimos.

Radiation Protection Operation was assisted in analyzing body burdens resulting from radiation incidents. In one case, three people were found to have inhaled I-131 and some other isotopes. The Na-24 body burdens of the people at the Recuplex accident were measured and the neutron doses that produced them were estimated. The activity of gold fillings in their teeth was also estimated. Assistance was also given in this incident by placing neutron monitoring equipment at the site of the accident and measuring the radiation levels.

The positive ion accelerator operated satisfactorily during the month.

A shadow shield was used with the precision long counter in order to determine the contribution to its counting rate from room-scattered neutrons. Fair agreement with deductions from inverse-square experiments was found for radioactive neutron sources. The extent of room scattering was found to be essentially independent of orientation with respect to the beam for neutrons produced with the Van de Graaff. Inverse-square experiments with the Van de Graaff are proving difficult because of variation with time in the neutron emission of the targets and because of mutual interference between the counter being tested and the counter used for monitoring. IBM computations were made to explore still another method of analyzing data from inverse-square experiments; no improvement over the present method was found.

A pulse-shape discriminator circuit that is a modification of two described in the literature has proved to be best tested so far. The circuit operates for neutrons as low as 0.8 - 1.0 Mev and there is hope of pushing to still lower energies. Stilbene crystals have given the best discrimination. Anthracene and NE150 plastic respond about the same, but not as well as stilbene. The use of anthracene at liquid nitrogen temperatures is being explored.

The plutonium source we measured earlier was returned to the calorimeter. Its heat output has increased by about one percent. There still remains about a 10% discrepancy between the measured and the calculated expected value of the heat output. The source was shipped to Mound Laboratory to obtain a cross-check on our measurement.

The triennial meeting of the International Commission on Radiological Units and Measurements was attended at Montreux, Switzerland. Major contributions from this laboratory included work on definitions of quantities and units, stopping powers, W values, and neutron source standardization.

Radiation Instruments

Fifty automatic recharging quartz fiber dosimeters of the established design were ordered from an off-site vendor. Considerable investigation was made with a microscope of the charging mechanism for the experimental units. A slight difficulty of occasional random sticking of the gold-plated 10-micron-diameter quartz fiber to the center rod was noted. It appeared that a minute amount of welding might be happening; however, close examination of the gold-plated fiber indicated the gold might be wearing away after continuous, high rate cycling. This effect destroys the conducting property of the fiber and prevents recharging. Because of this, a stainless steel fiber and stainless steel center rod were incorporated into one experimental dosimeter. Initial testing indicates superiority of the steel over the original method at continuous, high-rate cycling. It should be emphasized that the sticking problems generally occur after extensive cycling which would, under normal irradiation conditions, equate to long, continued use of the dosimeters.

A series of subminiature binary circuits and incandescent bulb driver circuits were completed and tested for use in the pocket, signalling, dose meters. In addition, an amplifier, monostable multivibrator, and an astable multivibrator were designed with miniature transistors to drive the binaries and also to drive an incorporated resonant air column alarm. Suitable alarm range switching and reset circuits were also completed in breadboard form. Thus, all necessary circuits for use with the recharging pencil dosimeters in the pocket, signalling, dose meters have been experimentally completed. Fabrication was started on a number of complete printed circuit board units. All circuitry operates from one 1.4-VDC mercury cell.

Further gamma background investigations were made at the Biology animal farm with a NaI crystal and phototube assembly completely surrounded by shielding talc of about one foot thickness. Since the results were similar to previous unsatisfactory tests, the proposed method of counting various radionuclides in hogs under outdoor conditions was temporarily discarded.

All necessary experimental tests were satisfactorily completed on the experimental coincidence-count type alpha counter for standard HAPO plutonium-collection air filters. Fabrication was started on a final system for Radiation Protection Operation.

[redacted]

[redacted]

An experimental transistorized logarithmic response pulse amplifier was completed in breadboard form and is ready for complete testing when time permits.

A prototype experimental low voltage dc to dc converter was completed including fabrication of one unit on a printed circuit board. The converter circuit will be used with rechargeable nickel-cadmium batteries in several experimental portable radiation-detection instruments. The converter tested satisfactorily from -10 F to +130 F with a five milliampere load current with the input voltage varied from 3.8 VDC to 3 VDC as would be expected from the Ni-Cd battery. The output voltage, adjusted as desired from 12 to 15 VDC, varied by less than 20 millivolts for the stated conditions. The one completed converter was installed and tested in an experimental transistorized portable GM instrument. Final testing will be done after the ordered Ni-Cd batteries arrive.

Circuit development and fabrication continued for both the experimental combination logarithmic and linear response and the six-decade logarithmic-response-only area radiation monitors. The original designs have been changed considerably in attempts to meet various HAPO requirements for continuous, alarming, and recording radiation area monitors.

Experiments were started for the detection of minute amounts of $Sr^{90}Y^{90}$ collected from a liquid solution onto filter paper. The detector is a 0.01-inch-thick by 5-inch-diameter sheet of NE-102 (terphenyl-in-polyvinyltoluene). The noise level from the Dumont 6364 phototube was too great to permit proper detection sensitivity, and a low-noise EMI/US-9530-S five-inch phototube was ordered for further experiments.

The general circuit design was nearly completed for the experimental portable mast system for Atmospheric Physics studies of air movements. All necessary commercial instruments were ordered and fabrication was started on portions of the designed circuits.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

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

Work continued on setting up the ion-optic test bench for experimental studies. Construction continued on the ion detector to detect ions by the use of post acceleration of secondary electrons and scintillation detection. The variable 60 kv power supply for these studies was received and put into

operating condition. The electrical system for the ion source was tested for breakdown at 10 kv with a simple ion source installed.

TEST REACTOR OPERATIONS

Operation of the PCTR continued routinely during the month. There were no unscheduled shutdowns.

Two sets of foils were irradiated for the Non-Metallic Material Development Operation.

A series of pressure calibration tests was made during the month to determine the variation of the reactor reactivity air pressure coefficient with the core loading. The pressure coefficient varied from -0.043 ¢/mb to -0.059 ¢/mb for the three lattices tested. The data obtained are tabulated below. All of the lattices were natural uranium-graphite in which only the number of uranium channels and the lattice spacing were changed. The effects of introducing voids into the core were also measured.

TABLE I

Lattice Spacing Inches	No. of Uranium Columns	¢/m bar
7 1/2	9	-0.059
3 3/4	25	-0.050
3 3/4	37	-0.043
7 1/2*	9	-0.066*
3 3/4**	37	-0.045**

* Removed 72 1-3/4" diameter graphite plugs from the core.

** Removed 37 1-3/4" diameter graphite plugs from the core.

Tables of Reactivity vs. Period for PCTR and TTR Experiments

New tables of reactivity vs. period have been generated for use in PCTR and TTR experiments, using the IBM 7090 programs, PERIOD-REACTIVITY. This program was written, and modified for this particular application, by G. F. Bailey of Reactor Design Analysis, NPR Project Section, IPD.

Tables have been prepared for various combinations of the delayed neutron fraction, β, and the prompt lifetime, ℓ. The combinations selected are shown below:

[redacted] [redacted]

1234463

β	ℓ
0.0064	1.5 millisecond
	1.0
	0.6
	0.23
0.0053	1.5
	1.0
	0.6
.0051	0.5

The tables are for U^{235}, Pu^{239}, and U^{233} fueled cores in the PCTR, and for the TTR, which is fueled with U^{235}. The tables will be useful both in doing the experiments, and in insuring compliance with safety specifications which limit absolute reactivity additions.

The range of periods covered in the tables is from 15 to 500 seconds. The time increments are small enough to eliminate the need for interpolation within the tables.

The tabulation of reactivity vs. period for $\beta = 0.0064$ and $\ell = 1.0$ millisecond is given in HW-73457[11] along with tabular values of corrections to the reactivity of the PCTR for changes in the atmospheric pressure or reactor temperature.

Copies of the tables for the other values of β and ℓ are available on a loan basis from the Supervisor, Experimental Reactors Operation.

Calculations in Support of Revised TTR Hazards Report

A solution for the neutron flux in the TTR as a function of time has been obtained for a ramp input of reactivity of three cents per second. The addition of three cents per second represents a more realistic maximum than the value of four cents per second used in the calculation reported last month. The IBM 7090 code HAIREK was used.

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(11) P. F. Nichols, "Tabular Values of Reactivity Versus Period, and Reactivity Corrections for Use in PCTR and TTR Experiments," HW-73457, April 23, 1962.

CUSTOMER WORK

Weather Forecasting and Meteorological Service

There have been considerable activities this month in plant assistance work. Meteorological support and follow-up analyses were made available to the Emergency Control Center in connection with the Criticality Incident. Past analyses and guidance were also made available to RPO in connection with several stack releases to the environment. In addition, there was a review of the CPD Hazard Appraisal for RPO and consultation with Occupational Health on oxides of nitrogen released by FPD.

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	90	84.5
24-Hour General	60	85.2
Special	207	87.4

Precipitation for April was near normal while temperatures and wind speeds were both above normal. There were notably high winds on the 6th, 24th, and 27th.

Instrumentation and Systems Studies

Circuitry and mechanical modifications were made on the Automatic Conveyor Laundry Monitor designed and fabricated for Protective Equipment Decontamination, CE&UO. The alpha probes for the second detection station were partially completed in fabrication although light-shielding difficulties were encountered due to poor quality double-aluminum-coated Mylar which had too many pinholes. The control system for the automatic garment holder and drop mechanism was started. To improve the alpha detection sensitivity, several simple wire frames were devised to provide constant positioning of the coveralls, and with the frames, less than 1000 d/m of Pu^{239} could be easily detected in actual contaminated garment tests. An alternate scheme of using air jets and large, soft rubberized air bag rollers was proposed to substitute for the wire frames, and this idea will be investigated.

Fabrication was nearly completed in the 328 Shop on the detection head assembly for the coincidence-count type alpha air filter counter for counting of standard HAPO 4" x 8" air filters. This system was designed for Radiological Development and Calibrations of Radiation Protection Operation. Similar fabrication continued on the field-use prototype Continuous Coincident-Count Alpha Air Monitor, also designed for Radiation Protection Opera-

[redacted] [redacted]

tion, which will detect and alarm for continuous air-alpha concentrations of 2×10^{-11} μc/cc within 90 minutes.

Fabrication was completed and tests were scheduled for the scintillation, transistorized Automatic Columbia River Radiation Monitor designed for Environmental Studies and Evaluation, RPO.

Assistance and advice was given Radiological Physics concerning the gasoline-engine-powered 1.5-kilowatt electrical generator purchased for use with a gamma spectrometer for investigations of in vivo radionuclides in certain Alaskan natives. Work was satisfactorily completed.

All main instrument fabrication was completed on a transistorized aurally-indicating alpha detection monitor which can be used with a standard HAPO air proportional type alpha probe. Circuit design was started for a special miniature emitter follower circuit to be used with the probe to permit the driving of long cables. All tests were completed on the main instrument which was designed for both Finished Products Chemical Technology and Control Operation, CPD.

A full-time instrument engineer has been assigned to work associated with Physical Metallurgy's Radiation Effects Facility at 105-KW. The data logging system was studied in detail and it was determined that the poor reliability is almost completely due to electro-mechanical stepping switches, relays, micro-positioners, servo motors, etc. The cost and feasibility of modifying the system, incorporating solid state compounds where possible, is now under study. A highly sensitive Leeds and Northrup potentiometer and null detector is used for reading thermocouple voltages. The equipment had ground loops and leakage currents, which rendered the temperature readings inaccurate. This condition was alleviated by installation of guard circuits and proper grounding. The equipment now measures thermocouple voltages to the accuracy of the equipment specifications (± 0.015% - 0.5 μv).

On April 23, one of the expansion capsules at KW was found in the "down" condition (approximately 400°C). The cause was an over-temperature trip in the capsule temperature control loop. Before tripping, the capsule reached a temperature which changed the phase of the test material and ruined the experiment. Upon investigation, the cause was found to be a faulty strip chart recorder-controller in the control loop. The system is now being studied to determine if a more fail-safe design can be incorporated. The stability of one of the temperature control systems is poor (oscillations are present). A dummy capsule is being installed with a

complete temperature control loop. Tests and analysis will be performed on this system in an effort to optimize the response and stabilize the loop.

The contract for the digital voltmeter for the radiation effects facility was awarded to Non-Linear Systems, Inc. for about $11,000. The proposed instrument appears to be of excellent design. The sixty-point Data Logging System is out for bids, with at least four manufacturers active in preparing proposals. Bid closing date is April 30. This system will scan and read out (on typewriter and tape punch) temperature and pressure data from in-reactor test sample capsules. Preliminary development was begun on a solid-state digital timer to initiate the creep capsule data logging cycle. The present Chronolog device has proven unreliable.

Recent changes to the heater arrangement on the FPD 333 Building autoclaves have changed their temperature control characteristics. Recording equipment was installed to measure the step function response of one of the autoclaves operating under the new conditions. The data have been recorded but the analysis is not complete to date.

The reference system for calibrating micro-displacement readout systems, to be used by Physical Metallurgy Operation for in-reactor creep measurements, has been moved to the 326 Building. Work is in progress to reassemble and check out this system, and calibration of the third generation transducer is expected to begin early in May.

Optics

The Groove Depth Microscope has been installed in a hood at 234-5 Building. Tests with the equipment installed confirmed our calibration tests and the customer reports that he is able to repeat readings to within 70 microinches. He also suggested an improvement which makes the measurements easier to take and may increase the accuracy of the instrument. A new illumination and knife edge unit was designed based upon his suggestion. This unit has been fabricated and installed and does indeed promise to increase reading accuracy.

A work order was received to cover the cost of fabricating a similar unit for Lawrence Radiation Laboratory. The design was completely reviewed and modified where experience has shown improvements could be made. This unit is now being fabricated by Tech Shops.

A phone conversation with Mr. Farmer of Dow Chemical Company, Rocky Flats, Denver, Colorado, revealed that Dow Chemical is fabricating three groove depth microscopes for their use. A sketch of the optical layout was sent

to them several months ago.

We are in the midst of the rather "tricky" job of making an ellipsoidal reflecting microscope objective for a high temperature microscope for Ceramic Fuels Operation. The ellipsoid is approximated by five spherical zones during grinding. These zones are blended to a smooth ellipsoid during polishing. The mirror was tested during this past month and found to be under-corrected. The new corrections required to bring it to the desired shape were calculated.

Fabrication of the traverse mechanism to be used in the four-inch I.D. pipes under the 200 Area storage tanks was completed by J. A. Jones. The Optical Shop fabricated the target and image forming mirror. We have just completed final assembly and calibration.

An autocollimator has been designed and fabricated to check the squareness of replacement reactor nozzles. These are made by a new extrusion process which leaves the squareness of machined surfaces in doubt. When these are not square, there is danger of malfunction of fuel charging equipment.

A photometer head for use in molten salt solutions has been designed and is being fabricated for Process Control Development.

A proposal for demonstrating the feasibility of an electrical readout process tube distortion traverse mechanism for NPR was written.

During the four-week period (April 1-April 30) included in this report, a total of 448 man-hours of shop work was done. The work included:

1. Fabrication of components of a radiometer for Physical Measurements, HLO.
2. Fabrication of a quartz connecting link for the LVDT calibration equipment.
3. Repair of two crane periscope heads for Purex.
4. Repair of a borescope for Radiation Testing Operation, IPD.
5. Repair of a film editing device.
6. Fabrication of zone electrophoresis equipment for Radiological Chemistry.
7. Fabrication of an alignment scope for Plant Equipment Engineering, IPD.
8. Fabrication of a lamp adaptor for a borescope for Irradiation Testing, IPD.
9. Fabrication and aluminizing of a hemispherical scintillation crystal for Experimental Nuclear Physics.

10. Fabrication of optical components of the eight-foot-long traverse mechanism for Facilities Engineering Operation, CPD.
11. Repair of B Building periscopes as requested by Facilities Engineering Operation.

Physical Testing

Service testing work proceeded routinely. A total of 4,978 tests were made on 4,928 items, representing some 32,558 feet of material. The greater part of this material continues to consist of tubular components. Test work included: 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 (steam detergent cleaning and vapor degreasing); and ultrasonic (flaw detection and thickness measurements). Work was done for 30 different HAPO components representing most of the operating departments and service organizations, and other AEC contractors. Advice was given on 53 different occasions on general testing theory and applications.

Physical Testing Operation has resumed responsibility for operation of the tube shop at White Bluffs. Work on the NPR pressure tubes was completed on April 9. The only work remaining in question will be the amount of rework on those tubes that may be damaged during installation. Such work will be back-logged and completed on a group basis.

Work has been resumed on the remaining spare PRTR pressure tubes. These tubes will be etched preparatory to autoclaving. The PRTR tubes previously tested, pickled, and autoclaved, were straightened at the C-25 Building and have been shipped to Stores to await installation in the reactor as needed.

All sheath tube testing has been completed. Miscellaneous work was performed on a number of tubing jobs: stainless steel exchanger tubing is to be examined for the nature of discontinuities in the material; ultrasonic thickness measurements were made on some samples of aluminum process tube and Inconel tubes were tested.

Field testing activity consisted of X-ray and gamma-ray work at PRTR (rupture-loop, primary system, and helium system), NPR (instrument leads and test coupons), DeLaval pump impellers, Purex No. 5 spare concentrator, and coolant headers for NPR fabricated by J. A. Jones at North Richland. Fluorescent penetrant tests were conducted on over-bore nozzles, pump

impellers, and a pump casting in the 100-B Area. Ultrasonic thickness measurements were made on a hydrogen peroxide tank in the 100-H Area.

Special assistance was given to IPD maintenance in the examination of splitter blades used for process tube replacement. A series of breakages in the reactor with consequent delays and cost to the tube replacement work, led to the examination of a trial log of used blades. Eight out of thirty were found to contain cracks. All new blades are being inspected.

Considerable effort was expended on the NPR reactor problems in the 300 Area laboratories. A five-foot section of 18" O.D. NPR primary loop pipe was received from General Electric Company, Schenectady, for testing. The pipe was welded by a technique developed at the Schenectady plant. Four sections of the weld were scribed and cut to provide 40 guided bend and 4 macro specimens. Samples were tested and the weld was found to be of acceptable quality. The HAPO weld procedure test piece evaluation was completed with the testing of ten guided side-bend specimens. The weld was found to be of acceptable quality. An additional cycle was completed on the longitudinal crack propagation study of Ladish and Taylor-Forge welds.

A meeting was held with representatives of Testing Methods, FPD, to report what work has been completed on the NPR primary loop 18-inch O.D. piping. Samples were supplied to Testing Methods for further development of ultrasonic inspection methods.

A review of the research and development work on stress analysis with X-ray diffraction and eddy current methods was presented to IPD-NPR representatives.

Side-bend tests and ultrasonic tests of fatigue samples were completed and shipped to 100-D for pressure and vibration fatigue tests. A grain size count was made on two specimens from one of the fatigue failed specimens. The samples were taken about 18 inches apart along the longitudinal weld. Grain measurements were made of the base metal, heat affected zone, and the weld metal. Fourteen test pieces were prepared for a study of heat treatment condition by the eddy current test method. The coupons were heat treated to a variety of conditions to establish eddy current response.

A sample of 2-5/8" O.D. NPR connector piping has been procured and submitted to Tech Shops for fabrication of an ultrasonic standard. Ultrasonic examinations will be used in conjunction with other methods of inspection to evaluate the quality of the piping.

Four carbon steel pins and three carbon steel nuts, representative samples of Gray-lock connectors for use in the NPR, were nondestructively and destructively examined. Magnetic particle inspections, ultrasonic examinations, hardness tests, fracture tests, bow measurements, heat treatment,

and metallographic methods were utilized to evaluate the samples.

Preliminary development began on methods for nondestructively and destructively determining the cladding thickness on approximately 35 miles of nickel clad copper tubing to be used for conductor sheathing in the NPR. The possibility of utilizing eddy current techniques for nondestructive spot checks and random sampling with destructive measurements was suggested.

Extensive test work has been done on both NPR venturi inlet connectors and connector piping. Primary emphasis has been on magnetic particle and flattening tests to determine as-received quality. Additional tensile and impact specimens were prepared and shipped off-site for testing.

An ultrasonic test was developed for examination of approximately 50 feet of 3-1/2-inch O.D. stainless steel seamless piping located in the high pressure thermal hydraulic testing system at 189-D Building. Examination of the piping revealed one section of the vertical riser to contain an indication of rejectable magnitude. The section was removed from the system, visually examined, radiographed, fluorescent penetrant tested, and metallographically examined. No apparent cause for the ultrasonic indication was detected by visual examination, radiography, or fluorescent penetrant tests; however, metallographic examination disclosed a discontinuity of sufficient magnitude to enable detection by ultrasound. As a result of the examinations, inspection of 4-1/2-inch O.D. piping also located in the system was requested. A section of the 4-1/2-inch O.D. piping was submitted to Tech Shops for fabrication of an ultrasonic standard. An ultrasonic test will be developed and ready for application when the current hydraulic tests are completed and the system shut down and access to the headers can be obtained.

Ultrasonic testing was performed on the Purex concentrator tank in the 200-E shops. A total of 38 indications, exceeding the standard, were found on the circumferential bottom weld. Correlation of the ultrasonic indications with radiographic evidence of discontinuities was good.

Assistance was given in calibrating and measuring the accuracy of Beckman meter heads in the 234-5 Building.

Additional development work on slit radiographic techniques for the examination of I&E fuel elements continued with fabrication of a circular disk to provide complete contact with the male end of the fuel element. A 30-mil-wide slit that goes to the center of the disk is used. The disk provides more control of scattered radiation, but requires such precise alignment with the central beam that production use of the technique is

impracticable. Films of different speeds and different rotational rates were also tried.

Work continued on use of autoradiography to demonstrate the efficiency of incremental loading of fine PuO_2, fine UO_2, and coarse UO_2 into fuel rods. Results showed gross segregation and prompted investigation of other fuel loadings with questionable uniformity. Various methods for fixing film to the fuel elements were tried. The best method consisted of molding 0.040-inch lead strips around the film and rod. This method was fast and economical. The lead foil intensified the exposure of the film and cut down the exposure period about 25%.

Gamma-ray radiographic techniques were developed for resolving thermocouples in the center of zirconium-clad UO_2 fuel rods.

Deformation tests were made on FPD water mix spools to determine maximum deformation at a minimum load of 32,000 lbs. Load deformation curves were furnished for each of 40 spools and 22 fuel elements.

Development and fabrication continued on the mechanical components for the ultrasonic test which is being developed to detect cracks in installed reactor process tubes.

Development of an ultrasonic unbond test for fission product transient fuel capsules continued. Test standards were readily detected with a two-crystal transmission method. Fabrication of capsules for testing is proceeding.

ANALOG COMPUTER FACILITY OPERATION

The major analog computer problems considered during the month were:

1. NPR Simulator
2. Reactor Instrumentation Studies
3. "C" Column Simulation

Two classes on analog computation were presented to members of Operations Research and Synthesis Operation and of FPD. A presentation on analog computation was given to members of the class taking Fundamentals of Data Processing.

Techniques were developed and demonstrated for using the analog computer to solve optimization problems. A function of two variables can be plotted automatically in a three-dimensional perspective presentation on graph paper which allows one to pick out the extremum points. The technique is

considered highly valuable for determining optimum values of system parameters and for model fitting applications. A presentation of the FPD quality control model on graph paper was found to be simple and fast.

Eighty percent of the GEDA equipment and ninety-four percent of the EASE equipment was in operable condition during the month. Computer utilization was as follows:

GEDA	EASE	
21	133	Hours Up
98	21	Hours Scheduled Downtime
21	0	Hours Unscheduled Downtime
28	14	Hours Idle
168	168	Hours Total

A complete check was made on all GEDA integrators, amplifiers, and function generators to determine their operability. A period of over a week was spent in correcting the troubles found at this time. To place the computer in proper operating condition, a very extensive step-by-step overhaul must be done. The installation of a remote control hookup for the old Brush recorder has been completed. The EASE computer has had trouble with the reference power supply several times in the last two weeks. A new maintenance routine for the EASE power supplies is now being used. Each power supply is now being removed from the computer once every six weeks to be completely overhauled on the bench while a spare is placed in its location in the computer.

INSTRUMENT EVALUATION

Following one year of continuous charge and discharge cycles on 250 ma-hr Ni-Cd batteries, a number of larger 2.5 and 4 amp-hr batteries were ordered for evaluation and field use in a redesigned transistorized GM portable instrument.

Tests on particular commercial 0-10 millivolt chart recorders have indicated excessive noise and dead-band problems. The difficulties became apparent during long-term recording of diode oven noise tests on a combination logarithmic and linear response experimental area radiation monitor.

The field tests at Redox and Purex continued successfully on the prototype combined alpha, beta, gamma scintillation transistorized hand and shoe monitor. After five continuous months of operation with only a few probe light-leak repairs (a simple task) necessary, the instrument was

scheduled for return to the 300 Area for tests in the 325 and 327 Buildings. The standard HAPO drawings and specifications have been completed in a cooperative effort with RPO and four of the instruments have been ordered from an off-site fabricator.

Several brands of double-aluminized Mylar film (0.9 mg/cm^2) were tested and some of the material was found to be useless for scintillation probes due to excessive pin holes and scratches. Of representative samples from three manufacturers, only one proved to be acceptable. We advise that all double-aluminized Mylar be visually inspected with a strong light before application.

One vendor prototype Model II Scintran, of 65 to be fabricated to Nucleonic Instrumentation design off-site, was received and partially tested. The general performance of the instrument and the alpha scintillation probe was very good. Only a very few minor changes need to be made before fabrication can be started on the remaining 64 units.

The electronic circuit drawings were nearly completed for the prototype transistorized scintillation alpha portable "poppy", and evaluation tests were nearly completed on the large area beta-gamma air light pipe scintillation probes for the beta-gamma hand and shoe monitor.

Following a previous evaluation rejection of 60 out of 100 commercial pencil-type dosimeters of the 0-600 r type, the manufacturer re-submitted 60 new units. Fifty-eight of these were found to be acceptable. These dosimeters were heavily used during the 234-5 Building incident problem.

Paul F. Gast

Manager
PHYSICS AND INSTRUMENT RESEARCH
AND DEVELOPMENT

PF Gast:mcs

CHEMICAL RESEARCH AND DEVELOPMENT OPERATION

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Alum Production Studies

Water treatment at the IPD reactors includes (in part) control of the pH of the water and the addition of alum (aluminum sulfate) as a clarification step. The pH reduction for corrosion prevention may be controlled by either free sulfuric acid addition or by the addition of alum which hydrolyzes during water treatment.

Currently, several of the IPD plants are manufacturing aluminum sulfate for water treatment in a continuous flow process by reacting crude bauxite with a blend of water and an excess of 93 w/o H_2SO_4. Analysis of the crude bauxite by dissolution in 60 w/o H_2SO_4 gave 55.3 w/o available Al_2O_3 and 15.8 w/o insoluble material, largely hydrated aluminum silicates.

The water is added to the acid to attain the necessary reaction temperature by heat of mixing and to prevent aluminum sulfate precipitation, although the reaction once started is exothermic and proceeds to the boiling point generating considerable steam and foaming violently. A series of studies to collect more process information on the operation of present plants as well as to aid in the design of future apparatus has been concluded. Highlights of the studies follow.

In well-stirred vessels conversions of available Al_2O_3 to alum as high as 92 percent were obtained with five to eight pounds of bauxite per gallon of 93 w/o acid, one gallon water per gallon acid, and apparent (no allowance for foaming) liquid holdup times of three minutes, at operating temperatures of 130-135 C. Operation with poorer agitation under these conditions gave only 60 to 90 percent conversion, reaction temperatures as high as 160 C and greater accumulation of unreacted solids in the reactor.

The conversion of Al_2O_3 is not a strong function of holdup time in the range two to ten minutes or of bauxite-to-acid ratio at ratios below eight pounds of bauxite per gallon of acid. It is a strong function of water-to-acid ratio, however. Poor

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conversions and a greater dependency on holdup time were obtained with 1.5 and 2.0 gallons of water per gallon of acid than with 1.0 gallons of water per gallon of acid.

Two reaction vessels in series performed little better than a single stirred vessel. A column-type agitated cocurrent flow reactor gave at least as good conversion as the stirred vessel, although difficulty with plugging of inlet bauxite slurry lines near the head tank prevented steady state operation.

New Production Reactor Crib

The expected release rates of fission products from the NPR crib to the Columbia River, resulting from the discharge of diverted coolant containing rupture debris, was estimated by use of a two-dimensional model.

Except for I-131 and Sr-90 the estimated amounts of all other radiologically-significant fission products reaching the Columbia River as a result of discharging rupture debris to the NPR crib are negligible.

The maximum rate of release of I-131 will occur about one month after cribbing and peak about three hours after the first trace of waste enters the river. Based on the assumption that only one percent of the total I-131 released is soluble, the amount of I-131 entering the river can be calculated. For a "usual" rupture, expected release of 50 g uranium, the maximum concentration in the river at Ringold will not exceed 2.7×10^{-6} MPC (thyroid) or 9.0×10^{-3} NPR disposal criteria (3×10^{-4} MPC). Moreover, the concentration of I-131 in the river will not exceed NPR disposal criteria (thyroid) unless the loss of uranium per failure exceeds 5000 g.

The time required for the first traces of Sr-90 to reach the river after cribbing is estimated to be 10 years. If all of the Sr-90 released is assumed to be soluble, a maximum river concentration 0.23 and 23 times the NPR disposal criteria (bone) for 50 and 5000 g of uranium, respectively, will be reached within hours after the first trace of Sr-90 reaches the river and will decrease slowly for the next year or until supplemented by Sr-90 from subsequent ruptures.

It should be emphasized that all of the above estimates were based on flow quantities derived from a two-dimensional crib model and that pessimistic conditions and assumptions were selected. Ionic dispersion, a phenomenon which would minimize peak release rates,

was ignored. However, until crib outflow is derived in three-dimensions and the phenomenon of ionic dispersion is evaluated, more practical estimates cannot be provided.

Ground Water Temperature Studies

The gradient of the water table beneath the 100 Areas was at its yearly maximum slope during the past month. The rate of ground water movement toward the river also reached a maximum during this same period. A gradual increase of from 20 to 40 feet per day in the rate of ground water movement was observed at 100-B Area over the past six months, based on isotopic analyses of water samples obtained in this area. The rate of movement is expected to decrease with rising river levels and lower water table gradients. The temperature profiles of the 100 Area monitoring wells should also soon decrease since the temperature of the river is now about 10 C colder than that of the ground water.

Analyses of raw intake water collected from the pumps in the 181-B Building show concentrations of Cr-51 up to 3×10^{-6} μc/ml. Previously, the highest detected concentration of Cr-51 at this sampling location was 9×10^{-7} μc/ml. The current result is in agreement with Cr-51 and gross beta analyses of 200 Area water which is pumped from the 181-B facility.

Airborne Particulates in Reactor Operations

Sampling and analysis of airborne particulates during tube replacement operations continued. Enough data will shortly be available to permit completion of this study. Iodine-131 was detected in samples collected at 100-B but was believed to be from an unusual I-131 release from a separations area stack.

Effluent Monitoring

The As-76 prototype monitor was moved to the 146-FR Building, 100-F Area. The only mechanical or electronic difficulty experienced, since the monitor was placed in service on April 12, was caused by a loose rectifier tube in the amplifier. However, some difficulties were met in the sampling and detector systems.

Contamination build-up caused the background to double in the detector system cell over two weeks of operation. The background was reduced significantly by decontaminating the cell with water and detergent. The sample results did not change, which indicated the background build-up was not affecting the As-76

analysis. Of more importance is the fact that I-135 interference of the As-76 count was experienced for the first time. The presence of I-135, a fission product, indicated a possible leaking fuel element. The fact that the monitor detected the additional radioactivity might be applied to rupture detection. Useful application for rupture detection or ways to eliminate I-135 interference are being sought.

Calibrations were made with As-76 spikes. A value of 0.011 μμc As-76/ml/count was obtained as a conversion factor. This factor applies to a 7.5 minute count, a 274 ml sample volume and a spectrometer channel width of 10 percent.

Uranium Oxidation and Fission Product Volatilization Studies

The draft of a report, "Fission Product Release from Uranium - The Effect of Irradiation Level," was completed. An explanation was developed for the results obtained in the experimental work. Formation of fission gas bubbles at high irradiation levels appears to play a key role in the release of volatile elements.

SEPARATIONS PROCESSES

Disposal to Ground

Preparations are being made to core sample the Redox 108-SX waste tank sludge. An equipment set-up which is essentially a duplicate of that used in sampling the Purex 103-A tank sludge will be used.

Results of four soil column tests to evaluate the soil uptake of strontium from Redox 0-2 decant waste (276-S crib) were received from Redox Analytical Control Operation. Thirteen column volumes of typical waste were passed through the columns without detectable breakthrough of a Sr-85 tracer. To date the crib has received two column volumes of waste, and it now receives waste at a low rate of approximately 0.01 column volume per year. These test results and waste disposal data indicate that a replacement crib will not be needed for a number of years.

Iodine-131 Sampling in Gaseous Effluents

The possibility that attendant constituents in plant effluents may reduce the efficiency of charcoal for adsorbing I-131 has been indicated in recent tests. Lower than predicted I-131 retention in stack sampler charcoal capsules and for an oxidizer off-gas charcoal trap have been reported. Several experiments were

completed this month to determine the effect of some possible constituents on the I-131 adsorption and retention on charcoal. In these experiments the small stack sampler capsules were used as the test charcoal, with a linear flow rate of carrier or flush gas of about 100 feet per minute. Iodine in the laboratory tests was generated by heating carrier-free I-131 to 300 C. Conclusions reached from these studies are that:

1. The charcoal capsules are intrinsically efficient for I-131 generated in the laboratory either when the carrier gas is air, air containing NO_2, or air containing hexone.

2. Hexone and NO_2 in air do not remove I-131 adsorbed on charcoal.

3. I-131 in stack gas is adsorbed much less efficiently than I-131 generated in the laboratory.

The lowered efficiency of charcoal for I-131 adsorption in process streams was not explained, and further study is planned.

A report was drafted which describes the study of the efficiency of stack sampling systems for I-131.

Purex Process

Because of current difficulties with flooding in the 2D-2E Purex column system, transfer rates were determined for the extraction of uranium from synthetic 2D feed-scrub solution into plant 2DX(R7). For comparison, similar transfer rates were determined for extraction into first cycle solvent (G5), and into make-up solvent (R1A). Before use, all organics were treated with a typical second cycle solvent wash--carbonate solution followed by water.

The results showed that extraction into the R7 was significantly lower than into either of the other two solvents. There was little difference between G5 and R1A. Stripping experiments showed almost identical transfer rates for all three solvents, bearing out the quite well established fact that the extraction rate is a more meaningful parameter than the stripping rate (or distribution coefficient) in assessing solvent quality of this type. The R7 also showed poor disengaging behavior, which likewise indicates generally poorer quality than the comparison samples.

Samples of both G5 and R7 were subjected to carbonate-permanganate treatment followed by water washing. The R7 formed almost stable emulsions in the latter step and an additional carbonate wash was

used with somewhat better disengaging of the phases. The extraction rate studies were than repeated on both materials. The results showed very little change for the G5 sample whereas R7 was markedly improved. Graphical analysis showed that a two-fold increase in the first order rate constant had been effected by permanganate washing the R7 sample.

Purex Solvent Extraction Studies

Pilot plant studies to characterize the current Purex plant 2D and 2E column instability and inefficiency problems were continued during the month. Tests of Purex plant demineralized water, 2DX, 2EX and 2DU (2EF) demonstrated that the major contributor appeared to be the 2DU stream, although all of these streams contributed in some degree to the 2E column instability problem. This fact, together with the absence of any pronounced adverse effect of using the plant 2DX in the 2D column, suggests that the primary "do-bad" enters with the 2DF stream (a stream not tested in the "cold" pilot plant) and follows the organic stream. It should be noted, however, that these conclusions are based on use of plant organic streams collected when the plant performance was not extremely bad.

A comparison of the 2E column performance with various combinations of pilot plant and Purex streams is shown in the following table. Rates were held constant at an L/V of 1, a total flow of 1000 gph/ft^2, and temperatures of 50 to 55 C.

Source of Stream	Instability Frequency Cycles/Min.	Color Line Ft. from Bottom	HTU, Ft.
Pilot Plant 2DU and 2EX	62 ± 5	7 - 9	1.2*
Pilot Plant 2DU; Purex 2EX	49 ± 2	~ 9	1.3-1.5*
Purex 2DU; Pilot Plant 2EX	47 ± 3	>10	1.8
Purex 2DU; Purex 2EX	42 ± 2	>10	2.2
2DU made in Pilot Plant from Purex 2DX; Pilot Plant 2EX	53 ± 2	~ 9	1.3-1.5*

*Estimated from color line position

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A superior "sandwich-type" cartridge was developed for the 2E column to enable the plant to operate at higher capacities and under more efficient pulsing conditions. Essentially this is a graded 23 percent free area nozzle plate cartridge with 3/4-inch thick clusters of linear polyethylene plates inserted. The recommended nozzle plate spacing was two inches in the top six feet and three inches in the remainder; the plastic plate spacing was 7-1/2 inches in the bottom six feet, 10-1/2 inches in the middle six feet, and 13-1/2 inches in the top six feet. This cartridge was demonstrated to operate satisfactorily under the above conditions at 72 cycles/min using Plant 2DU and 2KX. The optimum HTU was about 1.1 ft. (waste loss was 0.004 percent using a 15 ft. tall cartridge).

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 objective of the recent studies was to minimize the amount of inert salts in the resultant uranium(IV) nitrate solution. The majority of the runs were made on a 500 to 2000 ml scale. One run was made on a 150 liter scale.

Highlights of the findings are:

1. Uranium(IV) nitrate can be formed by the addition of aluminum dust to an acidic uranyl nitrate solution containing either sulfamic acid or hydrazine as the holding reductant. No inert salts such as aluminum nitrate are needed.

2. In order to prevent the formation of the black precipitate described last month (cf. HW-73202 C), the quantity of aluminum dust added must be less than that needed to reduce all of the uranyl nitrate and the final solution must contain free nitric acid.

3. With hydrazine, yields of uranium(IV) nitrate as high as 2.6 moles per mole of aluminum nitrate were attained.

4. In general, the yields increased with a decrease in the mole ratio of aluminum added to uranyl ion.

5. Although the use of an inert gas blanket increases the uranium (IV) nitrate yield, it is not a necessary condition for the reactor.

6. At the high uranium(IV)-to-aluminum ratios, 40 to 60 percent of the initial uranyl nitrate is reduced.

7. Solutions as high as 0.9 molar in uranium(IV) nitrate were produced.

8. Generally, the reaction is complete in about four hours if the solution is initially heated to 50 C.

9. The reaction is exothermic. Without cooling, a solution initially at 50 C will rise in temperature 0 to 50 C depending upon the original uranyl nitrate concentration.

10. The reoxidation rates of uranium(IV) nitrate in contact with air were determined. First order reaction constants of 0.03 day^{-1} and 0.12 day^{-1} were measured with hydrazine and sulfamic acid, respectively, as the holding reductant.

Although the uranium(IV) nitrate so produced contains dissolved aluminum, the weight of the aluminum is but 1/40 the weight of iron(II) sulfamate contained in the conventional reductant solution of the same reducing potential.

Ion Exchange Contactor Studies

The addition of a plug valve between the elution and scrub sections of the Jiggler has substantially reduced undesirable back-mixing. An experiment using a nitric acid and thorium feed solution and 50-100 mesh Dowex-1 ion exchanger was conducted at an operating temperature of 25 C. Results from this manually-operated run indicate very satisfactory absorption and elution. Pertinent data for a 2.7 ft. absorption and 3 ft. elution section are:

Stream	Flow Rate mls/min.	HNO_3 (M)	Th g/l	$NaNO_3$ (M)
XAF	723	7	5.1	2.0
XAS	210	7	0	0
XAW	933	7	0.008	-
XCX	100	-	-	-
XCP	210	7 to 0.5	14.4	-

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These values are averages from four ten-minute cycles after attainment of steady state. Analyses of resin loading and sodium are not yet complete. The $NaNO_3$ is being used for study of effectiveness of scrubbing.

Resin Stress Studies

An unexplained variation (0.048 to 0.220) in the coefficient of friction, μ', (cf. HW-72902 C) obtained for 10-20 mesh resin as opposed to an almost constant value of the same parameter (0.048) for 20-40 mesh resin led to the use of an oscillograph to measure values under transient conditions. This instrument can record simultaneously signals from both the vertical and radial stress transducers. In addition, its galvanometer-light beam arrangement has greater sensitivity to the rapid changes of force resulting from continuous resin particle re-arrangement in the experimental bed than the conventional amplifier-recorder previously used.

The variation of the coefficient of friction for the 10-20 mesh resin appears to be closely related to its history of nitration. A consistently low value is obtained with resin that has very recently been nitrated, whereas a consistently high value is obtained for resin that has been in contact with either distilled or process water for over two weeks. In one case even a 24-hour residence in distilled water resulted in a marked increase in the coefficient.

The average vertical stress for any one series of runs can be readily predicted mathematically from the variation in radial stress versus bed height data and a force balance made on the resin bed.

Plutonium Recovery from RMC Line - Continuous Task I Oxalate Supernate

The 234-5 RMC line flowsheet employs continuous precipitation of plutonium from 2.7 to 3.1 molar nitric acid by addition of one molar oxalic acid. The plutonium oxalate precipitate is removed by continuous mechanical filtration and the supernate, which contains 0.2 to 1.0 g/l plutonium, is concentrated by evaporation and the plutonium recovered by solvent extraction at Recuplex. Replacement with a simple ion-exchange process would not only accomplish both concentration and recovery but would also render the button line independent of Recuplex. Several very promising anion-exchange runs, with actual RMC line supernate acidified to 7 M HNO_3, were reported last month. The feed employed in those

runs contained a precipitate which was subsequently found to contain most of the plutonium, resulting in ion-exchange feed concentrations of about 0.1 g/l vice one g/l. It has since been found that increasing the nitric acid concentration to 8.6 M dissolves the precipitate. Two additional column runs have been made at this acidity. These runs gave even higher resin loadings (30 to 40 grams of plutonium per liter of resin) than the former, although the 20 grams/liter obtained at 7 M nitric was sufficiently high to make the process feasible. The increased loading at the higher acidity appears to be due both to the higher feed plutonium concentration and to a further suppression in the ionization of oxalic acid. In this regard, it was observed that the color of the feed changed to a much more intense green as the nitric acid concentration was increased from 7 M to 8.6 M.

The scheme proposed for 234-5 installation would employ three critically safe columns operated in a cyclic manner, i.e., two would be loaded in series while the third was being eluted. This mode of operation would combine the advantages of near-continuous operation, high product concentration (up to 40 g/l), and very low waste losses.

WASTE TREATMENT

Cesium Removal, Purification and Concentration from Purex Formaldehyde Treated Waste

Work continued on a flowsheet for separation and purification of cesium from Purex formaldehyde-treated waste, using mineral columns. Emphasis was shifted from ammonium acetate to ammonium carbonate solution for elution of cesium from clinoptilolite. The carbonate is preferred because it can be removed below 100 C, leaving an aqueous solution of cesium carbonate. Elution of cesium from clinoptilolite with ammonium acetate or carbonate is greatly enhanced by employing elevated temperatures. Elution is 98 percent complete at 8 column volumes for either 2 N ammonium acetate at 80 C or 4 N ammonium carbonate at 60 C. Higher temperatures are not practical for ammonium carbonate because of gas evolution due to decomposition. The cesium from this solution can then be loaded on an inorganic zeolite column for storage and/or shipment. The zeolite in the second column should have a high cesium capacity but not necessarily a high selectivity. Several alumino-silicate materials are being examined for this purpose.

The two synthetic zeolites, Zeolon and Linde AW-500 on which cesium kinetics and capacities were reported last month, have been

examined for cesium selectivity. Synthetic neutralized waste supernatant solution was used as the cesium feed stock. The column volumes to 50 percent breakthrough were 29, 43 and 21 for Zeolon, AW-500 and clinoptilolite, respectively. Since the kinetics of AW-500 are also superior to the other materials tested, its use in alkaline waste treatment appears most promising.

Elution of Cesium from Clinoptilolite after Heating

Columns of clinoptilolite, loaded with Cs-137 from a synthetic formaldehyde-treated waste, were heated different lengths of time and then eluted with ammonium nitrate. The percent eluted for the experimental conditions are summarized in the following table:

Temperature, C	Time Heated, hr.	Percent Eluted at 20 Col. Vol.
400	2.5	94.5
400	48	90.5
400	150	92.5
200	48	92
200	150	99

The percent Cs-137 eluted remained essentially constant after 20 column volumes except for the first column from which elution continued until 96 percent was eluted at 60 column volumes.

Recovery of Neptunium(IV) from Purex Formaldehyde Treated Waste with D2EHPA

Studies of the feasibility of recovering Np from Purex plant FTW by extraction into D2EHPA-TBP-Diluent solvents were continued. As previously reported, greater than 97 percent extraction of neptunium(IV) was obtained in a single batch contact of equal volumes of 1965 version FTW with 0.04 M D2EHPA-0.02 M TBP-Shell Spray Base. In agreement with ORNL experience, extraction of Np(IV) from synthetic FTW was found to be inversely proportional to $[H+]^2$ at acidities above 1-2 M and directly proportional to $[H+]$ at acidities below 1-2 M. However, even at acidities as high as 4.5 M, 96 percent of the Np(IV) was extracted in a single batch contact with an equal volume of 0.04 M D2EHPA-0.02 M TBP-SSB. Contact of this organic with an equal volume of 0.5 M oxalic acid removed over 97 percent of the Np(IV). Oxalic acid stripping has the advantage over stripping with NaOH in that stripped Np and Fe are not precipitated.

Solvent Extraction Removal of Fission Products from Purex FTW

Tracer-level studies to develop a D2EHPA solvent extraction flowsheet for separating strontium and rare earths from Purex FTW were continued. Mixer settler runs were made with HEDTA-complexed, acetate-buffered FTW (1965 composition) feed at pH 4 and solvent containing 0.3 M D2EHPA-0.15 M TBP-Soltrol 170. At an ℓ/v of 0.5 and with seven extraction stages, strontium and cerium recoveries were greater than 99 percent and promethium recovery was about 80 percent. Excellent decontamination from iron, chromium, ruthenium and zirconium-niobium were obtained. Cesium and sodium DF's were 26 and 10, respectively. Simulated extraction feed solutions (1965 FTW made 0.19 M in HEDTA and 0.35 M in citrate) traced with Sr-85 and Ce-144 were subjected to Co-60 radiation to a total dose of 7×10^7 R. The distribution ratio for Sr-85 between the irradiated solution and 0.3 M D2EHPA-0.15 M TBP-Shell Spray Base was 21 compared to 30 with a non-irradiated control. Similarly, the K_d for Ce-144 was about 320 compared to 700 with the control. The decreased K_d's are believed to result from increased solvent loading with inert iron and chromium (cf. HW-73202 C) rather than from effects of decomposition products of HEDTA.

Solidification of Low-Level Wastes

Difficulties in obtaining nondeliquescent solids from highly alkaline wastes, such as coating removal waste, by concentration with hot air or air-CO_2 mixtures prompted study of the formation of "concretes" from such wastes. Synthetic coating removal waste was concentrated by a factor of three and mixed with Portland cement or plaster of Paris. About 0.5 grams of plaster of Paris per ml of concentrated waste was required to produce a rock-hard solid; about one gram of Portland cement per ml of concentrated waste was required. Plaster of Paris has the further advantage that it does not set to a solid for several hours at these proportions while Portland cement sets rapidly. Plaster of Paris slurries poured into a tank would have time to flow before setting.

Product Forms for High Level Wastes

Bench-scale experiments were started to study ways of incorporating 4 x 8 beads of synthetic alumino-silicate zeolite in a matrix of material of high thermal conductivity. Synthetic zeolites have been reported to have attractive ion exchange capacity for use in packaging Sr-90 and Cs-137, but has the unattractive feature of low thermal conductivity. The effective thermal conductivities of the beads in air and in helium at 200 F were found to be 0.1 and

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and 0.15 Btu/hr-ft-°F. At typical Sr-90 loadings, internal temperatures of the order of 1700 F could be expected in a 12 inch diameter cylinder of the material (with helium). This temperature could be reduced to 880 F if the effective thermal conductivity were increased to 1.0.

A two-inch diameter by 8-inch long cylinder of the beads was successfully cast in a Wood's metal matrix by pouring the molten metal into a restrained bed of zeolite beads. Attempts to repeat the experiment with molten aluminum were less successful because of the high surface tension of molten aluminum, and the aluminum oxide which formed in air at the melting point, interfered with the flow. Addition of bismuth did not lower the surface tension enough to help. Additional studies are planned with other alloys and molten salts.

In other studies with synthetic zeolites, Linde 4-A and 13-X were found to shrink at 800 C; at 950 C the shrinkage was to 50 percent of the original volume.

The recently discovered fluxing effect of $Sr(OH)_2$ on SrO casts some doubt on previously reported findings concerning the fluxing power of lithium fluoride. The partial decomposition of $SrO_2.8H_2O$ at low temperature forms $Sr(OH)_2$, which fluxes SrO at 550 C to a dense melt, but further decomposes at 800 C (or by radiolysis) to SrO and H_2O. Although lithium fluoride has proved to be a fluxing agent for SrO, the quantitative effect is not yet known. A fluxed cake of SrO, $Sr(OH)_2$, and LiF that melted at 500 C became solid when heated to 800 C, the decomposition temperature of $Sr(OH)_2$.

In other packaging studies, cesium alum, $Cs_2Al_2(SO_4)_4.24\ H_2O$ was found to lose water up to 300 C. It is then stable up to 650 C but suffers a 25 percent weight loss on further heating to 900 C. Cesium oxalate, $CsH_3(C_2O_4)_2$, decomposes at 250-300 C and above 500 C loses weight slowly by volatilization of Cs_2O. Molybdic acid and lead silicate are two fluxing agents that show promise for consolidating cesium 12-tungsto-phosphate.

Remote Welding

Conditions suitable for welding fission product containers with the tungsten inert gas welder using argon as shield gas were determined and 38 samples were welded. No leaks could be detected in any of the samples with the mass spectrometer leak detector.

Ten samples were welded using helium as the shield gas and a leak was detected in one of the samples. Solutions to other problems are required before helium can be considered as a candidate for shield gas.

A spring loaded swinging arm torch positioner was used to weld all 48 samples. The operation of it is superior to the former air cylinder positioner.

TRANSURANIC ELEMENT AND FISSION PRODUCT RECOVERY

Assistance to Purex Head-End Fission Product Recovery

A large number of laboratory experiments (with tracer-level, synthetic solutions) were performed to (1) help define optimum oxalate precipitation conditions for the separation of cerium from strontium, and (2) to elucidate the effect of hydrogen peroxide on the sulfate precipitation of rare earths from FTW. Significant findings follow.

In the present Purex strontium flowsheet, the carbonate product is dissolved in nitric acid and cerium is then precipitated as the oxalate, leaving the strontium in the supernate. Under strongly acid conditions (> 1 M HNO_3), plant experience shows that over 95 percent of the strontium stays in solution, but the cerium decontamination is not as good as desired. At lower acidities, cerium decontamination is very good, but strontium loss is excessive. Experiments were designed to explore in detail the effects of pH, sulfate, lead and oxalate on the oxalate step. Cerium precipitation was found to be affected by pH and oxalate concentration but not by lead or sulfate. Also, neither sulfate nor lead had an adverse effect on strontium behavior under the conditions examined. In fact, there was a consistent correlation of decreased strontium precipitation with increased lead additions, an effect which is contrary to that expected (and which will be studied further). In all cases, cerium decontamination factors exceeded 10, and often 20, while 90 to 100 percent of the strontium remained in the supernate--a separation which would be quite satisfactory if it could be duplicated in plant equipment.

It was recently observed that hydrogen-peroxide inhibits the precipitation of rare earth sulfates. This unexpected phenomenon was studied further since it not only has potentially adverse implications for Purex rare earth recovery but also might conceivably afford an alternate way to separate rare earths from strontium. Using tartrate-complexed FTW and the standard lead-carrier sulfate

precipitation scheme, over 90 percent of both cerium and promethium were precipitated over a pH range from 1 to 2.5. However, with addition of 5 ml of 30 percent hydrogen peroxide per 100 ml of FTW, virtually none of the rare earths were precipitated at pH 2 or 2.5. At pH 1.5, about half of the promethium and one-fourth of the cerium remained in solution. At pH 1.0, over 90 percent of both elements precipitated. Additional experiments with varying amounts of peroxide are planned.

Path of Technetium in the Purex Plant

Analyses have been completed for technetium-99 in a series of Purex plant samples. These support the following preliminary conclusions:

1. About 20 percent of the technetium continues to leave the plant with the uranium product.

2. Less than one percent follows the plutonium.

3. About 70 to 80 percent leaves the plant with the HAW.

4. About 10 percent recycles via the 3 WB stream. Of this 10 percent recycle, 8 percent is contributed by the IDW and 2 percent by the 2AW.

5. Of the technetium reaching the partition cycle, over 90 percent follows the uranium into the IBU stream.

Further sampling will be carried out to confirm this data and to determine how the path of technetium varies with time and operating conditions. Additional Redox samples will also be analyzed as soon as available.

A-Cell Decontamination

A-Cell decontamination work has been completed. No major difficulties were encountered. Repairs to building equipment are presently being made. A-Cell will be ready for equipment installation by April 30.

Cesium Solvent Extraction

Laboratory study of the solvent extraction of cesium from Hanford waste solutions continued. Major items studied and the results obtained included the following:

1. 2,4,6 trinitro - 2' - trifluoro-methyl diphenylamine, 2,2',4,6 tetranitro diphenylamine, trinitro diphenylamine, and trinitro-dichloro diphenylamine were compared with dipicrylamine in somewhat more detail than reported last month. Extraction of cesium from either "as received" (pH 10) 103A supernate or from FTW (tartrate complexed and adjusted to ~pH 8.5) was negligible with all except dipicrylamine, which gave cesium E_a^o values of 4 and 23, respectively, from the two wastes. However, addition of 0.1 or 0.2 M additional NaOH resulted in very nearly comparable cesium extraction with all of the extractants.

2. A series of other nitroparaffins was compared with nitro-benzene as diluents for dipicrylamine. These included nitro-toluene, 1-nitropropane, 2-nitropropane, nitrocyclohexane, o-nitroanisole, and 2-nitrobutane. None gave quite as high extraction coefficients as nitrobenzene; however, all were in the same "ball park" and any one of these solvents could probably be used successfully. Nitrobenzene has the highest flash point (190 F) of any of these solvents for which this data was available.

3. Irradiation stability measurements in the cobalt-60 source indicated that the "half life" of dipicrylamine is about 4×10^7 R, a stability more than adequate for the intended application but low enough to insure that any dipicrylamine which found its way to the waste tanks would soon be destroyed.

4. The stripping, or back-extraction, of cesium from DPA-nitro-benzene solutions was studied as a function of nitric acid and dipicrylamine concentration. The value of E_a^o ranged from about 0.1 to 0.01 as the acidity increased from 0.1 M to 1 M and was not strongly dependent on DPA concentration.

5. Dodecylphenol and o-phenylphenol were tried as cesium extractants, using both zylene and nitrobenzene as diluents. Extraction coefficients for cesium from simulated 103A supernate and "1965" FTW, using 1 M extractant, were very low (maximum value 0.16). Addition of caustic (equivalent to 0.15 mole/liter) caused precipitates to form in the organic phase, except with 1 M o-phenylphenol in nitrobenzene. In the latter case the cesium extraction coefficient was increased to 1.3. This extractant may merit further study.

ANALYTICAL AND INSTRUMENTAL CHEMISTRY

Cesium Analysis by Solvent Extraction

A series of experiments were performed to determine the effect of various potential interferences in the cesium analytical method (which uses dipicrylamine-nitrobenzene as extractant). Of the ions tested, only sodium, potassium and ammonium interfered; however, (using 0.025 M DPA) sodium levels of 0.5 M and potassium and ammonium concentrations of somewhat less than 0.05 M were acceptable, assuming a single stage of extraction and equal volumes of aqueous and organic. Further studies are being made with 0.08 M DPA, which should allow higher concentrations of competing ions to be tolerated.

EQUIPMENT AND MATERIALS

Hydraulic Equipment

A Deming jet pump has been received and installed in its test stand. Initial operation has been smooth and quiet for 120 hours, pumping hot water (195 F) at 25 gpm against a head of approximately 75 feet. Operation of the jet has been satisfactory for temperatures up to 201 F; at 202 F the capacity and head developed by the pump decrease markedly because of the reduced net positive suction head which permits cavitation within the pump, and loss of priming by the jet.

Further tests have been completed with a 75 gpm steam jet submerged in boiling water and employing a cold water inlet quench. Overall dilution approximated 30 percent, including cold water and steam dilution. Steam was supplied at 80 psig and final delivery temperature was from 88 to 90 C.

Gasketing Materials

In the continuing search for a radiation resistant material to replace B plant polytetrafluoroethylene gaskets, two more materials have been considered. The two materials which have been tested in a B plant connector are asbestos Flexitallic®, and polypropylene.

A white asbestos two-inch pipe size 150 ASA Flexitallic gasket was tested with 115 psig steam for 13 days. With a torque of 400 ft.lb. there was no apparent leakage. In a second test, a Flexitallic gasket was compressed between flanges that were 1/2 degree from parallel. The gasket compressed 0.028 inches more on one

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side than it did on the opposite side. The gasket was tightened with 200 ft.lb. torque. After six days of 115 psig steam service there was no apparent leakage.

Polypropylene was found to be unsatisfactory due to rapid flow.

Non-Metallic Materials

Bavick 11, a transparent plastic material produced by J.T. Baker Co., discolored and swelled in nitric acid and dissolved slowly in hexone. It was essentially unaffected by Purex HAX, Recuplex CAX, carbon tetrachloride, and 25 percent DBBP-75 percent carbon tetrachloride. Test duration was 40 days at room temperature. As a control, Plexiglas II UVA® was tested in the same solutions for the same test duration. The effect was substantially identical.

A fibrous potassium titanate thermal insulation was tested by static immersion at room temperature for 36 days. This material dissolved in nitric acid after one hour and softened in water after one day. It was unaffected by Purex HAX, Recuplex CAX, caustic soda, carbon tetrachloride and hexone.

Static immersion tests in nitrobenzene (diluent for the dipicrylamine cesium solvent extraction process) were made on a number of plastic materials. Only polytetrafluoroethylene, polyethylene, and a silicone rubber were unaffected.

Corrosion in Formaldehyde Treatment Reactor

Tests to determine corrosion of 304-L stainless steel under a range of conditions simulating operation of the Purex plant formaldehyde treatment prototype vessel are in progress. Samples of 304-L were exposed to boiling solutions of the following compositions: 0.5 to 2.0 M HNO_3 - 0.0 to 2.5 M H_2SO_4 - 0.35 M $Fe_2(SO_4)_3$ - 0.4 M $NaNO_3$. The highest corrosion rate observed (1.3 mils/mo) occurred in the solution containing 2.5 M H_2SO_4 and 2.0 M HNO_3. All other rates were 0.5 mils/mo or less. Similar tests except with Cr(VI) and Cr(III) present at 0.01 M are in progress. Preliminary observations indicate significant corrosion acceleration with Cr(VI) present; effects of Cr(III) are not evident at this time.

Effect of Ferrite on Corrosion Resistance of Welds

Corrosion test samples were cut from six all-weld-metal pads purposely prepared to provide a wide range of ferrite content.

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The ferrite content, as measured by an Elco Gage, ranged from zero to 28 percent. Test samples were exposed to boiling 65 w/o HNO_3 and to a boiling synthetic Purex IWW solution. No correlation between corrosion rate and ferrite content was found for these test samples.

Filter Screens for Strontium Oxide

Nickel, platinum and stainless steel were examined as possible filter screen materials to withstand strontium oxide and strontium oxide-lithium fluoride at high temperature. The tests were made at 750 C. Stainless steel (430) corrosion was excessive (8 - 17 mils/mo) with both. Nickel ('A' type) corrosion resistance was good (ca. 1 mil/mo) and platinum resistance was excellent (0.1 mil/mo).

PROCESS CONTROL DEVELOPMENT

Turbine Meter Evaluation

Turbine-type meters are being evaluated as a means of obtaining improved reliability of organic stream flow measurements over that provided by rotameter instruments currently in use. Two small (one gpm) meters have been under test for approximately six months in the experimental C-column facility. Their performance has been generally satisfactory for this purpose (there have been some corrosion effects in an acidic aqueous stream). A plant-sized turbine meter (150 gpm) has been installed in a test loop to determine its operational characteristics. The test program devised for the meter concentrates on determining its fidelity under intermittent flow conditions. The 2DX stream in Purex has been selected for a subsequent test of the meter.

Glass Scintillator for Counting Alpha Particles

A low silica, cerium-activated scintillating glass has been successfully used for counting thorium alpha particles, in solution, in a laboratory test. A multiplier phototube and preamplifier are required for an adequate signal-to-noise ratio but the equipment complexity relative to other methods, such as silicon PN alpha particle counting, is more than offset by the higher counting rates obtained. As an example, the count rate obtained with the scintillating glass was 41,000 cpm versus 900 cpm obtained using a diffused junction silicon PN alpha counter when exposed to the same plutonium alpha source in air. Laboratory calibration in a 7/8-inch diameter by 1/8-inch thick Teflon® sample cell resulted

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in a count rate of 6100 gpm for a solution concentration of 300 g/l of thorium. Background was reduced to about 200 cpm after a single water wash of the cell.

The alpha counting system described above will be thoroughly evaluated on the product stream of a thorium resin separation column in the 321 Building prior to considering it for installation on a separations plant stream.

Neutron Monitor in Purex E-6 Tank

The need for monitoring plutonium concentration in the E-6 tank has resulted in development effort on an in-tank neutron monitor. A probe containing a BF_3 tube and preamplifier appears to be practical for this application. A novel instrument connector jumper has been devised to make installation of the neutron monitor possible. The new jumper will provide four to six instrument connector heads using an existing single penetration in the cell wall. Successful application of this idea can alleviate a long-standing problem of limited instrument connectors in both Purex and Redox plants.

Boiling Point Determination

A device to determine the boiling point of Redox dissolver solutions is under development. If the boiling point can be measured to within 0.5 C, it will provide useful information concerning the extent of uranium dissolution. Boiling point determinations are subject to substantial error, and initial efforts have been directed toward experimental study of the accuracy that can be expected. Temperature measurements using a standard thermohm bulb and resistance recorder were found to be accurate within 0.15 C. A heated probe was tested which enabled measurement of the boiling point of water to within one degree centigrade. However, changes in the rate of boiling water and salt solutions produced measured boiling point variations of about 0.6 C. The information obtained to date casts doubt on the feasibility of achieving 0.5 C accuracy under plant conditions, but tests are being continued to determine methods of minimizing errors inherent in the measurement.

pH Flow Cell Design

At the present the E-3 tank is sampled and pH is determined by use of a one-drop pH electrode in the Gilmont sampler. Radiation exposures to personnel using this method are quite high and an alternate method is needed. By installing a pH flow cell and appropriate shielding in the in-line cavity of the E-3 sampler,

the pH can be obtained without personnel exposure. Design has been completed on a system which provides the required pH readout and permits remote buffering of the electrode.

C-Column Data Analysis

Withdrawal of 200 ml samples from a sequence of ports on the experimental C-column perturbs column operation and introduces some error into analysis of the resulting data, since steady state operation is assumed. An estimate of this error was made for each port and the sequence of port analyses was determined which minimizes the average absolute error over all ports. It was assumed in this determination that (1) withdrawal of the samples causes the column contents to drop an equivalent volume and hence introduces a concentration discontinuity at the sample port, (2) this disturbance decays back to the steady state value in a manner similar to heat conduction, that is, the process is described by the partial differential equation:

$$\frac{\partial v(t,x)}{\partial t} = K \quad \frac{\partial^2 v(t,x)}{\partial x^2}$$

where $v(t,x)$ is the aqueous phase uranium concentration at time t after a sample is withdrawn and at a distance x up the column, minus the steady state concentration, and K is a constant which has been determined experimentally, and (3) the samples are withdrawn at intervals of three minutes.

With these assumptions, the minimum average error that can be obtained was found to be about 0.3 gms/liter with a maximum error of about 0.5 gms/liter for one of the ports. The minimization is currently being repeated to determine the sequence resulting in the minimum average percentage error over all the ports.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Co-Deposition of Uranium and Plutonium Dioxides - Additional studies were performed to improve PuO_2-UO_2 electrolytic co-deposition techniques. In this instance, 2 w/o thallium as TlCl was added to a 2.6 LiCl-NaCl-KCl (1.5 w/o Pu, 5 w/o U) melt prior to electrolysis at 600 C with a chlorine gas sparge. It was hoped that the thallic ion formed by the chlorine would act as an auxiliary oxidant to increase the plutonium(IV)/plutonium(III) ratio in the melt and thus increase the PuO_2/UO_2 ratio in the deposit. The PuO_2/UO_2 ratio was indeed increased by a factor of three, from 0.007 to 0.02. Extrapolating from these results, to produce a deposit containing 1 w/o PuO_2 in UO_2 by this method would require a Pu/U ratio of 0.15 in the starting melt; or, for 25 w/o U, 3.8 w/o (50-65 g/l) Pu. The increase in the PuO_2/UO_2 ratio was accompanied by a marked decrease in the current efficiency, from 77 to 23 percent.

A brief investigation was made of the effect of melt temperature upon uranium/plutonium separation during electrolysis with an HCl gas sparge. With a 1.7 LiCl-KCl (4 w/o U, 4 w/o Pu) melt, the plutonium/uranium ratio in the oxide product remained nearly constant at about 0.02 (a separation factor of 52) over the temperature range from 500 to 700 C. Another observation from this work was that the plutonium separation factor is about the same with either a Cl_2 or an HCl sparge.

UO_2 Crystal Growth Studies - During the past quarter, work has been resumed on the electrodeposition of UO_2 in the form of dense, shaped, self-supporting pieces. At present, the growth of cylindrical hollow rods on rotating graphite (1/4-inch welding rod) cathodes is being studied. After a few trials with a starting melt of 2.5 KCl-$PbCl_2$ (25 w/o U), attention was switched to the KCl-LiCl (33 w/o U) system at 500 C. Using an HCl gas sparge, a 0.5 volt cell potential and a 5 ampere current, UO_2 cylinders about one inch in diameter have been built up on 2-inch long cathodes in a period of 16 hours. The deposits are composed of typical radially oriented columellar grains.

A systematic study is being made of the variables governing bulk density, O/U ratio, impurity content, and other important properties. Thus far, it appears that UO_2 bulk densities of 10.8 -

10.9 with corresponding lithium impurity contents of less than 10 ppm, can be achieved by using melt temperatures of 500 C or higher (bulk density and purity both fall of markedly with a 450 C melt temperature); and that higher bulk densities are associated with lower O/U ratios.

Electrochemistry of Uranium in Molten Chloride Salt Solutions - During the past quarter, the planned study of the electrochemistry of uranium in molten NaCl-KCl has been nearly completed, and work has been started on the NaCl-LiCl system. Recent measurements in NaCl-KCl melts may be summarized as follows:

1. EMF measurements for the cell reaction

$$UO_2 + Cl_2 = UO_2Cl_2$$

have been made as a function of temperature and uranyl chloride concentrations. Thermodynamic values for the reaction at 993 K were calculated, the results giving evidence of decreasing complexation of the uranyl(VI) ion with increasing uranyl(VI) concentration. For a concentration increase from 0.074 to 1.37 molal, the free energy change for the reaction (ΔF, cal/mole) was found to increase from -31,900 to -24,600; the entropy change (ΔS, entropy units) increased from 6.7 to 16.5; and the heat of reaction (ΔH cal/mole) changed from -25,200 to -8,200. An unmeasured junction potential has thrown an uncertainty into these values. Estimates of the error indicate any effect to be small, however.

2. Impedance measurements were continued with a modified impedance bridge. The results were much better than those obtained earlier, because of a much lower noise-to-signal ratio. Qualitatively the reactions previously proposed

$$(1)\ UO_2^{++} + e^- = UO_2^+$$

$$(2)\ UO_2^+ + e^- = UO_2$$

were confirmed in low uranium concentrations. The change in the potential of the second reaction as a function of uranium concentration shows the consecutive single-electron reactions to predominate up to a concentration of about 0.01 molal. At about this concentration, the potentials of the two reactions became about the same, and at greater concentrations only the overall reduction of uranyl(VI) to UO_2 is observed.

3. Using impedance measurement data, calculations have been made of the rate of reduction of uranyl(VI) to uranyl(V). Since the data did not follow simple theory, the suggestion of Laitinen and Randles [Trans. Faraday Soc. 51:54 (1955)] was followed, and corrections were made for the effect upon the impedance measurements of the adsorption of reactant ions onto the electrode surface. These corrections brought the data into line and permitted rough calculation of the rate constant as about 1.6 cm/sec. Assuming the transfer coefficient to be 0.5 (a reasonable value for a reaction with this large a rate constant), the exchange current densities were calculated to be 0.13 to 1.28 amp/cm^2 for uranyl(VI) concentration from 1.35 to 13.3 millimolar. The rate constant and exchange densities of this reaction are thus about the same as those for the reduction of Cd^{++}, Zn^{++} and Ag^{+}, which are considered to have fast electrode reactions.

Pilot Plant Electrolysis - Pilot plant electrolysis studies were continued in an 80 liter quartz pot with the 60 w/o KCl-40 w/o LiCl molten salt system. Melt preparation included dissolution of U_3O_8 feed with chlorine at 600 C and drying with chlorine and hydrogen chloride at 535 C. After the first run reported here a pre-electrolysis step formerly used for melt drying was discontinued without adverse effects on UO_2 product characteristics.

In the February monthly report (cf. HW-72902 C) it was noted that UO_2 product of small crystal size was produced from a melt containing 0.35 w/o iron. Using the same melt and impure feed a run was made at a lower reference potential, 0.33 volts, to determine the effect on product characteristics. Shutdown of the induction heater unit due to a water pressure drop caused the electrolysis temperature to drop from 535 C to 360 C rendering the results inconclusive. However, inspection of the product deposited before the temperature excursion indicated product characteristics similar to those in the earlier run.

The impure melt was discarded and a run was made with fresh salt and pure U_3O_8 feed. Eighty-five pounds of dense, large crystal UO_2 was electrodeposited in 54 hours of operation at 0.77 reference volts and 200 amps. Uranium concentration in the melt decreased from 30 to 13 percent during the deposition. The product had an O/U ratio of 2.0037 and a bulk density of 10.78 gm/cc. After crushing the density was 10.95 gm/cc or 99.8 percent of theoretical. Impurities in the bulk product after a water rinse were 70 ppm Li and 23 ppm K. In addition, some corrosion product from the Hastelloy D cathode was bonded to the product surface, where the product was deposited on the cathode.

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UO_2 Crushing and Classifying - A toothed, single-roll crusher made of stacked, staggered-tooth milling cutters operated successfully in crushing a variety of grades of electrolytic UO_2. Crusher product matched current particle size specifications in three ways: (1) all product passed a 6-mesh screen, (2) about 60 percent of the product was greater than 30 mesh, and (3) less than 10 percent was less than 200 mesh size. A certain fraction of the existing product must be further crushed to increase the percent of minus 200 mesh particles. This crushing will be studied with various equipment, including a modified roll crusher, double-roll crusher with adjustable roll clearance, and a commercial face mill.

Hydraulic classification has been abandoned due to failure to achieve efficient separation of particles in the minus 100 mesh size range. Various combinations of a basic "sorting column" arrangement failed to produce the desired separation. A sonic vibrating double-deck screen will be relied upon for classification in an integrated crushing/classifying system currently being designed.

Heating with Alternating Current - A.C. heating studies were made in a two-liter quartz pot containing 60 w/o KCl and 40 w/o LiCl and uranium concentrations of 20-30 w/o. Power was supplied by a 60 cycle A.C. source capable of delivering 320 amps at 12 volts (3840 watts).

Two different carbon electrodes configurations were tried. The first, 1/2-inch x 2-inch x 14-inch bars immersed six inches required 3840 watts to maintain the melt at 530 C. Localized inefficient heating was observed near the surface of the melt. The second electrode assembly was a "paddle" type with a 2-inch square blade connected to a 12-inch handle shrouded with a piece of quartz tubing to prevent heating at the surface of the melt. Only 800 watts of power were necessary to maintain the melt at 530 C.

Loosely bonded crystalline deposits were obtained on the A.C. electrodes when heating a wet melt. The deposit analyzed 52.6 percent uranium(VI). No appreciable deposit was obtained when a dry melt was used; however, a slight deposit was obtained after heating 8 hours at 8 volts and 100 amps. Analytical results on this deposit have not been reported.

UO_2 was deposited using a separate pair of carbon electrodes while heating with A.C. A 73 percent D.C. current efficiency

was obtained and the deposit appeared unaffected by A.C. Analytical results of the deposit have not been reported.

Graphite Containers for Molten Salt Baths - Previous work has indicated that molten alkali halide salts leach through dense graphite crucibles in one or two days, the exact time depending on the wall thickness and the structure of the graphite. In an attempt to overcome this disadvantage in an otherwise suitable corrosion-resistant material, a graphite crucible with 1/2-inch thick walls was impregnated with barium chloride (melting point 960 C) by submergence at 1100 C for 24 hours. After cooling, a 60 w/o KCl-40 w/o LiCl salt mixture was held in the crucible at 500 C. Although $BaCl_2$ is soluble in the melt, the leaching of the lower melting salt through the crucible was retarded four to six days, indicating that the barium chloride had significantly retarded the leaching process. Although this method does not solve the leaching problem by itself it may prove useful in combination with other frozen wall techniques.

An induction-heated crucible with 60 w/o KCl-40 w/o LiCl molten salt has been studied. A graphite crucible with a six-inch outside diameter and a 1/2-inch wall was heated with a ten kilocycle induction coil with a 9-inch inside diameter. The intervening space was filled with a one-inch thick annulus of quartz sand contained in a concrete-asbestos pipe with 1/2-inch wall thickness.

In one run the crucible was operated continuously for ten days at 450-550 C, with more salt being added as the liquid leached through the graphite. One temperature excursion permitted the entire void space in the sand to be filled with salt. Average wall temperatures outside the sand varied from 250 to 300 C during this period. The freezing point of the salt is 350 C.

The salt bath was then frozen and melted five times to test the mechanical strength of the graphite. Examination of cross sections of the crucible revealed no cracking of the graphite. The electrical coupling characteristics of the graphite did not change during the run. Further efforts will be directed toward increasing the heat flow through the non-conductive external construction materials to minimize the amount of salt outside the graphite.

Metallic Materials of Construction - Samples of tungsten, 'A' nickel, Ni-15Al and Ni-13Al were exposed for single six-hour periods in a Cl_2-sparged 40 w/o LiCl-60 w/o KCl melt at 600 C. The Ni-Al samples corroded at rates of 3 and 5 mils/mo, respectively; the other two corroded

at rates in excess of 50 mils/mo. The two Ni-Al alloys were exposed to the same melt with 10 w/o UO_2Cl_2 present. Corrosion rates for both were about 65 mils/mo and severe preferential attack of segregated phase was evident.

Aqueous Dissolution of PRTR Fuels

The second of two 5-inch long rods of irradiated (60 percent of goal) PRTR spike fuel (Al-1.8 percent, Pu-2 percent Ni) was declad and partially dissolved in the 222-S cubicle dissolver. The Zircaloy-4 cladding was readily removed in 6.5 hours by boiling solution initially 4.0 M NH_4F - 0.5 M NH_4NO_3. Attempts to activate the declad rod in boiling 6 M HNO_3 - 0.03 M NH_4F containing 0.005 M or 0.05 M $Hg(NO_3)_2$ failed. The rod appeared to dissolve at either $Hg(NO_3)_2$ concentration at a rate (ca. 1 mg/min-cm^2) expected for dissolution in nitric acid alone. Activation of the partially immersed declad rod was achieved readily in boiling 1 M HNO_3 - 0.015 M $Hg(NO_3)_2$ - 0.01 M NH_4F. This behavior parallels that observed in experiments with the first rod as reported last month (cf. HW-73202 C). As soon as activation occurred, concentrated nitric acid was added to make the solution about 6 M in HNO_3 and to completely immerse the rod. The rod remained activated and dissolved rapidly. The rate of acid consumption indicated that the irradiated rod, once activated, dissolved at the same high rate as has been observed for non-irradiated spike fuel alloy. Further experiments will be done with fully irradiated (100 percent of goal) spike fuel alloy to determine whether or not the above activating procedure is applicable.

RADIOACTIVE RESIDUE FIXATION

In-Cell Calciner

Several modifications and test runs were made on the mock-up during the month. Because of some difficulty in positioning the melt-pot cart under the column with the manipulators, a drive mechanism was constructed to move the cart in and out remotely. Side travel will still be done manually with the manipulators. In other manipulator tests, a spray nozzle was changed remotely and the off-gas and feed lines were moved from the spray calciner to the pot and back. Following one run, the column was also flushed down with water by use of spray nozzles which insert into the calciner. A high degree of cleanliness was achieved. The new 15-turn induction coil for the melt pot was completed and tested. Eighty to 90 percent of rated power was transferred to the melt pot through coaxial cables equivalent to those

anticipated for in-cell use. Most components for the r-f feed-through assembly have been fabricated, as has the 4000-ampere step plug for the DC power for the spray calciner. The high voltage power supply for the electrostatic precipitator has been received but is requiring modification to provide an intermediate collector voltage. The larger Variac transformers have also been received. Conceptual design of bus bars, hangers, transformer racks, etc. is complete and all detailed prints should be available by May 1 such that in-cell electrical work can begin. The in-cell gamma analyzer, which was dismantled during cell decontamination, is being modified slightly and reassembled, but in front of the right-hand window rather than the left, away from the heat and high background of the calciner, and away from the high-current and r-f conductors.

Analytical results have been received both on the recent spray calciner test run and on the first pot calcination test. In the spray calcination run, the de-entrainment factor across the stainless steel filter was 1300. The electrostatic bubble scrubber decreased the off-gas dust loading by a factor of seven when the voltage was turned on, even though the applied voltage was much less than the design value (since the high voltage generator for use with the unit had not been received). The off-gas dust loading was not decreased significantly by the two absolute filters, which indicates that a very high degree of clean-up was accomplished by the condenser and scrubbers, etc. The overall de-entrainment factor was about 10^6. The powder collected in the several spray calciner tests was melted down in the pot at a temperature of 850-900 C. The initial 7-inch deep layer of powder melted down to 1-1/4 inches for a final density greater than 3 g/cc. Some moisture and abundant sulfate fumes were evolved during the melt down.

The pot calciner run was made at a recommended boildown rate of 1.8 $l/hr.ft^2$, or 0.52 gal/hr for the 5-inch pot. Eighteen liters of simulated Purex waste were processed. Boildown was completed in nine hours. Off-gas samples showed very little carryover during this stage. After boildown, denitration was accomplished at 50 percent power, with some iron showing in the off-gas. Power was then increased to 75 percent and sulfates began to come off, as evidenced by white fumes in the condensate trap and scrubbers. After one hour the pot thermocouples indicated that melt-down had occurred (temperature suddenly rose to 850 C). The pot contained 9.3 pounds of melt whose density of 2.69 g/cc represents a volume reduction of 11. The run was uneventful except for the post calcination discovery that the 3/4-inch off-gas line had almost

plugged. Condensate samples showed that a large amount of sodium (78 grams) reached the condensate during the desulfation phase, suggesting that extensive foaming may have occurred and that this foaming may have caused the off-gas line fouling.

Synthetic Zeolites

A laboratory investigation of the cesium capacities and loading kinetics of synthetic zeolites continued.

Cesium capacities were determined in small 5.5 g columns of 0.25 to 0.50 mm grain size zeolites at 25 C. Cesium influent concentration was 0.08 M CsCl with Cs-134 as a tracer. Cesium capacities as determined at the 50 percent breakthrough point were 3.62 meq/g for standard Decalso, 2.65 meq/g for Linde 13X, 2.52 meq/g for Linde AW-500 and 0.78 meq/g for Linde 4A. Somewhat greater cesium capacities were obtained for Linde 13X under ideal loading conditions.

Loading kinetics of Linde 13X were examined, using two cesium concentrations, two temperatures and zeolite size ranges. With a 0.1 M CsCl influent loading, rates were 0.017 sec^{-1} and 0.023 sec^{-1} at 25 C for 0.70 to 1.00 mm and 0.25 to 0.50 mm size ranges, respectively. Temperature had little effect on loading rates between 25 and 40 C. With the 0.70 to 1.00 mm grain size range, full loading was attained in 10 minutes and 90 percent loading in 3.5 minutes. Full loading of the 0.25 to 0.50 mm size range required 8 minutes and 90 percent loading 2.3 minutes.

An influent concentration of 0.1 M CsCl is just in the region where particle diffusion is the ion exchange rate controlling mechanism. With organic resins, reaction rates in the particle diffusion region normally vary inversely with the particle radius squared. Linde 13X and other synthetic zeolites allow substantial flow of solution into the particle. The result is that synthetic zeolites show no such dependence of loading rate on the particle radius.

In the region of film diffusion rate control, a 0.001 M CsCl was used as influent. Initial loading rates for cesium of 0.0068 meq/ml/sec and 0.0033 meq/ml/sec were obtained for 0.25 to 0.50 mm and 0.70 to 1.00 mm, respectively. Approximately 50 minutes were required to full loading for the 0.25 to 0.50 mm size range at 25 C. In the film diffusion region, initial reaction rates vary inversely with particle radius.

Intermediate- and Low-Level Wastes

A sample of Purex coating waste from the 101-BY tank was treated with CO_2 near boiling temperature until a precipitate appeared. Preliminary results indicate that the precipitate carried strontium. The supernatant solution was diluted by a factor of 10^4 to simulate a high entrainment condensate such as might be produced by in-tank solidification of coating waste. The simulated condensate was passed through a column of clinoptilolite to determine the amount of radioisotopes removed. The cesium DF was greater than 100. Very little, if any, ruthenium was removed. Strontium and cerium concentrations in the influent were too low (less than MPC) to permit measurement of the uptake of these isotopes.

Condensate Treatment

Micro Pilot Plant Run 25 was performed to evaluate the decontamination ability of Amberlite 200 (a polystyrene resin) in the hydrogen form and to observe the hydraulic characteristics of the bed when the feed is not prefiltered with activated carbon. Over 6000 column volumes of steam-stripped Purex Tank Farm condensate waste were treated before the run was ended. Cesium-137 and strontium-90 concentrations in the effluent were less than their respective MPC_w throughout the run. The pressure drop per foot of bed depth across the resin column at a flow rate of about 1 gpm/ft^2 increased from about 0.1 psi at the beginning of the run to about 5.5 psi at the end of the run. The increase in pressure drop was primarily a result of the bed acting as a filter to remove sticky organic particulate matter present in the feed to the column. Although the organic matter penetrated the entire bed, it caused the agglomeration of a one-inch layer of resin at the top of the bed. Backwashing the bed with water was only partially effective in removing the organic. Mechanically breaking up of the agglomerated mat which had settled to the bottom of the column during the backwashing operation was necessary; in addition, a small amount of acetone was used to remove residual organic adhering to individual particles. It is believed that in a larger unit backwashing with hot water along with mechanical agitation will satisfactorily clean the bed prior to regeneration.

Release of Radiocesium from Clinoptilolite

Cursory tap water leaching tests were made on the 1350 C melts from the cesium-clinoptilolite volatilization studies. Leaching rates of the glass decreased with time ranging from 2.6×10^{-7}

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g/cm^2/day for the first eight days to 3.3 x 10^{-8} g/cm^2/day after 18 days. This appears to be in the same range reported for other glasses, but if the leaching rate continues to decrease there could be an order of magnitude difference after one year.

Clinoptilolite Beneficiation Studies

The presence of large quantities of high grade, coarse-grained clinoptilolite in the John Day Formation of eastern Oregon was confirmed. Arrangements were made with the Oregon Department of Geology and Mineral Industries for sampling of the formation. Samples will be compared with those from Hector, California, and Jersey Valley, Nevada, to help determine the optimum material and most desirable properties of the clinoptilolite for adsorbing cesium from wastes.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Geology and Hydrology

A program was written for the IBM-7090 computer to calculate flow times for three-dimensional saturated flow from a river bank lagoon or crib to the river. The program was successfully run and the initial results are being evaluated. The final generated results will be in the form of a table of dimensionless ratios which can be used to obtain solutions for various flow systems.

A program deck and listing of the "General Fit" subroutine was received from the Bureau of Standards at Boulder, Colorado. The Fortran program, written for a CDC-1604 computer, is currently being studied preliminary to revision for IBM-7090 use. This subroutine is a method for obtaining an optimum mathematical expression for the ground water potential everywhere on the Project using the potential measurements obtained from irregularly spaced wells. The expression will be used later as part of the input to generate soil permeabilities on a regular grid network. These permeabilities are necessary information for electrical analog simulation of ground water flow on the Project.

Two wells close to the 200-East Area B-Plant swamp show maximum ground water temperatures within the sand and gravel bed that lies upon basalt and beneath silts and clays of the lower Ringold Formation. The gravel bed is probably in part an alluvial fan deposit of an earlier eastward flowing stream. Sharp temperature

increases also were noted in several wells to the east and east-southeast of the B-swamp. In these cases the increases and maximum temperatures occurred in aquifers within the conglomerate member of the Ringold Formation beneath confining beds of silt and clay. These data suggest that warm waste water, probably from the B-swamp, is moving to and at a depth within the Ringold Formation.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Thorium-232 Bioassay Procedure

The Hanford thorium bioassay procedure was modified to include neutron activation analysis of the separated Th-232 and gamma counting of the Pa-233 produced. With a one day irradiation and a ten minute count, a sensitivity of 10^{-3} µg is easily obtained. Blank urine indicates a Th-232 level of about 10^{-2} µg. The Pa-233 is separated from the irradiated Th-232 by extraction into di-isobutylcarbinol with a yield of about 90 percent. It may be possible to measure the Pa-233 without chemical separation after a two-week decay period if results of the analyses are not needed in a shorter period.

Radiation Chemistry

Differences in activation energies of hydroxyl radical reactions between certain solutes and erioglaucine were calculated from reaction rate measurements made at different temperatures. The available data are tabulated below together with the protection indices:

Compound	Protection Index	Activation Energy Difference (kcal mole^{-1})
Glycine	8.3×10^{-4}	1.0 ± 0.4
Alanine	4.8×10^{-3}	1.2 ± 0.2
Tyrosine	4.7×10^{-1}	0.89 ± 0.08
Ethanol	1.40×10^{-1}	1.8 ± 0.1

It is interesting to note that while the protection indices of the three amino acids differ by factors of up to 500, their activation energies are not statistically distinguishable at even the 75 percent confidence level. In contrast, these activation energy differences are all distinguishable from that of ethanol. A possible conclusion is that radical attack on the amino acids is at some site in the structure common to them all, i.e. $\left[\begin{smallmatrix}-CH-COOH \\ -NH_2\end{smallmatrix}\right]$

[redacted] [redacted]

and that reaction at this site is characterized by an activation energy of ~1.0 kcal mole^{-1}. Differences in the actual rates can then be ascribed to steric factors involved in the approach of an HO· toward the sensitive site and to the molecular diameters of the compounds which determine their collision frequencies.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition in Conduits

Measurement of size distributions of particles used in recent deposition experiments in a 1/2-inch diameter vertical tube permitted more reliable comparison of the observed percent deposition with that predicted from previously developed equations. The actual particle size distribution resulted in closer agreement than when an estimated size distribution was used. The observed percent deposition was still from 5 to 8 percent greater than that predicted from the earlier regression equation for the deposition constant and the assumption of exponential deposition with tube length. The data fit significantly better an equation modified to show a 5.0 power dependence of deposition constant on velocity and a 4.0 power dependence on particle diameter, rather than the 5.26 power and 4.26 power dependence, respectively, shown in the equation representing the best fit to the bulk of the data from early experiments. The good agreement with the modified equation gives added confidence that the form of the equation is correct and that exponential deposition in a long tube actually occurs.

A multivarient graph was prepared from which the particle deposition can be quickly estimated when the particle size and density, air velocity, conduit diameter and length are known. A nomograph for calculating the deposition constant was also constructed. These will be included in a topical report which was prepared in draft stage.

W. H. Reas

Manager
Chemical Research and Development

WH Reas:cf

BIOLOGY OPERATION

A. ORGANIZATION AND PERSONNEL

W. C. Berlin, assigned to Plant Nutrition and Microbiology, terminated on April 13 to accept a position with the local AEC.

R. H. Schiffman, Biological Scientist in Aquatic Biology, terminated on April 20 to accept a position with NASA, Ames Research Center, Mountain View, California.

P. A. Olson was appointed Acting Manager of Aquatic Biology, effective April 23, 1962.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

The study testing the effect of reactor effluent on migrating-size chinook salmon fingerling was continued at 1706-KE. No effect of effluent on growth or mortality is yet evident at the tested levels of 3, 5, and 7 per cent.

Columnaris

Young salmon have been given repeated exposures to the columnaris organism to determine whether the fish can acquire immunity to the organism. Results to date are inconclusive.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Phosphorus

Cichlids fed food containing P^{32} levels of 0.25, 0.1, and 4.0 μc/g have shown no differences in mortality attributable to the isotope after four months.

Strontium

Four 100-day-old miniature offspring at the 625 μc Sr^{90} per day feeding level died or were killed when semi-moribund. These animals were born to swine that had been fed 625 μc Sr^{90} per day for approximately five months at the time of parturition, thus they were exposed to Sr^{90} in utero, and subsequent to birth, received Sr^{90} in the milk from their dam. At weaning (six weeks of age) they received 156 μc Sr^{90} per day until about one week before death when their feeding level was elevated as scheduled to 312 μc Sr^{90} per day. At autopsy, the prominent lesions were hemorrhage throughout the body, varying from petechiae to extensive extravasations

of blood. This extensive hemorrhage correlated well with the severe thrombocytopenia noted prior to death.

A terminal pneumonia was noted in one animal and a rather antibiotic resistant strain of Escherichia coli was isolated from the lung. Bacterial cultures of material from liver, spleen, post-cervical and mesenteric lymph nodes, and heart blood of the other animals yielded both E. coli and a Streptococcus. Their presence in these organs suggested a terminal bacteremia contributed to the death of the animal. Both were rather antibiotic resistant, but the significance of this has not been determined.

These deaths are the first of F_1 generation animals (exposure in utero, during suckling period, and subsequent to weaning daily Sr^{90} in feed) at any feeding level to be attributed to toxicity of Sr^{90}. At lower levels of feeding, the F_1 generation animals have shown only minimal changes or none at all. The ages of F_1 animals at these lower levels are 125 μc/day - 13 months, 25 μc/day - 30 months, and 1 and 5 μc/day - 36 months.

Three F_1 generation females on the 125 μc Sr^{90}/day level farrowed normal litters.

Experiments were performed to study the absorption of Sr^{85} and Ca^{45} by isolated mitochondria from rat liver incubated in vivo for 10 minutes at 30° in a solution containing ATP, succinate, cytochrome C, Mg, KCl and sucrose at pH 7. In the absence of stable strontium or calcium approximately 68 per cent of the Sr^{85} and 85 per cent of the Ca^{45} was absorbed. Addition of 0.9 μM Ca reduced Sr^{85} absorption to 11 per cent and Ca^{45} absorption to 40 per cent. The addition of 0.9 μM Sr to the basal media increased Sr^{85} absorption to 84 per cent and Ca^{45} absorption to 92 per cent. The opposite effect of the two ions is of interest in view of their chemical similarity.

Comparative Toxicity

One female miniature pig injected intravenously with 6.4 μc Ra^{226}/kg body weight about 7 months ago (when approximately four years old), died after exhibiting clinical symptoms of a severe nephritis. Gross lesions suggestive of a chronic nephritis were noted at necropsy. Another animal in this same experimental group is showing similar severe symptoms of chronic nephritis. Control animals and animals of similar age which received Sr^{90} remain normal.

Iodine

Two young lambs were intravenously injected with I^{125} and the pattern of localisation within the thyroid was determined at 5, 15, 30, and 60 minutes by autoradiography. Preliminary results indicate that satisfactory localization at the cellular level was realised using this isotope.

Neptunium

Rats were injected intraperitoneally with 10, 20 and 30 mg Np^{237}/kg body weight in order to study the symptoms of chemical toxicity. Neptunium was

administered as the citrate and all animals survived for 30 days, at which time they were sacrificed for radiometric and histologic examination. Excretion of neptunium in the urine during the first 24 hours was a function of dose level, approximately 13 per cent being excreted at the 10 mg/kg level; 29 per cent at the 20 mg/kg level and 32 per cent at the 30 mg/kg level.

An intravenous injection of 1.5 mg (1.0 μc) of Np^{237}/kg body weight in sheep caused moderate liver damage detected by radioiodinated rose bengal test. (In an extension of this study, and we hope conclusion, two sheep were injected with 0.75 mg (0.5 μc) of Np^{237}/kg. No damage (or only minimal) is expected at this level.)

DTPA appears to be very much less effective in promoting neptunium excretion than it is in promoting plutonium excretion. Percentage neptunium excretion was increased from about 25 per cent in controls to about 44 per cent in the treated animals over a three-day period. The relative ineffectiveness of the DTPA may be an indication that the neptunium is present in tissues as a monovalent ion, possibly NpO_2^+, which would be less effectively chelated than a divalent or trivalent ion.

Plutonium

Comparative toxicity studies with Pu^{238} and Pu^{239} on an equivalent microcurie basis indicate that the Pu^{239} is substantially more toxic. This would suggest that chemical effects are important in the acute toxicity of Pu^{239}.

Four pigs were injected intradermally with either 5 μc Pu^{239} nitrate at 12 or 20 sites or 1 μc at 24 or 88 sites and sacrificed either one or seven days after injection in order to determine the extent of translocation. After 1 and 7 days, 5 and 12 per cent of the injected dose were found in the regional lymph nodes and 2 and 6 per cent in the liver in the swine injected with 1 μc plutonium per site. It appeared that greater relative translocation was observed in the animals with 1 μc than with 5 μc sites.

Milk Transfer

Preliminary data available on the transfer of Cm^{244} and U^{233} from plasma to milk following single intravenous administration to lactating sheep reveal considerable difference in the metabolism of the two elements. Milk concentrations of U^{233} were less than 10 per cent of that found in plasma, whereas with Cm^{244} milk concentrations were four to five times those observed in plasma. The values for Cm^{244} are slightly higher than those previously observed for americium.

In a continuation of the above study, lactating sheep were given intravenous doses of Ca^{45}, Sr^{90}, Ru-Rh^{106}, Pm^{147}, or Ra^{226}, and plasma, whole blood, and milk concentrations of the radionuclides followed for 10 days post-injection.

Inhalation Studies

One beagle dog died two and one-half years after depositing about 4 uc Pu^{239} by inhalation of a plutonium dioxide aerosol. Death was associated with respiratory embarrassment. The acuteness of the dog's final collapse paralleled that of dogs that died two or three months after deposition of 50 to 100 μc Pu^{239}.

Although Neo-synephrine, DTPA, and Pluronics (polypropyleneglycolethylene oxide polymer) administered as aerosols after inhalation of $Ce^{144}O_2$ particles all markedly increased the clearance of Ce^{144}-Pr^{144} in rats, DTPA was most effective. In two experiments the lungs of rats killed three weeks after exposure to Ce^{144} and treated with DTPA aerosols contained about 8 per cent of the Ce^{144}-Pr^{144} found in the untreated controls. Pluronics and Neo-synephrine were less effective, reducing the lung burden to about 25 per cent of the controls. These experiments are being repeated in dogs with both $Pu^{239}O_2$ and $Ce^{144}O_2$.

Pulmonary function tests have been developed for dogs. Compliance can be measured on anesthetized dogs and is expected to indicate changes that might occur in pulmonary tissue as a result of radiation before changes can be detected radiographically. Analysis of blood for O_2, CO_2, and N_2 are also being completed on dogs before and after exposure to radioactive aerosols.

A computer program is being developed in cooperation with Data Processing Operation for reduction of excretion, tissue distribution, and retention data obtained in radioactive particle inhalation experiments. Early experience indicates the program will be particularly useful for fitting curves to daily excretion data.

Radiation Protective Agents

Two compounds, Aluminon (ammonium aurin tricarboxylate) and calcein W (fluorescein imino diacetic acid) were tested for their ability to protect rats irradiated with 1,000 r X ray. No animals survived but death was significantly delayed in the treated animals. Further studies with these and similar agents are in progress.

Experiments were conducted to study the enhanced acid phosphatase in mice injected with foreign bone marrow. There was some indication that alkaline phosphatase was elevated in mice receiving homologous bone marrow two weeks following 950 r X ray. It is planned to attempt confirmation with histochemical techniques.

Cellular Biology

The effect of D_2O on viability of cichlid eggs was examined. While control eggs developed in five days, 30 per cent D_2O delayed hatching time 24 hours. However, five days later D_2O-treated eggs began to die and all were dead by the 7th day.

Eggs exposed to 50 per cent D_2O did not develop and all were dead by the 5th day.

The results with eggs are in contrast with earlier observations in which newly hatched cichlids survived and fed when exposed to 30 per cent D_2O.

Thymus of white rats approximately three months old were irradiated with various doses to determine what changes occurred in the thymocyte population. Preliminary results indicate that the small lymphocytes decrease in number following exposure. This decrease appears to vary with dose. At two days after irradiation, several division figures were observed in the larger reticulum cells.

Plants

The accumulation of Zn^{65} by plants irrigated with Columbia River water is being examined. Preliminary experiments are concerned with the form of Zn^{65} in the river. Raw river water was filtered through 0.3 u millipore filters and thence through exchange resins. Thirty to 50 per cent of the Zn was removed by filtration, 35 to 45 per cent by cation resin and 5 per cent by anion resin. Essentially all of the Zn was removed by the extraction. Values are quite variable and seem to be related to suspended material in the water.

The effect of streptomycin on Cs^{137} uptake in barley plants was compared to chloramphenicol effects. Like chloramphenicol, streptomycin reduced Cs^{137} accumulation. The reduction caused by chloramphenicol has been attributed to its so-called specific action in blocking protein synthesis, however, the data obtained with streptomycin leads to the suggestion that the inhibitory effects may be due to an over-all impairment of metabolism.

Plant Ecology

The chlorophyll "A" content in leaves from common desert shrubs of the Hanford Reservation were compared during late April. Bitterbrush and sagebrush, poth perennials, contained 52 and 27 mg chlorophyll "A" per gram wet weight as compared to 18 and 15 grams, respectively, for hopsage and rabbitbrush, both of which are deciduous. Levels of the deciduous plants contained an average of 69 per cent moisture as compared to 44 per cent for the evergreens.

Project Chariot

Strontium analyses on 13 caribou collected in the Alaskan Arctic during 1961 were completed. The average levels found in flesh during 1961 and 1960 were as follow:

Date	No. of animals sampled	Avg. pc Sr^{90}/g Wet Wt.	Avg. pc Sr^{90}/g Std. Dry Wt.	Max. positive measurement pc Sr^{90}/g wet wt.
1960	15	< 0.017	< 0.067	0.022
1961	13	< 0.025	< 0.095	0.017

(These values are comparable to the measurements on three Alaskan caribou reported by A. R. Schulert in the April 13, 1962 issue of Science.)

It appears that cesium in flesh of caribou is actually more important from the radiation hazard aspect than Sr^{90}. Cesium-137 concentrations in flesh of caribou killed during 1961 are:

Collection site	Number of animals	Average pc/g wet wt.
Colville River	9	9.9
Ogotoruk Creek	12	1.1
Noatak River	22	12.0

The difference of cesium content in caribou from different locations is probably attributable to ecological factors such as differences in rainfall and food habits of the animals.

Goose Nesting Survey

Eighty per cent of Canada goose nests within the Hanford Reservation terminated during the month. There were 122 successful nests, which is 26 per cent fewer than observed during 1961. It now appears this will be the least productive nesting season observed since studied began in 1953.

Internal Emitter Committee

The April meeting of the Internal Emitter Committee, postponed because of the criticality incident, was rescheduled for May. Dr. R. F. Foster will be the guest.

HA Kornberg
Manager
BIOLOGY OPERATION

HA Kornberg:es

C. Lectures

a. Papers Presented at Society Meetings

F. R. Mraz, "Intestinal Absorption of Ca^{45} and Sr^{85} as Affected by the Alkaline Earths and pH," Federation Meetings, Atlantic City, New Jersey, April 16-18, 1962.

W. C. Hanson, "Project Chariot," American Chemical Society, Desert Inn, Richland, Wash., April 26, 1962.

b. Off-Site and Local Seminars

H. E. Erdman, "Radiation Damage to Genetic Material," Exchange Seminar, Washington State University, Pullman, Wash., April 24, 1962.

E. M. Uyeki, "The Regenerating Liver as a Model for Mitosis," Exchange Seminar, Washington State University, Pullman, Wash., April 3, 1962.

c. Seminars (Biology)

None

d. Miscellaneous

L. K. Bustad, "Impact of Atomic Energy on Agriculture Today and Tomorrow," Annual FFA Banquet, Benton-Kiona High School, Benton City, Washington, April 23, 1962.

D. Publications

a. Documents

Kornberg, H.A. and Staff of the Biology Laboratory. 1962. Biology Research Annual Report for 1961. HW-72500.

McClellan, R.O., J. R. McKenney, and L. K. Bustad. 1962. Metabolism and Dosimetry of Cesium-137 in Male Sheep. HW-72511 (February 1962).

b. Open Liberature

Ballou, J.E. 1962. Removal of deposited plutonium by triethylenetetramine hexaacetic acid. Nature 193: 1303-4.

Cline, J.F. 1962. Effect of nutrient potassium on the uptake of caesium-137 and potassium and on discrimination factor. Nature 193: 1302-3.

Nakatani, R.E. 1962. A method for force-feeding radioisotopes to yearling trout. Progressive Fish-Culturist 24: 56-59.

OPERATIONS RESEARCH AND SYNTHESIS OPERATION
MONTHLY REPORT - APRIL, 1962

ORGANIZATION AND PERSONNEL

There were no changes in organization or personnel during the month.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Fuels Preparation Department

The first rough draft of the formal report presenting the empirical models which relate dimensional distortion during irradiation to reactor environment has been completed. The report is being coauthored with FPD personnel.

Preliminary steps have been taken to incorporate additional calculations in the MERCY Program. When completed, this routine will permit a rapid evaluation of measurement performance at a given measurement station.

The production test proposed to evaluate alternate fabrication routes for the Alsi process uranium core was redesigned to include an additional route, gamma extrusion. Alternate designs were proposed depending on whether or not it proves feasible to cut four bars from the original dingot rather than three bars as is normally done.

A previous study had shown the existence of a segregation pattern along the uranium ingot for certain impurities and for grain size. Subsequent to this study, the cores were subjected to the UT-2 tester in order to determine if similar segregation existing for UT-2 voltages. The resulting data were analyzed, and expressions were found relating UT-2 voltages to position within the ingot.

A brief analysis was made of frequency of cracks in overbore fuel elements. The negative exponential distribution was shown to fit the data. This arises when the underlying frequency distribution is Poisson with varying parameter λ. Such a situation seems reasonable in this instance.

A design was proposed for a PWR comparing bond characteristics of ingot and dingot uranium with and without lead plugs. These will be compared at various combinations of Alsi and preheat times.

Personnel having the responsibility for development of the hot die sizing process held an orientation meeting in which the many process variables

[redacted] [redacted]

were identified. Subsequent to this, an initial test design was formulated. Preliminary attention will be given the preheating and press variables.

As requested, a critique is being made of a proposed document reporting on the effects of certain NPR process variables on the integrity of the zirconium-beryllium bonding layer.

A document was issued presenting the results of an analysis of warp data from 18 recent NPR extrusions. The document, coauthored with FPD personnel, recommended that evolutionary operation be investigated as a means of better understanding the effects of process variables and hence improving quality.

A study is being made of NPR wall thickness data to determine its behavior along an extrusion.

A preliminary analysis has been made of clad thickness data from some 100 NPR extrusions. This is aimed primarily at comparing vendors.

Further data have been analyzed from experiments designed to establish cooling rates for NPR fuel elements after the Beta heat treating. An additional experiment was designed and recommended.

Calibration formulas were derived relating NPR fuel element clad thickness as a function of film density. A discussion of the error sources was included in the resulting report. This enables a determination to be made of when a given calibration differs from previous ones.

Some work was done in connection with the development of a tester designed to detect uranium contamination in the bonding layer. Counting statistics are used in this application.

Due to a combination of circumstances, there are on occasion rather gross discrepancies between official forecasts of requirements for a given fuel element type and actual demands. A study was made to see if such discrepancies could be objectively anticipated. A model was developed on the assumption that long range forecasts should be fairly accurate, and hence, discrepancies in one direction for previous months should be followed by discrepancies in the other direction. This model is being tried informally to investigate its applicability.

Irradiation Processing Department

Evaluation programs for reactor incident data have been written and debugged. Although these data are being evaluated primarily for use in constructing the computer simulation of reactor operations, discussions with IPD personnel

have indicated that they would be useful for management control purposes. For this reason, arrangements have been made to present to interested IPD personnel a project program report, analytical information description, and data recording system during the first part of May. Meanwhile, other parts of the study are progressing.

Uniform corrosion data are being analyzed using existing corrosion models in order to compare them and possibly derive an empirical model if deemed necessary. This is important both from a fuel and process tube standpoint.

As reported previously, additional total count data were taken on fuel elements from the suspect lot which had fuel elements in it from one canning line known to have operated outside of process specifications for an undetermined amount of time. These data were analyzed, and a recommendation was given for disposition of the suspect lot.

Preliminary work was done in investigating the feasibility of using gamma monitor data, Panellit pressures, and exit water temperatures in a more objective manner to permit earlier rupture detection. Although statistically it appears that this can be accomplished, it is questionable whether or not a workable system can be developed.

Work continued on the problem of estimating the probability of detecting defects in welded primary piping for the NPR project. Currently, a mathematical model of defect frequency and size is being constructed preparatory to a statistical analysis of data from guided bend tests.

Work continued on the provision of a reliability analysis of various reactor safety systems employing a "k out of n" type of trip logic.

At the request of members of IPD, R and E a talk on a reliability algebra for four-state safety devices was presented to the group.

Chemical Processing Department

One of the current problems in CPD's accountability system is the lack of good agreement between the measurement by the ratio method of the amount of plutonium received from IPD and the measurement of the amount of plutonium nitrate obtained from the primary separation plants. The possibility of using the isotopic dilution method developed by AERE at Harwell for the measurement of receipts from IPD was suggested. In addition, the feasibility of using the relationship between plutonium nitrate concentration and specific gravity for verification of measured plutonium content of primary plant output was examined by multiple regression analysis of available data.

The use of weekly measurements of the average plutonium content and part-by-part variability has operated quite satisfactorily for control of minimum plutonium content requirement. The use of measurement of plutonium^{240} content of every part by neutron counting for process control along with a random sample for measurement by means of a mass spectrometer of the process average and part-to-part variability has not proven entirely adequate. The random sample disclosed a significant change in both process average and part variability that was not detected by the routine control measurement. Further attention is being given this problem.

Further work on the statistical risk problem associated with the shipment of radioactive materials awaits a response to a request for information from the Interstate Commerce Commission. Correspondence has also been exchanged with the National Safety Council.

Additional consultation was given in interpreting extensive work sampling data collected by CPD personnel.

Mathematical expressions have been developed which express the self-interaction coefficients of annulus-shaped containers with interior radial baffles. Such containers are used for storing fissile materials, and their self-interaction coefficients are important in criticality studies.

Document HW-73467, "A Contour Completion Algorithm", has been issued.

Medical treatment injury statistics are currently being analyzed to give appropriate confidence limits for parameters occurring in the mathematical model.

Relations Operation

Further salary curves were fitted as requested. In addition, help was given in interpreting some of these curves.

Advice was given in connection with appropriate techniques for obtaining representative samples of plant personnel for survey purposes.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HLO

2000 Program

Further work was done on the problem of estimating the parameters in the system of nonlinear differential equations which express mass transfer dynamics of the extraction process as a function of column position using organic and aqueous

uranium concentration data taken at 17 ports along the column during a single experimental run. The analog computer is being used to obtain preliminary guesstimates, which, in turn, are the initial values used in the nonlinear least squares 7090 calculation. Since the analog and digital fit metrics are slightly different a transformation is being developed to map analog estimates into digital ones.

The absorptiometer calibration curves for the backup set of four test cells were completed. These calibration functions and cells will be used for future experiments if the original test cells are damaged.

A discussion was held concerning the fitting of reservation water table data to an appropriate nonlinear diffusion model. Hopefully, the end product of the analysis will be a mathematical expression for the permeability function in the diffusion equation which can be used to estimate soil permeability characteristics beneath the reservation.

Theoretical studies to determine the heat transfer properties of a proposed annulus-shaped container for high thermal source materials have been completed. The model was evaluated for a wide range of its design parameters on the 7090 computer, and the findings reported.

A formal solution has been obtained to the problem of determining the steady-state nonviscous flow pattern of a fluid in a cylindrical tank which has been equipped with an axially located circulating device. A computer program is being written to evaluate these flow patterns for a variety of tank dimension and circulating devices. The problem was suggested by Engineering Development, CR and D, as part of their studies on waste concentration methods.

Computations were made to determine the theoretical particle size and proportionate mix factors necessary to produce maximum density for several experimental models of vibrationally compactified fuel elements. First results from the actual tests were excellent.

3000 Program

As agreed upon with the Manufacturing Services Laboratory in Schenectady, three 7090 generated magnetic tapes were supplied for the purpose of checking out the prototype Gorton controls under development there. Word has been received that the test series was completely satisfactory and supplied valuable information on reliability, accuracy, speeds, accelerations and auxiliary functions.

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4000 Program

Irradiation Effects on Structural Materials

Sets of experimental data are being collected to test the appropriateness of the tentative format for the creep raw data portion of the mechanical properties raw data file.

Plutonium Fuels Research

A statistical evaluation of plutonium analyses on eight types of plutonium oxide pellets was completed. The analyses were performed over a two-year period by several analysts using different pellet sampling techniques. The evaluation included estimation of the time trends in the data, bias and precision of each of the pellet sampling techniques, and, in addition, 95 percent confidence interval estimates of the average plutonium content of each pellet type.

Further work was done on precision and accuracies studies of several methods of estimating trace boron content. A statistical analysis is in progress to correlate our analytical laboratories data with Battelle boron content estimates based on reactivity measurement to determine (1) whether there is a bias between the two procedures and (2) the relative precision of each.

5000 Program

Actinide Element Research

A FORTRAN program to determine cubic crystal indices from X-ray diffraction data was tested with five sets of data. An improved version using a least squares fit of the lattice constant has been written and is currently being debugged. Consideration is also being given to the more complicated mathematical problem of indexing hexagonal crystals.

Division of Research Programs

A rough draft report was completed on the search anddetection problem. Further work was done with ZERO search and planer plot routine. Work continued on the general definition of the IRA Mark-II Program file. In this connection, weekly discussions have been held with EDPO personnel. A statistical analysis of laboratory blanks and duplicate analyses was completed and the results reported.

The SPEC program for the quantitative resolution of a time dependent gamma energy spectrum was rewritten without the use of matrix calculation subroutines. The revision uses about 50 percent of the memory required by the matrix language program. The rough draft of a report, "Fixed Time Estimation of Counting Rates with Background Corrections", was almost completed.

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6000 Program

Biology and Medicine

Work continued on fitting a multicompartment model to data from a retention study on fish. A program must now be written to solve the resultant system of differential equations.

Analysis of data from a study of discrimination of strontium over calcium was begun.

Other

Atmospheric Diffusion Studies

Work continued on the fitting of tentative diffusion curves to the APE data using the nonlinear least squares program. Several discussions were held concerning the statistics of particle sizing in connection with an experimental program to estimate the scavenging efficiency of rain drops as a function of their average diameter. Propagation of error variance formulas were supplied for several scavenging efficiency estimators.

Several sets of data were analyzed to determine the relation of excretion in the urine to time after intake.

A nomograph was constructed for use in evaluating a formula of the form:

$$y = a_0 x_1^{a_1} x_2^{a_2} x_3^{a_3} / (x_4 x_5 + a_4)^{a_5} \quad .$$

The last two in a series of lectures on probability and statistics were presented to participants in the Engineering "A" Course.

Carl C. Bennett

Manager
Operations Research and Synthesis

CA Bennett:dgl

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PROGRAMMING OPERATION

APRIL, 1962

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Combined Cycles

The combined cycles studies are planned to show the effects of fuel element geometry, reactor design, fuel management strategy, and economic climate on fuel cycle costs and bred fuel values. These studies involve a large number of interrelated nonlinear variables; therefore, it is difficult to assess the impact of a given variable in studies of a single specific reactor type. It becomes necessary to parameterize the variables choosing values likely to show both the direct impact of the variable and the degree of its interaction with other variables of the study.

Such studies are further complicated by the difficulty of representing the neutron reaction phenomena in a model suitable for use in computer analyses.

The model originated by C. H. Westcott has gained wide popularity for use in survey type codes. Unfortunately, it has limitations identified by Westcott for use in analyzing reactors having hard neutron spectra. Plutonium isotopes have high neutron thermal cross sections and, therefore, tend to harden the neutron spectrum when used as a fuel enrichment even more rapidly than, for instance, does U-235. The 1 ev resonance of Pu-240 further complicates the analysis.

MELEAGER, the burn-up code used in the combined cycles study involving ten fueling schemes summarized in Table I, utilizes the Westcott flux and cross section model. One of the advantages is that MELEAGER adjusts the cross section model during the burn-up calculation. A more sophisticated model could increase the expense of burn-up calculations to a prohibitive degree for survey work if the spectrum were similarly adjusted during burn-up.

Preliminary results indicate that some of the more interesting combinations of fuel elements types and reactor parameters involve hard neutron spectra that change with irradiation. The Westcott model is least valid for hard neutron spectra. In addition, Phoenix and reduced density fuels appear attractive in hard neutron spectra and the Zoned Spectrum irradiation begins with a hard spectra. The MELEAGER data heretofore reported for all three of these concepts are likely valid but the exact combination of isotopes and neutron spectra producing the desired result are likely to change with an improved neutron reaction model. For these reasons, the calibration of the Westcott model has assumed importance and urgency.

TABLE I

IDENTIFICATION OF FEED COMPOSITIONS BY SERIES NUMBER

Assigned Series No.	Fuel Combinations	Plutonium Compositions Used for Fissile Enrichment Pu-239 %	Pu-240 %	Pu-241 %	Pu-242 %
1,000	U-235 in uranium		none		
2,000	U-233 in depleted uranium		none		
3,000	Plutonium in depleted uranium	70	18	11	1
4,000	Plutonium in depleted uranium	95	5		
5,000	Plutonium in depleted uranium	31.3	33.6	25	10.1
6,000	U-235 in Th-232		none		
7,000	U-233 in Th-232		none		
8,000	Plutonium in Th-232	70	18	11	1
9,000	Plutonium in Th-232	95	5		
10,000	Plutonium in Th-232	31.3	33.6	25	10.1

SPECTRUM V (described in HW-71953) is being used to calibrate the Westcott flux and cross section model. This work, being carried out with the assistance of the Applied Physics group, has indicated a sizable discrepancy between the spectral index, as calculated in MELEAGER code, and the spectral index calculated by SPECTRUM V code when enriching with the amount of plutonium typically associated with minimum fuel costs. Of course, the discrepancies are minor if one uses well-moderated cases with natural enrichment. As indicated, C. H. Westcott continually cautions readers to limit the use of the so-called Westcott cross section to such cases. Westcott does not, however, demonstrate the specific limitations of the cross section and flux spectrum model. The interest in using MELEAGER code for spectrums somewhat harder than those for which the Westcott system was designed is so great that the model must be calibrated and, if possible, extended to cover this region. It is not enough to know that errors exist; it is necessary to know their magnitude and direction, and, if possible, to supply sufficient bias to allow the use of the Westcott model with reasonable accuracy. In the Westcott system, cross sections are analytically averaged for each isotope and constants are derived for use in the following equation:

$$\hat{\sigma} = \sigma_0 (g + rs)$$

where

$\hat{\sigma}$ = effective Westcott cross section

σ_0 = 2200 meters/second cross section for monoenergetic neutrons

g = a constant, typical of a specific isotope reflecting its neutron absorption properties in the thermal neutron energy range

s = a constant, which primarily reflects the neutron absorption properties of a specific isotope in the epithermal energy range

r = a spectral index reflecting the relative amounts of thermal and epithermal neutrons present with larger r's indicating a so-called harder neutron spectrum.

The spectral index r becomes especially powerful in this equation when plutonium isotopes are analyzed because the plutonium isotopes have rather large "s" factors.

The Applied Physics group has attempted to calibrate MELEAGER code against critical mass measurements with plutonium and has demonstrated reasonable results using the spectral indexes previously derived from the Multigroup code (75 groups to 4.2 ev), SPECTRUM V. This code uses a rather elegant slowing-down model which has been checked against cadmium ratios and other spectrum measurements in GA-2544 by Beyster, et al, of General Atomics. The r's calculated by MELEAGER code using geometrical SDPV's* were much higher as was expected and gave extremely pessimistic results when used to calculate reactivity of plutonium enriched cases. Thus, it appears that the Westcott system of cross sections could be used with reasonable results for considerably harder neutron spectra with an improved spectral index model or formulation.

The MELEAGER formulation for spectral index, r, follows. Most of the terms are unique to MELEAGER and are obscured further by the limitation of the Fortran coding system. As a consequence they are defined further:

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* SDPV is the volume and flux weighted slowing down power of the moderator normalized to the fuel.

$$r = \frac{SIGMA + SNF}{\frac{SDPV \times TNL}{RTM} - S_{sum}}$$

where

$$SIGMA = \sum_{o}^{i} \overline{\sigma}_{o_1} y_1 g_1$$ summed over-all fuel isotopes

and $\overline{\sigma}_o$ is the 2200 meter/second absorption cross sections for monoenergetic neutrons and y is the nuclei/barn cm, g being the Westcott constant previously discussed.

$$SNF = \left(\frac{f}{1-f}\right) \sum_{o}^{fuel}$$ and f is the thermal utilization factor (SNF is a means of introducing nonfuel neutron absorption).

SDPV is the volume and flux weighted slowing-down power of the moderator normalized to the fuel.

TNL = thermal nonleakage probability

$$RTM = \sqrt{\frac{\pi}{4} \frac{T_n}{T_o}}$$

where T_n in neutron temperature in ^oK and T_o is 293 ^oK (20 C) standard and

$$S_{sum} = \sum_{o}^{i} \overline{\sigma}_o T S$$

where $\overline{\sigma}_o$ and Y are, as before and S is the Westcott parameter unique for each fuel isotope.

This formulation was examined with the intent of empirically improving its applicability to hardened neutron spectra. An examination of uncorrected MELEAGER Spectral indexes (r's) indicate that they are too high for any isotope present in appreciable concentration, but that they are even higher for isotopes having large "s" factors such as Pu-240. An early attempt to correct for r by using a simple function failed when appreciable quantities of Pu-240 were present.

A modified MELEAGER deck suitably labeled to indicate that it contains unusual experimental alterations gives spectral indexes that track those from spectrum code for single isotopes and mixtures (see Table II) calculated by the following equations:

$$r_1 = \frac{SIGMA + SNF}{\frac{SDPV \times TNL}{RTM} - \left(\frac{1}{1 + \frac{k_1 S_{sum}}{SDPV}} \right) S_{sum}}$$

which is operated on by the equation

$$r = \frac{r_1}{1 + k_2 r_1}$$

where k_1 and k_2 are constants supplied as input to MELEAGER code.

k_1 is adjusted to give suitable influence to the epithermal absorptions and serves to give self-shielding to this factor because in the formulation it is multiplied by S_{sum}.

k_2 then adjusts the degree of self-shielding applied to all absorptions.

The spectral index formulation in modified MELEAGER has also been tried with mixtures of plutonium isotopes with varying amounts of boron present with similarly excellent results.

After additional checking of the modified MELEAGER formulation and cross sections they can be used to determine the magnitude of the conservatism of the standard MELEAGER code with respect to bred fuel values and fuel cycle costs for the various studies in progress.

TABLE II

A COMPARISON OF STANDARD AND MODIFIED MELEAGER SPECTRAL INDEXES WITH SPECTRUM V RESULTS FOR SINGLE ISOTOPES OF VARYING CONCENTRATIONS

Concentrations(1)	Spectrum r	MELEAGER Uncorrected r	Modified MELEAGER r(2)
U-235			
0.0002	0.0736	0.086	0.083
0.0004	0.137	0.1605	0.0148
0.0006	0.191	0.229	0.205
0.0010	0.281	0.356	0.301
0.0015	0.370	0.501	0.396
Pu-239			
0.00004	0.040	0.049	0.0474
0.00010	0.098	0.1137	0.106
0.00020	0.191	0.219	0.194
0.00040	0.336	0.3914	0.322
0.00080	0.507	0.5702	0.481
Pu-240			
0.0004	0.105	0.1487	0.102
0.0010	0.265	0.598	0.235
0.0020	0.453	11.82	0.4191
0.0030	0.566	--	0.569
0.0040	0.632	--	0.695

(1) Nuclei (barn cm of homogenized cell).
(2) Using $K_1 = 1$ and $K_2 = 0.5$

Zoned Fueling of Plutonium Enriched Reactors

The fuel element fabricating and jacketing cost, FEFJ, for an element that contains plutonium is higher than for an element that contains only uranium. This cost, which is called the "delta" FEFJ cost, is approximately independent of the amount of plutonium in the element over limited ranges. Thus, the total FEFJ cost of a reactor that is partially enriched with plutonium can be reduced by concentrating the plutonium into a few elements (as opposed to uniform enrichment in which every element will contain some plutonium). That is, the

average FEFJ cost per fuel element can be decreased by "zoning" the reactor such that some elements contain only uranium while the others contain a mixture of plutonium and uranium. This can be expressed as:

$$\overline{FEFJ} = FEFJ_u + V_r \triangle$$

where:

$\overline{FEFJ}$ = reactor averaged FEFJ cost per pound of fuel

$FEFJ_u$ = FEFJ cost for fuel elements containing uranium only

$\triangle$ = additional FEFJ cost increment for fuel elements that contain plutonium

V_r = fraction of the fuel elements in the reactor that contain plutonium.

The plutonium value calculations reported in HW-72217 for self-sustaining plutonium recycle assumed that the product $V_r \triangle$ was constant throughout the recycle series. However, if the zoning criterion is unchanged and the amount of recycle plutonium increases in successive steps, then V_r cannot be constant and this is equivalent to forcing $\triangle$ to vary. A more correct formulation was programmed that calculates V_r for each step in the recycle series and uses this value to determine $\overline{FEFJ}$ based on input values of $FEFJ_u$ and $\triangle$. The criteria used to calculate V_r are that the plutonium is mixed with depleted uranium and the initial fissile enrichment is the same for all of the fuel elements in the reactor.

Table III shows the reactor averaged FEFJ costs using the old and new calculational methods for the APWR, while Table IV shows the corresponding plutonium values in each case. It should be noted that the physics results were obtained from MELEAGER-CHAIN which uses the $\triangle \cdot V_r$ = Constant calculational method. Reminimizing with the $\triangle$ constant, V_r variable method would make a significant difference in these results.

The plutonium values shown in the next to the last column in Table IV are about \$2/gram lower than the values shown for Case APWR-IV in HW-72217, because the incremental charges for fabricating plutonium-bearing fuel are about twice as high in Table IV as for Case APWR-IV. Since the refined values shown in the last column of Table IV are about \$3.00/gm less than the inflated values shown in the adjacent column, this suggests that the plutonium values shown for Case APWR-IV in HW-72217 are high by about \$1.50 per fissile gram.

TABLE III

FEFJ COSTS FOR THE APWR USING TWO DIFFERENT FEFJ FORMULATIONS

Step Number	Fraction of the Fuel Elements that Contain Plutonium	Reactor Average FEFJ, $/Pound Fuel	
		$\triangle \cdot V_r$ = Constant[1]	$\triangle$ = Constant, V_r = Variable[2]
1	0.0	$40.00	$40.00
2	0.142	50.00	50.00
3	0.183	50.00	52.87
4	0.190	50.00	53.36
5	0.200	50.00	54.06
6	0.204	50.00	54.35
7	0.205	50.00	54.42

(1) Based on $FEFJ_u$ = $40 per pound fuel and $V_r \triangle$ = $10 per pound fuel.

(2) Based on FEFJ = $40 per pound fuel and $\triangle$ = $70.32 per pound fuel ($\triangle$ chosen so that FEFJ in Step 2 will be identical with the $\triangle \cdot V_r$ = constant method).

TABLE IV

PLUTONIUM VALUES AND FUEL COSTS IN THE APWR RESULTING FROM DIFFERENT FEFJ FORMULATIONS

Item	$\triangle \cdot V_r$ = Constant[1]	$\triangle$ = Constant V_r = Variable[2]
Fuel Cost, mills/kwh_e	2.58	2.61
Plutonium values in each step, $/gram (fissile)		
1	----	----
2	7.60	6.00
3	8.90	6.15
4	9.25	6.45
5	8.40	5.40
6	7.60	4.55

(1) and (2) See notes on Table III.

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Phoenix Fuel: Effect of Specific Power

Phoenix fuels have been studied at various specific power levels to determine the effect of Pu-241 decay and Am-241 decay on the exposure achievable with a given fuel composition. At specific powers of about 30 watts/cc the exposure is greatly reduced in comparison to the exposure achievable at powers of 100 and 300 watts/cc. The specific power of central station power plants is generally expressed in megawatts per ton of fuel (MW/T) and for UO_2 fuel will often be in the range of 10 to 15 megawatts per ton of uranium. This corresponds to 110 to 165 watts per cm^3 for UO_2 at theoretical density. The equivalent power in megawatts per ton for fuel with a density of 3 gms Pu/cc operated at a volumetric specific power of 165 watts/cc would be 50 megawatts per ton of plutonium.

It has been determined that the reduction in obtainable energy yield is caused primarily by the poisoning effect of americium-241 (Am-241). The information presented here points out the importance of specific power in fuels with high 241 content. The Pu-241 content of the fuel is important from reactor control and economics standpoints both because some Pu-241 is lost as a thermal reactor fuel by radioactive decay and because the Am-241 which is formed from Pu-241 decay is a strong absorber of thermal neutrons.

An investigation is presently being made to determine the impact on fuel costs of Am-241 formation in Phoenix fuels. It appears that it will be important to operate plutonium fuels containing a large percentage of Pu-241 at high specific power in order to keep the ratio of Pu-241 decays to Pu-241 fissions as low as possible.

The MELEAGER burn-up code was used to find the exposure for three groups of cases as shown in Table V. The specific power level was varied in each group.

In the first group, both the Am-241 formation constant and the Pu-241 decay constant were set equal to zero. These cases show how burn-up would occur if Pu-241 did not decay and if Am-241 was not formed.

In the second group, Am-241 was allowed to form as if Pu-241 were decaying at the normal rate. However, the code was operated in a manner such that there was no loss of Pu-241 concentration by decay. The results of these cases shows the effect on exposure of the Am-241 formation.

In the third group of cases the formation constant for Am-241 was equal to the decay constant for Pu-241 and both events (decay of Pu-241, formation of Am-241) were allowed to occur.

Table V lists the exposures obtained for the three sets of conditions mentioned. In computations for Table V the initial plutonium composition was: 62 percent Pu-239, 20 percent Pu-240, 15 percent Pu-241, and 3 percent Pu-242; the fuel density was 3-gram plutonium/cc; the neutron flux energy spectrum was quite hard. The specific power is listed both in megawatts thermal per ton of fuel and in watts per cubic centimeter. The exposure is listed in corresponding units.

The time involved in burn-up for each case is given both in days and years in Columns 6 and 7, respectively. It should be noted that there is a maximum obtainable lifetime due to Am-241 formation. Sufficient work has not been done to determine at just what specific power this maximum life would be at this fuel density but it appears to occur somewhere near 100 MW/ton. The limit in terms of years will be different for Phoenix fuels developed in hard spectra and in soft spectra.

At low specific power, a significant amount of Pu-241 (13-year half-life) decays by β^- to Am-241 (470 year half-life). The 2200 m/s neutron absorption cross section of Am-241 is $\sim$ 585 barns. Several low-lying resonances (0.3, 0.6, 1.3 ev) of Am-241 are greater than 3000 barns and other resonances exist at higher energies in the epithermal range. Thus, it is a very strong absorber, especially in hard spectrum reactors. The 2200 m/s capture cross section is variously listed between 600-700 barns while the 2200 m/s fission cross section is only about 3.2 barns. The cross sections employed in the MELEAGER burn-up code make use of the Westcott effective cross section notation $\hat{\sigma} = \sigma_0 (g + rs)$ where the g and s factors reflect the thermal and epithermal absorptions, respectively, and r is the spectral index.

The effective Westcott Am-241 cross section in a Phoenix fuel varies typically from $>$ 3000 barns at start up to $\sim$ 1400 barns at the end of burn-up.

Am-241 forms the Am-242 isomer with a 16-hour halflife with a preferential ratio of 14 to 1 over that of the $\sim$ 100-year halflife Am-242 isomer. Although both 242 isomers have large fission cross sections, it would be necessary to expose the 16-hour halflife Am-242 isomer to a high flux level in order to fission it before β^- decay and formation of Cm-242 occurs. From available information, Cm-242 has an $\sim$ 80-barn thermal neutron absorption cross section and no fission cross section. Thus, the net effect of Am-241 is to act as a strong poison, its daughters and decay products contributing very little to the fissions in thermal spectrum reactors.

TABLE V

TABLE TO ILLUSTRATE THE IMPACT OF AM-241 FORMATION ON EXPOSURE IN PLUTONIUM FUELS WITH HIGH 241 CONCENTRATION

No Pu-241 Decay and No Am-241 Formation

Power		Exposure		Fuel Life	
MW/Ton	w/cc	MWD/T	MWD/cc	Days	Years
10	33	650,000	2,150,000	65,000	178.1
30	100	650,000	2,150,000	21,700	59.5
100	330	650,000	2,150,000	6,500	17.8
300	1,000	650,000	2,150,000	2,160	5.9
Very high		650,000	2,150,000	Very short life	

Am-241 Formation Only (No 241 Decay)

10	33	19,000	63,000	1,900	5.2
30	100	93,000	307,000	3,100	8.5
100	330	620,000	2,050,000	6,200	17.0

With Pu-241 Decay and With Am-241 Formation

10	33	17,000	56,000	1,700	4.7
30	100	74,000	245,000	2,470	6.8
100	330	510,000	1,690,000	5,100	14.0
200	660	< 650,000	< 2,150,000	< 3,250	< 8.9
300	1,000	< 650,000	< 2,150,000	< 2,160	< 5.9
Very high		650,000	2,150,000	Very short life	

Idealized U-235

The rough draft of a report titled Uranium Price Schedules and Bred Fuel Values, HW-72219, has been prepared. The prime objective of this report is to demonstrate the theoretical basis for pricing bred fuels in proportion to the cost of fully-enriched uranium from the diffusion cascade. The cost of burning uranium is highly dependent upon the enrichment and the burn-up fraction, but it is not obvious that the value of bred fuel in an equivalent situation will be independent of these factors. This report demonstrates that if the bred fuel has different chemical properties but the same nuclear properties as U-235 (i.e., an "ideal" U-235), its value would be practically equal to the cost of fully-enriched uranium regardless of the burn-up situation.

Code Development

A routine to extrapolate the fuel cost component curves in QUICK-MINIMIZER was written and is debugged. The expression for the extrapolated curve has the form:

$$Y(X) = A\left\{1 - \exp\left[-B(X-X_0)\right]\right\} + Y(X_0)$$

This expression was formulated to match the polynomial fit generated by MINIMIZER at the "cutoff" enrichment X_0. The arbitrary constants A and B in this expression are determined by specifying that the derivative and the second derivative of the extrapolated curve will match those of the polynomial at $X = X_0$.

When the derivative and the second derivative of the polynomial have the same sign at X_0 the argument of the exponential will be positive. In this case the extrapolation will not damp out but will increase or decrease without limit. The increasing exponential is physically realistic but as the fuel cost cannot decrease without limit this expression will be replaced by a linear junction that will match the value of the polynomial and its derivative at X_0.

An exponential function of the same form will be used in PROTEUS to help fit the isotopic compositions of longlived fuels. The arbitrary constants will be determined by matching the actual data points at the ends and in the center of the range of interest.

Improved data have been prepared for use in testing MINIMIZER subroutine for QUICK fuel costs from GRADED APWR fuel systems in addition to BATCH fueling. New MINIMIZER tests have been devised and are being run to replace data prepared in December, 1961. These include illustrations which will show the effectiveness of MINIMIZER subroutine in interpolating and extrapolating data from both batch and graded fuel systems.

Two sets of identical cases were processed through QUICK, one from MELEAGER data cards, the other from MELEAGER cards run through PROTEUS. The cases compared favorably. Pretreated cases resulted in an average of two percent lower fuel costs from QUICK. Another set of cases is to be processed for a firmer checkout.

A DATAMAKER program has been written and is being debugged. This program will generate data cards for CHAIN and should reduce the amount of work and time required for making up CHAIN decks by a significant factor.

OTHER ACTIVITIES

Radionuclide Heat Sources

When all of the possible radionuclides which may have potential for use as heat sources are examined from the standpoint of a few practical criteria the list shrinks to a total of only six isotopes. These are: strontium-90, cesium-137, promethium-147, plutonium-238, and uranium-232 (and its daughter, thorium-228).

The criteria for screening or selection which reduces the list to this small number include the following:

1. Half-life in the range of 1 to 100 years.
2. Heat output totalling more than 0.1 watt per gram of the pure isotope.
3. If the heat source is a fission product, the fission yield shall be greater than 0.1 percent.
4. The isotope as delivered for heat source preparation shall not cost more than $2000/initial watt when manufactured on more than a prototype scale. (This allows Pu-238 but excludes tritium.)
5. Candidates must not require processes involving isotopic separation for either target or product material.
6. Sufficient data must be at hand to define a realistic and near future production process. (This excludes materials produced only in accelerators and certain fast reactors.)
7. The product must not involve the use of extremely rare elements such as radium-226, americium-243, or lead-210.
8. Inert gases are unsuitable (Kr-85).
9. When produced by steps of neutron irradiation and chemical processing no more than two steps of each may be required.
10. The thermal neutron cross section of the target shall not be less than two barns.
11. The thermal neutron cross section of the product must not exceed 1500 barns.

Although Sr-90, Cs-137, and Pu-238 have already been selected for a number of heat source applications, this study points up that Pm-147 should be receiving additional emphasis. Although it has a half-life of 2.6 years, which is much shorter than four of the others, it is the only one competitive with Pu-238 in that with aged material (to let Pm-148 decay out) its gamma energy (from bremstrahlung) is sufficiently low to make it possible to avoid the special heavy shielding problems associated with all of the other four isotopes. In addition the study continues to support the position of U-232 and Th-228 as outstanding heat sources.

W. K. Woods

Manager,
Programming

WK Woods:jm

RADIATION PROTECTION OPERATION REPORT FOR THE MONTH OF APRIL 1962

A. ORGANIZATION AND PERSONNEL

Transfers within the Section during the month included R. W. Simmons transferring to Environmental Studies and Evaluation from Radiation Monitoring, and R. J. Beaver transferring to Radiation Monitoring from Environmental Studies and Evaluation, both effective April 2. Roberta K. Lattin was transferred from Contract and Accounting to External Dosimetry effective April 16. Alvin J. Stevens was reactivated effective April 30.

B. ACTIVITIES

Occupational Exposure Experience

On April 7 about 11:00 a.m., a nuclear excursion occurred in the K-9 vessel of the Solvent Extraction hood in the Recuplex facility at the 234-5 Building. At the time of the event, there were 22 employees in the building including four persons in the Recuplex area. Automatic activation of the criticality alarm system alerted all persons in the building of the event. Evacuation was prompt and effective.

Surveys of all persons who were in Z-plant were made at the 200-W First Aid Station shortly after the evacuation. No personnel were found to be contaminated. On the basis of surveys made for neutron activation of Na^{23} in the body and statements made by the involved employees, four employees were transferred to the Kadlec Hospital for further tests and observations.

All of the 24 employees in the Z-plant facility were wearing their film badge dosimeters as prescribed. Analysis of these dosimeters, analysis of blood samples, and examinations in the Whole Body Counter provided early preliminary estimates of the radiation doses received. For the four men in the room where the excursion occurred, their preliminary dose estimates were known four hours after the excursion. For the other 20 persons in the Z-plant facility, their doses were known six hours after the event. The four men who were in the room where the excursion occurred received radiation doses of about 110 rems, 43 rems, 19 rems, and 1.4 rems, respectively. All other persons in the Z-plant facility received a dose of < 1 rem due to the excursion except for one employee who was in a nearby office and received 1.7 rems.

The Hanford Laboratories home phone crash alarm system was activated shortly after the incident. Within one hour after the excursion, initial staffing of the Emergency Control Center and emergency supporting Laboratories' facilities was completed. A major portion of the Laboratories' staff in radiation protection, nuclear physics, and analytical laboratories functions was assigned to duties in support of the stabilization of the Recuplex facility

during April. Throughout the recovery period, normal Hanford radiation protection control limits were in force. The Manager, Radiation Protection, served on the Recuplex Advisory Council during the recovery period after the incident.

Three minor cases of plutonium deposition, each less than one percent of the maximum permissible body burden, were confirmed by bioassay analyses during the month. The total number of plutonium deposition cases that have occurred at Hanford is 288, of which 208 are currently employed.

PRTR personnel were exposed to dosage rates of 1.1 rems/hour, including 1 r/hour, while checking for leaks in A cell while the reactor was operating at 3 Mw. The high-level air alarm on the main exhaust duct in PRTR was activated when the moderator storage tank lid was removed, causing an emission of radioactive gas. Following repositioning of flux monitoring channels, personnel were bioassayed and the maximum result was 10.9 μc T/l. A small particle of irradiated zircalloy, which was found in the storage basin area, showed a dose rate of 26 rads/hour at one inch. The maximum dose rate measured during charge-discharge operations was 1.7 r/hour at a distance of 15 feet from a fuel element. This measurement is approximately twice that previously obtained and is probably caused by the element being removed from the reactor before sufficient radioactive decay. A primary system ion exchanger was replaced in dose rates to 250 mr/hour. Dose rates of 30 and 40 r/hour were noted near the remaining ion exchangers. Twenty employees received doses in excess of 25 mrem from internally deposited tritium. The maximum accrued dose was 460 mrem in a 30-day period.

Extensive decontamination of A cell in the High-Level Radiochemistry Facility was completed without incident. The average radiation background has been reduced from an original 70 rads/hour to approximately 10 mrads/hour. The maximum estimated exposure to personnel involved in the decontamination during the past two months was less than 1 rem. Bioassay and Whole Body Counter data confirm no indication of internally deposited fission products to personnel involved in this work.

Skin contamination to 60,000 c/m with nasal contamination to 40,000 c/m was detected on exit surveys of three employees who had been working with the ball 3X hoppers on the top of the 105-KW reactor. A similar occurrence was reported on another shift the same day for two employees doing identical work. Examination at the Whole Body Counter indicated the presence of I^{131}. Subsequent thyroid counts showed iodine deposition for four of the men that represented from about one percent to twelve percent of the permissible body burden, extrapolated to the day of the incident and with the thyroid the organ of reference. Thyroid checks of seven other 105-KW employees showed no significant I^{131} deposition. No explanation for the presence of the iodine was determined.

A report was received from radiation monitoring at 100-F Area that an IPD power operator had received I^{131} at Virginia Mason Hospital in Seattle. The report indicated that a CP dose rate measurement at the employee's neck showed about 90 mr/hour. Evaluation of the employee's film dosimeter for the 4-week period showed a dose of 1 r gamma. The employee's normal work assignment does not require entry into radiation zones. The necessary documentation to determine his occupational dose for the period was initiated.

A CPD pipefitter received plutonium skin contamination to 100,000 d/m on his forearms while working in the 233-S facility, Redox Operation. Although the employee was wearing surgeon's gloves taped to his overalls, the contamination was apparently transferred to the coverall sleeves and then to the forearms. The skin condition was reported as good after about two hours of decontamination efforts to remove the contamination. Special urine samples were requested for bioassay analysis.

Environmental Experience

Iodine-131 emitted from the Purex stack during the period April 2 to April 12 totaled about 44 curies. The high emission is believed to have resulted from dissolving "green" metal. Varying wind speeds and directions combined with the vagrant characteristic of I^{131} clouds dispersed the contamination so that none was detected on vegetation except in the immediate vicinity of the 200 East Area. A maximum level in milk of about 70 μμc/liter was found in the Ringold area.

Average fallout concentrations at various localities in the Pacific Northwest ranged from 2.4 to 13 $\mu\mu c/m^3$ during the month of April. Sporadic increases noted during the first week of April appeared to be the first evidence of the spring peaks expected from the USSR testing last fall.

Control of air-borne contaminants at the Hot Semiworks continued to be troublesome. Emissions of about 5 millicuries (undetermined beta-emitters estimated to be about half Sr^{90}) occurred on April 23 and April 27. No environmental contamination of significance resulted.

A total of 122 fish was taken from sampling locations on the Columbia River at Priest Rapids, Hanford, Ringold, Richland, Burbank, and McNary Dam. One hundred and thirty-five tissue samples from these fish were submitted to the laboratory for radiochemical analysis.

Ninety-eight produce samples were obtained for radiochemical analysis. These include milk from the Ringold, Riverview, Benton City, Mesa, and Eltopia areas as well as composite samples obtained from the Twin City Creamery. Milk samples totaled 103 gallons. Four pounds of Willapa Bay oysters, two pounds of ground round steak, four pounds of rhubarb, fourteen

pounds of asparagus, sixteen pounds of pasture grass, and fifty-one sets of beef thyroid glands were also obtained.

One flight was made over the project and adjacent areas for background data. Two other flights were made in connection with the Recuplex incident.

The output of P^{32} in reactor effluent remains steady but the quantities of As^{76} and Np^{239} continued to decline. A normal seasonal increase was observed in the quantity of Mn^{56} and Na^{24}.

Studies and Improvements

Electronic equipment for measuring the buildup of particulate activity on the filter sample collected at the Hot Semiworks process stack was installed during the month. Operation to date has been satisfactory.

All equipment has been installed at the 306 Building to sample radioactive contamination that may escape to the atmosphere from the portions of the thorium processing cycle.

Arrangements were completed with the USPHS to obtain samples of Columbia River water at Vancouver and ship them to Hanford. This cooperation between the two organizations saves dispatching a man to Portland every other week to obtain the sample.

General assistance was rendered to Security Operation to expedite the fabrication of the new security credentials. A photo blanking die was designed and an arbor press to hold the die was received. A work order for 90 storage racks capable of holding 18,000 film badge dosimeters was placed after the prototype was found acceptable. The old racks cannot be modified to accommodate the new badge dosimeters.

The automatic densitometer was moved into the soundproof booth in the 3705 Building. The modifications to the densitometer have essentially been completed. Doses to 1,000 mr from radium-gamma radiation can now be read on the densitometer without switching scales. The electric relays were installed on the power line leading to the densitometer that will automatically disconnect the densitometer when there is a power failure or large line fluctuations.

The manufacturer's pre-production Scintran was compared with the prototype. The workmanship and general appearance of the pre-production model was satisfactory. The geometry of the manufacturer's probe was better than the prototype. Nucleonic Instrumentation is evaluating the components and circuit.

C. VISITORS

Visitors consulting with members of the Radiation Protection staff during the month included:

A. M. Menoux - French Atomic Energy, Saclay
A. Barbreau - French Atomic Energy, Fontenay-aux-Roses

Dr. C. A. Paulsen - Consulting Endocrinologist, School of Medicine, University of Washington

Dr. C. C. Lushbaugh - Pathologist, Los Alamos Scientific Laboratory

Dr. G. A. Andrews - Oak Ridge Institute of Nuclear Studies

Stig O. W. Bergstrom - Atomic Energy Commission, Studsvik, Tystberga, Sweden

D. RELATIONS

A total of sixteen members of the Radiological Emergency Staff toured the Civil Defense Emergency Relocation Center located on Rattlesnake Mountain.

A safety and housekeeping inspection of all 300 Area Radiation Monitoring facilities revealed good general safety atmosphere with no significant infractions or deficiencies. Minor improvements suggested during the inspection were corrected promptly.

Radiation Monitoring personnel assigned to PRTR are attending formal classes covering the operational, safety, and radiological aspects of the Gas Loop. Tours were conducted through the Critical Facility for groups of 300 Area monitors. Two radiation orientations, one radiation refresher training course, and one lecture on emergency monitoring were provided to HLO and CE&UO customer components.

A total of eighteen persons attended the Disaster Level Monitoring training course before these sessions were discontinued as a result of the incident at Recuplex. This training program will be resumed at a later date.

E. SIGNIFICANT REPORTS

HW-72691-3 -"Summary of Radiological Data for the Month of March 1962" by R. F. Foster.

HW-73341 - "Calibration of Staplex Portable Air Samplers With an Anemometer" by L. F. Kocher.

HW-73538 - "Monthly Report - April 1962, Radiation Monitoring Operation" by A. J. Stevens.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

External Exposure Above Permissible Limits	April	1962 to Date
Whole Body Penetrating	3	3
Whole Body Skin	3	3
Extremity	2	2
Hanford Pocket Dosimeters		
Dosimeters Processed	5,532	13,368
Paired Results - 100-280 mr	1	22
Paired Results - Over 280 mr	0	3
Lost Results	0	0
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	10,097	38,249
Results - 100-300 mrads	193	1,369
Results - 300-500 mrads	17	142
Results - Over 500 mrads	8	59
Lost Results	18	100
Average Dose Per Film Packet - mrad (ow)	12.97	10.72
- mr (s)	29.65	27.97
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	2,362	5,926
Results - 50-100 mrem	3	4
Results - 100-300 mrem	1	2
Results - Over 300 mrem	0	0
Lost Results	5	8
Fast Neutron		
Film Processed	724	1,650
Results - 50-100 mrem	109	250
Results - 100-300 mrem	154	325
Results - Over 300 mrem	2	3
Lost Results	0	10
*Hand Checks		
Checks Taken - Alpha	52,650	132,103
- Beta-Gamma	65,506	209,335
*Skin Contamination		
Plutonium	45	71
Fission Products	56	181
Uranium	3	11
Tritium	0	0

*Includes two reporting periods from CPD.

Whole Body Counter	Male	Female	April	1962 to Date
GE Employees				
Routine	3	0	3	65
Special	77	4	81	119
Terminal	5	0	5	47
Non-Routine	24	3	27	110
Non-Employees	1	0	1	7
Pre-Employment	0	0	0	3
	110	7	117	351

Bioassay		
Confirmed Plutonium Deposition Cases	3	5*
Plutonium - Samples Assayed	322	1,757
- Results above 2.2×10^{-8} µc/sample	21	74
Fission Product - Samples Assayed	536	2,418
- Results above 3.1×10^{-5} µc/sample	0	14
Uranium - Samples Assayed	169	768
Biological - Samples Assayed	0	172
Strontium - Samples Assayed	50	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	5.3	2.5	55	3.8	1.9	46
Fuels Preparation**	0	0	0	0	0	0
Hanford Laboratories	11.2	5.0	18	11.8	4.1	18
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	4.2	1.7	32	0	0	0

Tritium Samples	Maximum	Count	Total
Urine Samples			
> 5.0 µc/l	61.6	112	
< 1.0 µc/l		30	
Samples Assayed			215
D_2O Samples			
Moderator	628.4 µc/ml	8	
Primary Coolant	116.8 µc/ml	8	
Reflector	408.3 µc/ml	8	
			24
Other Water Samples			
No. 5489 Drum	107.6 µc/ml		178
			417

*The total number of plutonium deposition cases which have occurred at Hanford is now 288, of which 208 are currently employed.

**Samples taken prior to and after a specific job during work week.

Calibrations	Number of Units Calibrated April	Number of Units Calibrated 1962 to Date
Portable Instruments		
CP Meter	923	4,012
Juno	248	1,080
GM	524	2,218
Other	192	804
Audits	110	417
	1,997	8,531
Personnel Meters		
Badge Film	2,668	7,024
Pencils	7,890	12,670
Other	749	1,893
	11,307	21,587
Miscellaneous Special Services	1,769	4,699
Total Number of Calibrations	15,073	34,817

Manager
RADIATION PROTECTION

AR Keene:ljw

FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

A revised submission of the Budget for FY 1964 and Revision of Budget for FY 1963 is being prepared. Research and development proposals previously submitted to HOO-AEC are being changed to agree with totals developed at a DRD budget review in Washington on April 13, 1962. Data submitted to Contract Accounting as a part of the HAPO budget will also be revised to reflect the new amounts.

HAPO requirements for irradiation unit services from IOO-AEC for the remainder of FY 1962 and for FY 1963 and FY 1964 were estimated and submitted to HOO-AEC for inclusion in the Budget for FY 1964 and Revision of Budget for FY 1963.

FY 1962 Financial Plan changes received by HOO-AEC in April and pertinent to Hanford Laboratory programs are as follows:

1. The 02 Program authorization for irradiation unit services was increased by $400,000 to a new total of $691,000 to provide adequate funding for the fiscal year.

2. The 04 Program was increased in total by $38,000, exclusive of irradiation unit services, comprising several individual program changes. Clarification from Washington-AEC is still being sought concerning the control points established for research and development and capital equipment in the Reactor and Fuels Materials Program.

3. The 06 Program was reduced by $50,000 to reflect the voluntary transfer of funds to another site at the Request of Washington-AEC. Although not reflected in the Financial Plan, the Division of Biology and Medicine has approved the diversion of $46,000 from research and development to capital equipment as requested by Hanford Laboratories.

An additional authorization of $84,000 was received from UCLRL in connection with Project Whitney. Total authorization for FY 1962 is now $654,000 including $25,000 for capital equipment procurement.

One special request account covering consultation by K. R. Merckx with APED on mechanical design and performance of nuclear fuel was established during the month and assigned code .4P.

A new "Hi-Spot" cost-budget status report was introduced in April. Its function is provision of details of research and development costs and personnel assignments to sub-section managers several days in advance of regular operating cost report issuance. This report will be employed during periods of the fiscal year when cost control and financial planning warrant. Three monthly reports, "Detail of Research and Development for Product Departments," "Detail of Hanford Laboratories' Research and Development Costs" and "Summary of Process Technology Costs" were issued by EDP equipment for the first time in April covering March costs.

Cumulative costs through April 29, 1962 in connection with the 200-W criticality incident total $15,729. This amount covers technical assistance provided by Hanford Laboratories but excludes administrative personnel and Radiation Protection services. It also excludes work orders issued to HLO components (such as Technical Shops) for specific fabrications or analyses.

General Accounting

While the total number of trips fiscal year-to-date by Hanford Laboratories personnel is running slightly behind the total at the end of April last year, total costs are higher as shown below:

(Amounts in thousands)	FY to Date 1961	1962
Attendance at Meetings	$ 30.0	$ 25.6
All Other	223.8	243.3
	$253.8	$268.9

Following is a summary of the status of letters or agreements covering specific actions requiring AEC concurrence:

No.	Title	Status
AT-221	Seattle World's Fair - Library Exhibit	Approved April 13, 1962
AT-223	University of Washington Primate Research Center - Executive Committee	In process
AT-228	Processing of Film Demonstrating the Alsi Canning Process	AEC returning with suggested alternative
AT-237	Summer Institute in Nuclear Energy for Engineering Faculty	AEC holding

During the month $133,670 of equipment was transferred to classified plant accounts from Equipment Work in Progress and $29,921 from Construction Work in Progress.

A report of results was issued for the physical inventory of movable catalogued equipment in the custody of Chemical Research and Development Operation. Two thousand five hundred and thirty-one (2,531) items valued at $2,204,175 were physically counted. Six items valued at $2,261 were missing and 32 items valued at $11,376 not previously recorded in property records were inventoried. These statistics compare to eight missing items and 37 pickup items in FY 1960 inventory. Analysis of the inventory results indicates that property control within CR&DO has improved as illustrated by the fact that three of the four sub-sections had no missing equipment and custodians appeared to be better informed as to the location of equipment.

Reconciliation of the inventory of Other Special Materials for the quarter ending March 31, 1962 is complete, and results will be reported in May 1962. A follow-up on recommendations and findings of the previous quarterly inventory is being made for inclusion in this report.

HLO material investment at April 1, 1962 totaled $26.2 million as detailed below:

Spare Parts	$ 364
SS Material	24 659
Reactor and Other Special Materials	1 194
(Amounts in thousands)	
	$26 217

Heavy water losses chargeable to operating cost for the month of April amounted to $21,459 comprised of PRTR Heavy Water BPID loss of $18,794 and scrap generated during April valued at $2,665.

The value of heavy water scrap accumulations at April 30, 1962 is $275,064 (22,309.78 lbs.). Approval has been obtained from Savannah River Operation to ship 36 drums valued at $112,553. It is estimated that total scrap returns in FY 1962 will be $210,000.

Eighty-two items valued at $79,859 were received at the Laboratory Equipment and Material Pool during the month of April. This included one furnace valued at $47,369. Nineteen items valued at $11,949 were loaned or transferred in lieu of placement of requisitions, 37 items valued at $19,436 were withdrawn by custodians, 82 items valued at $30,631 were excessed in connection with a review of items held in the storage pool for a period exceeding 24 months with no activity. There are currently 851 items valued at $545,537 physically located in the storage pool.

Materials on hand at the Laboratory Pool at month end consisted of the following:

Beryllium	1 035 gms.	$ 592
Gold	2 519 gms.	3 374
Palladium	2 178 gms.	2 483
Platinum	4 242 gms.	12 386
Silver	6 633 gms.	463
Zirconium	19 054.6 lbs.	309 590
All other material held for the convenience of others		328 888
Total materials		$563 125

We were advised by Contract and Accounting Operation that Yttrium has been reclassified from a Reactor Material to a General Supply item. Contract and Accounting has been requested to obtain a ruling as to whether Yttrium received from Lockland at no cost must be turned over to Stores Operation.

Progress of unitization action on projects is described below:

Completed
- CAH-921 Geological and Hydrological Wells - FY 1961
- CAH-914 Rattlesnake Springs Radioecology Facility

Near Completion
- AEC-167 Plutonium Recycle Test Reactor
- CAH-902 Uranium Scrap Burning Facility

In Progress
- CAH-888 Biology Laboratory Improvement - 108-F Building
- CAH-901 Structural Materials Irradiation Test Equipment - ETR

New Money Authorized HLO
- CAH-822 Pressurized Gas Cooled Loop Facility $35 000

Physical Completion Notices Issued
- CGH-842 Critical Facility
- *CAH-888 Biology Laboratory Improvements
- CAH-896 Stress-Rupture Testing Facility
- CAH-902 Uranium Scrap Burning Facility
- CAH-914 Rattlesnake Springs Radioecology Facility
- CAH-919 Air Conditioning - 314 Building
- CAH-921 Geological & Hydrological Wells - FY 1961
- **CAH-924 200-KW Induction Heating System - 306 Building
- AEC-167 Plutonium Recycle Test Reactor

Physical Completion Notices Issued (Continued)

*Physical Completion Notice for AEM Services has not been issued by GE. HOO-AEC Physical Completion Notice contains an accrual for $60,500, the amount authorized General Electric by work authority.

**AEM Services only.

Construction Completion and Cost Closing Statements Issued
CAH-914 Rattlesnake Springs Radioecology Facility
CAH-921 Geological and Hydrological Wells - FY 1961

The following contracts were processed during the month:

CA-340	Rutherford Aris
CA-326	M. E. Ensminger
DDR-147	Battelle Memorial Institute
CA-331	William M. Dickson
SA-167	Supplement - Turco Products, Inc.

New and revised OPGs issued are shown below:

OPG No.	New	Revised	Title
1.4		x	Reporting of Accidents or Special Incidents
22.1.1		x	Hanford Laboratories Organization
22.1.7		x	Finance & Administration Organization
3.1.5		x	Assistance to Families of Deceased Employees
3.2.4		x	Overtime
3.3.3		x	Strike Misconduct
3.4.4		x	Inquiries Relative to Employees
3.4.9		x	Reactivation of Employees
3.4.10		x	Part-time and Temporary Employees
3.4.12		x	Transfers and Promotions
3.4.13		x	Procurement of Employees
3.4.14		x	Removal from Payroll
33.6.1	x		Publication of Technical Articles or Books
55.1.2		x	Monthly Attendance Report
6.6		x	Conflicts of Interest
7.4		x	Security Education Program
7.6		x	Infractions of Security Regulations

OPG No.	New	Revised	Title
7.7		x	Procurement of Classified Services or Materials
7.11		x	Control of Code Designations
7.13		x	Authorization and Control of Visits and Visitors
7.16		x	Classified Document Removals and Control of Contraband
7.17		x	Movement of Classified and Radioactive Material

Personnel Accounting

Conflict of interest acknowledgments were received from 94 per cent of all exempt employees. Supervision has been requested to obtain acknowledgments from the remaining six per cent.

Following are the payroll statistics for the month of April 1962:

Number of HLO Employees Changes During Month	Total	Exempt	Nonexempt
Employees on payroll at beginning of month	1 437	688	769
Additions and Transfers In	11	3	8
Removals and Transfers Out	16	14	2
Employees on payroll at end of month	1 432	657	775

Overtime Payments During Month	April	March
Exempt	$ 5 443	$ 3 367
Nonexempt	25 641	20 842
Total	$31 084	$24 209

Gross Payroll Paid During Month		
Exempt	$ 622 613	$ 629 995
Nonexempt	426 457	500 884
Total	$1 049 070	$1 130 879

Participation in Employee Benefit Plans at Month End	April Number	April Per Cent	March Number	March Per Cent
Pension	1 292	99.4	1 289	99.5
Insurance Plan-Personal	361		361	
-Dependent	1 062	99.8	1 067	99.8
U.S. Savings Bonds				
Stock Bonus Plan	89	38.9	88	38.9
Savings Plan	74	5.2	75	5.2
Savings and Security Plan	1 079	89.7	1 079	89.3
Good Neighbor Fund	964	67.3	969	67.4

Insurance Claims Employee Benefits	Number	Amount	Number	Amount
Life Insurance	-0-	$ -0-	-0-	$ -0-
Weekly Sickness and Accident	10	873	15	1 033
Comprehensive Medical	57	4 287	73	5 096
Dependent Benefits				
Comprehensive Medical	128	10 023	141	11 939
Total	195	$15 183	229	$18 068

TECHNICAL ADMINISTRATION

Employee Relations

Fourteen non-exempt employment requisitions were filled during April; 25 remain to be filled.

A revision of HAPO OPG 3.5.2, Tuition Refund Program, was prepared and submitted for approval.

Professional Placement

Advanced Degree - Four Ph.D. applicants visited HAPO for employment interviews. Four offers were extended; one acceptance and three rejections were received. Current open offers total six.

BS/MS - Fifty program offers and ten direct placement offers were extended; offers accepted - 27 and four, rejections received - 54 and five, respectively. Current open offers total 95.

Technical Graduate Program - Six Technical Graduates were placed on permanent assignment; one new member was added to the rolls and one terminated. Current program members total 38.

Technical Information

The quarterly Technical Interchange Report was completed in a revised format.

The Library's roster of local individuals capable of translating or speaking fluently in foreign languages was made current.

ECONOMIC EVALUATIONS

The cost logic of a fuel element fabrication model was developed with Programming Operation personnel. Purpose of the model, to be programmed for computer application, is to calculate fuel element production costs with reasonable accuracy. Specifically the code will be applied to test conclusions and to compare cost results of varying production assumptions of Hanford Laboratories proposed U-235 and Pu fuel fabrication plant concepts.

Input cost classifications consist of the following:

Nuclear Material (accountability for each type)	$/Fuel Element
Non-Nuclear Material (accountability for each type)	$/Fuel Element
Variable Labor Cost	$/Hour
Equipment Cost (a variable cost)	$/Fuel Element
Space Occupancy (a fixed cost)	$/Year
Fixed Manufacturing Cost	$/Year
Other Fixed Cost	$/Year

Additional information required for each process step:

Step No., Description, Average time in the step, Recycle rate and to what step, Material value loss, Use Charge Rate, and Interest Rate on Working Capital.

PROCEDURES

A study report of materials administration in the Laboratories was issued by Business Systems Development Operation personnel.

FACILITIES ENGINEERING

Projects

At month's end Facilities Engineering Operation was responsible for 11 active projects having total authorized funds in the amount of $2,721,600. The

total estimated cost of these projects is $7,730,000. Expenditures on these projects through March 31, 1962 were $1,274,000. The appended report details project status.

The following summarizes the status of project activity in April:

Number of authorized projects at month's end - - - - - - - - 11
Number of new projects authorized - - - - - - - - - - - - - 0
Projects completed - 0

New projects submitted to the AEC - - - - - - - - - - - - - 2
CAH-962, Low Level Radiochemistry Building
CAH-963, Geological & Hydrological Wells - FY 1962

New Projects awaiting AEC authorization - - - - - - - - - - 4
CAH-917, Field Service Center - Atmospheric Physics
CAH-959, Graphite Machining Shop
CAH-962, Low Level Radiochemistry Building
CAH-963, Geological & Hydrological Wells - FY 1962

Project proposals complete or nearing completion - - - - - 2
CAH-958, Plutonium Fuels Testing and Evaluation Labs.--308 Bldg.
--- Facility for Radioactive Particle Inhalation Studies

Services

Engineering services were provided during April on the following activities with satisfactory progress shown on each:

Laboratory Modifications - 325, 321A and 3706 Buildings
Controlled Environment Facility - 108 Building
Split-Half Machine - Critical Mass Laboratory
Salt Bath Furnace Alarm - 306 Building
Cell Door Drive Modification - Whole Body Counter (747 Building)
Compressor and Motor Control Center - 209-E Building
Electrical Load Study - 108-F Building
Relocate Induction Heating Equipment - 314 Building

Twenty-nine equipment requisitions totaling $25,000 were issued during the month. The total value of material and equipment procurement in process is $500,000.

Plant Engineering effort was expended on:

Pressure system code work on: H-1 Loop, Burst Test Facility, PRTR "B" Cell Rupture Loop, 222-U Vessel and 189-D piping.
Five-year power forecast including firm and dump power usage figures for the coming two years.
3702 Building ventilation system maintenance and air balance.
309 Building fire alarm bell.

Maintenance and Operation

Costs for March were $159,188. Costs to date ($1,367,805) are 98.7% of forecasted expenditures. Improvement maintenance costs for March were $2,465.

The following tabulation summarizes waste disposal operations:

	March	February
Concrete Barrels	21	4
Loadluggers-Hot Waste	5	3
Crib Waste	260,000 gal.	220,000 gal.

Drafting

The equivalent of 178 drawings were completed during the month.

Major jobs in progress include fission product packaging, 280 ton extrusion press installation, PRTR as-builts, fuel rod ultrasonic vibrator modifications, A-C Columns - 321 Building (CGH-951), PRTR fuel element rupture test facility and shim rod housing, equipment modifications for fabrication of seven-rod PRTR fuel element, cladding cutter assembly for PRTR and neutron accelerator specimen support mechanism.

Construction

There were 80 existing J. A. Jones Company orders at the beginning of the month with a total unexpended balance of $142,081. One hundred sixty-eight new orders, three supplements and adjustments for underruns amounted to $118,764. Expenditures during the month on Hanford Laboratories work were $77,434. (Includes C. O. Cost.) Total J. A. Jones backlog at month's end was $183,411.

Construction activities completed during April were:

141-C Building - Install individual feeding stations
309 Building - Install shielding wall
325 Building - Re-install Dynapak moved from 306 Building

Manager
Finance and Administration

W Sale:whm

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 73514

DATE 4-30-62

PROJ. NO.	TITLE	FUNDING
CAE-822	Pressurized Gas Cooled Facility	4141 Operating

AUTHORIZED FUNDS					
$ 1,170,000	DESIGN $ 43,000	AEC $ 15,000	COST & COMM. TO 4-15-62	$ 1,104,422
	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	2-2-62*	DESIGN 4-29-60	
	CONST. 10-17-60	DIR. COMP. DATE 6-30-62	CONST 6-30-62	

ENGINEER

TRAAO-MERO - 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	93	91
PF	1.4	0	0
CPFF	22.1	99	99
FP	6.6	100	100
Govt.Eq.	69.9	93	90

SCOPE, PURPOSE, STATUS & PROGRESS

Piping stress calculations for new heater not yet completed by vendor. Effort is being made to improve delivery date on this unit.

Bristol-Siddeley is proceeding with window-pad bearings. Test by May 11, 1962 of first machine is still on schedule.

*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		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

1234556

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73514

DATE 4-30-62

PROJ. NO.	TITLE	FUNDING
CGH-857	Physical & Mechanical Properties Testing Cell - 327 Bldg.	0290

AUTHORIZED FUNDS				COST & COMM. TO 4-15-62	
$ 460,000	DESIGN $ 45,000	AEC $ -			$ 252,436
	CONST. $ 415,000	GE $ 460,000		ESTIMATED TOTAL COST	$ 460,000

STARTING DATES				EST'D. COMPL. DATES		PERCENT COMPLETE		
	DESIGN 11-2-59	DATE AUTHORIZED 9-22-61*		DESIGN 3-15-61		WT'D.	SCHED.	ACTUAL
	CONST. 2-12-62	DIR. COMP. DATE 12-15-62		CONST. 12-15-62				
ENGINEER					DESIGN	100	100	100
FEO - KA Clark					TITLE I			
MANPOWER		AVERAGE	ACCUM MANDAYS		GE-TIT. II	100	100	100
FIXED PRICE					AE-TIT. II			
COST PLUS FIXED FEE			34					
PLANT FORCES					CONST.	100	2**	2**
ARCHITECT-ENGINEER					PF			
DESIGN ENGINEERING OPERATION			760		CPFF	18	9	9
GE FIELD ENGINEERING					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***
	19	Value	203,000

*Original authorization for design was October 1, 1959.
**Based on revised schedule submitted to HOO-AEC for approval 4-13-62.
***Includes delivery charges, inspection and contingency.

1234557

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73514

DATE 4-30-62

PROJ. NO.	TITLE	FUNDING
CGH-858	High Level Utility Cell - 327 Building	0290

AUTHORIZED FUNDS					
$ 400,000	DESIGN $ 50,000	AEC $ -	COST & COMM. TO 4-15-62	$	369,121
	CONST. $ 350,000	GE $ 400,000	ESTIMATED TOTAL COST	$	400,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
	DESIGN 11-1-59	4-6-61*	DESIGN	2-15-61
	CONST. 5-15-61	DIR. COMP. DATE 6-1-62	CONST	6-1-62

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE	6	860
PLANT FORCES		
ARCHITECT-ENGINEER		35
DESIGN ENGINEERING OPERATION		930
GE FIELD ENGINEERING	1	45

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	80	85
PF			
CPFF	100	80	85
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 $205,625
Number of purchase orders placed 12 Value 205,625

The Flat Tensile Sample Former and two manipulators are other items of equipment not delivered. They are expected by 5-15-62.

The installation of cell plugs and windows is complete.

Mechanical and electrical work is expected to be complete by 5-15-62.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73514

DATE 4-30-62

PROJ. NO.	TITLE	FUNDING
CAH-866	Shielded Analytical Laboratory - 325-B Building	61-a-1

AUTHORIZED FUNDS					
$ 700,000	DESIGN $ 60,000	AEC $ 546,500	COST & COMM. TO 4-15-62	$ 129,085 (GE)
	CONST. $ 640,000	GE $ 153,500	ESTIMATED TOTAL COST	$ 700,000

STARTING DATES			ESTD. 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	10	1560
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	90	55
PF	3	1	1
CPFF	2	0	0
FP	95	100	55

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.

Installation of cell access doors and sleeve assemblies was completed. Outside form for high density concrete wall has been started.

*Original authorization for preliminary design was August 12, 1959.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73514

DATE 4-30-62

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 4-15-62	$ 535,269 (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	10	2320
COST PLUS FIXED FEE		
PLANT FORCES	5	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	96	94
PF	2	100	50
CPFF	57	100	97
FP (1)	10	100	100
(2)	31	85	88

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.

*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

1234560

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73514

DATE 4-30-62

PROJ. NO.	TITLE	FUNDING
CAH-888	Biology Laboratory Improvements	60-h-1

AUTHORIZED FUNDS				COST & COMM. TO 4-15-62	$ 53,088 (GE)
$ 420,000	DESIGN $ 44,000	AEC $ 359,500		ESTIMATED TOTAL COST	$ 420,000
	CONST. $ 376,000	GE $ 60,500			

STARTING DATES			EST'D. COMPL. DATES	
DESIGN 8-8-60	DATE AUTHORIZED 4-18-61*		DESIGN 3-31-61	
CONST. 7-10-61	DIR. COMP. DATE 3-31-62		CONST. 5-30-62	

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.

Hauserman partitions are installed. When doors are taped, air balancing will be done. Floor tile has been received in Seattle and laying remaining 2nd floor covering will start week of April 23rd.

All tie-ins are completed.

Painting contractor has completed painting outside surface of blockwalls.

Trip to Allied Engineering revealed need of additions to the operating mechanism. B. D. Bohna and General Electric have determined and agreed on items of work to be included. These items will be presented to Allied Engineering through Purchasing Operation. Allied Engineering estimates this work will require approximately ten days after notification to proceed.

*Original authorization for design was May 3, 1960.

1234561

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 73514

DATE 4-30-62

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 4-15-62	$ 321,952
	CONST. $ -0-	GE $ 385,000	ESTIMATED TOTAL COST	$ 5,100,000***

STARTING DATES			ESTIMATED COMPL. DATES	
DESIGN 3-15-61	DATE AUTHORIZED 10-27-61**		DESIGN 8-15-62	
CONST. 6-1-62*	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	70	70
TITLE I	11	100	100
GE-TIT. II	89	66	66
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.

The revised design schedule has been approved by AEC.

A revised project proposal is being routed in General Electric Company for approval.

Design is continuing on all phases as scheduled.

Satisfactory design progress was made during the month. A total of 206 drawings have been issued for comment and 29 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.

1234562

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73514

DATE 4-30-62

PROJ. NO.	TITLE	FUNDING
CAH-917	Field Service Center - Atmospheric Physics	61-J

AUTHORIZED FUNDS				
$	DESIGN $	AEC $	COST & COMM. TO	$
	CONST. $	GE $	ESTIMATED TOTAL COST	$ 154,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES		PERCENT COMPLETE			
						WT'D.	SCHED.	ACTUAL
DESIGN	6-30-62		DESIGN	11-1-62*	DESIGN	100		
CONST.	12-15-62*	DIR. COMP. DATE	CONST.	9-1-63*	TITLE I			
					GE-TIT. II			
					AE-TIT. II			
					CONST.	100		
					PF			
					CPFF			
					FP			

ENGINEER

FEO - JT Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facilities necessary to conduct atmospheric physics research and development programs.

The project proposal was submitted to the AEC on January 23, 1961.

There has been no change in the status of this report for some time.

*Based on AEC authorization by 5-15-63.

PROJ. NO.	TITLE	FUNDING
CAH-922	Burst Test Facility for Irradiated Zirconium Tubes	62-k

AUTHORIZED FUNDS				
$ 29,600	DESIGN $ 29,600	AEC $	COST & COMM. TO 4-15-62	$ 29,600
	CONST. $	GE $ 29,600	ESTIMATED TOTAL COST	$ 250,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES		PERCENT COMPLETE			
						WT'D.	SCHED.	ACTUAL
DESIGN	11-7-61	10-23-61	DESIGN	5-31-62	DESIGN	100	91	85
CONST.	6-15-62	DIR. COMP. DATE To be established at a later date	CONST.	12-1-62	TITLE I			
					GE-TIT. II	57	88	80
					AE-TIT. II	43	95	90
					CONST.	100		
					PF			
					CPFF			
					FP			

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER - Bovay Engineers	3	160
DESIGN ENGINEERING OPERATION	2	100
GE FIELD ENGINEERING		

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 review of the design for pressure vessels and piping by the third-party inspector has been requested.

The estimate of project cost is more than $20,000 over the project proposal amount.

Efforts are being exerted to reduce the project cost to the minimum consistent with safety, utility, operational adequacy and sound construction practices.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73514

DATE 4-30-62

PROJ. NO.	TITLE	FUNDING
CAH-927	Additions to the 271-CR Building-Waste Treatment Demonstration Facility	61-j

AUTHORIZED FUNDS					
$ 92,000	DESIGN $ 11,000	AEC $ 76,300	COST & COMM TO 4-15-62	$ 14,896 (GE)
	CONST. $ 81,000	GE $ 15,700	ESTIMATED TOTAL COST	$ 92,000

STARTING DATES			EST'D. COMPL. DATES		PERCENT COMPLETE			
DESIGN 6-15-61	DATE AUTHORIZED 5-15-61		DESIGN 2-5-62			WT'D.	SCHED.	ACTUAL
CONST. 2-15-62	DIR. COMP. DATE 7-31-62		CONST. 7-31-62		DESIGN	100	100	100
ENGINEER					TITLE I			
FEO - KA Clark					GE-TIT. II			
MANPOWER	AVERAGE	ACCUM MANDAYS			AE-TIT. II	100	100	100
FIXED PRICE								
COST PLUS FIXED FEE	3	110			CONST.	100	16	16
PLANT FORCES					PF			
ARCHITECT-ENGINEER		150			CPFF	33	75	90
DESIGN ENGINEERING OPERATION					FP	67	0	0
GE FIELD ENGINEERING								

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.

Directive No. AEC-194, Mod. 2, dated 4-5-62 authorized additional funds in the amount of $12,000 for a new project total authorization of $92,000 and extension of the completion date to July 31, 1962.

The J. A. Jones Company has completed underground piping and foundation work. The only work remaining for them is grading of the site.

The fixed price contract was awarded on 4-10-62.

The contractor for this portion of the facility started work at the site 4-24-62.

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SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73514

DATE 4-30-62

PROJ. NO.	TITLE	FUNDING
CAH-936	Coolant Systems Development Laboratory 1706-KE Building Addition	62-k

AUTHORIZED FUNDS					
$ 130,000	DESIGN $ 9,000 CONST. $ 121,000	AEC $ 115,000 GE $ 15,000	COST & COMM. TO 4-15-62 ESTIMATED TOTAL COST	$ 14,687 $ 130,000	

STARTING DATES			ESTD. COMPL. DATES		
DESIGN	9-8-61	DATE AUTHORIZED 4-5-62*	DESIGN	1-1-62	
CONST.	5-1-62	DIR. COMP. DATE 10-31-62	CONST.	10-31-62	

ENGINEER

FEO - KA Clark

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		166
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			
CONST.	100	0	0
PF			
CPFF			
FP			

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.

A contractor orientation meeting was held on 4-20-62 at which the contract requirements, including the special conditions were discussed.

Arrangements were made for start of construction on 5-1-62.

PROJ. NO.	TITLE	FUNDING
CGH-951	A-C Column Facility - 321 Building	0290

AUTHORIZED FUNDS					
$ 55,000	DESIGN $ 5,000 CONST. $ 50,000	AEC $ -0- GE $ 55,000	COST & COMM. TO 4-15-62 ESTIMATED TOTAL COST	$ 4,658 $ 55,000	

STARTING DATES			ESTD. COMPL. DATES		
DESIGN	1-30-62	DATE AUTHORIZED 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.	ACTUA
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II	0		
CONST.	100	12	12
PF	100	12	12
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide a closely integrated "A" column in series with the relocated "C" column to permit the development of a mathematical model for the mass transfer of uranium, as well as the exploration of the possibilities of computer optimization of a combined "A-C" extraction battery.

Specifications and purchase requisitions for procured items are being prepared.

Removal of some of the obsolete piping has been completed.

Installation of intermediate level platforms has started. Relocation of "C" column is in progress.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73514

DATE 4-30-62

PROJ. NO.	TITLE	FUNDING
CGH-955	Reactivation of the H-1 Loop - 105-H Building	0490

AUTHORIZED FUNDS				
$ 10,000	DESIGN $ 10,000 / CONST. $	AEC $ / GE $ 10,000	COST & COMM. TO 4-15-62	$ 10,000
			ESTIMATED TOTAL COST	$ 105,000

STARTING DATES			EST'D. COMPL. DATES		PERCENT COMPLETE		
DESIGN 4-15-62	DATE AUTHORIZED 3-29-62		DESIGN 8-30-62		WT'D.	SCHED.	ACTUAL
CONST. 7-15-62	DIR. COMP. DATE		CONST 12-15-62				
ENGINEER				DESIGN	100	NS	1
FEO - OM Lyso				TITLE I			
MANPOWER	AVERAGE	ACCUM MANDAYS		GE-TIT. II	100	NS	1
FIXED PRICE				AE-TIT. II			
COST PLUS FIXED FEE							
PLANT FORCES				CONST.	100		
ARCHITECT-ENGINEER				PF			
DESIGN ENGINEERING OPERATION				CPFF			
GE FIELD ENGINEERING				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 and meetings have been held with the customer and design to define area of work and coordinate proposed design concept.

PROJ. NO.	TITLE	FUNDING
CGH-957	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 4-15-62	$ 4,000
			ESTIMATED TOTAL COST	$ 40,000

STARTING DATES			EST'D. COMPL. DATES		PERCENT COMPLETE		
DESIGN 4-23-62	DATE AUTHORIZED 3-21-62		DESIGN 6-15-62		WT'D.	SCHED.	ACTUAL
CONST. 7-1-62*	DIR. COMP. DATE 11-1-62		CONST. 11-1-62				
ENGINEER				DESIGN	100	NS	5
FEO - DS Jackson				TITLE I			
MANPOWER	AVERAGE	ACCUM MANDAYS		GE-TIT. II	100	NS	5
FIXED PRICE				AE-TIT. II			
COST PLUS FIXED FEE							
PLANT FORCES				CONST.	100		
ARCHITECT - ENGINEER				PF			
DESIGN ENGINEERING OPERATION				CPFF			
GE FIELD ENGINEERING				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.

The project proposal was submitted to the Commission for approval on March 2, 1962.

Directive HW-535, dated March 21, 1962, authorized total project funds in the amount of $40,000.

Design was started on April 23, 1962.

*Construction may be initiated at an earlier date to take advantage of the presence [illegible] 300 Area prior to July, 1962.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW-73514

DATE 4-30-62

PROJ. NO.	TITLE	FUNDING
CGH-958	Plutonium Fuels Testing and Evaluation Laboratory-308 Bldg.	62-k

AUTHORIZED FUNDS	DESIGN $	AEC $	COST & COMM. TO	$
$	CONST. $	GE $	ESTIMATED TOTAL COST	$ 132,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES		PERCENT COMPLETE			
	DESIGN 7-1-62*			DESIGN 10-1-62*		WT'D.	SCHED.	ACTUAL
	CONST. 10-1-62*	DIR. COMP. DATE		CONST. 2-15-63*	DESIGN	100		
ENGINEER					TITLE I			
FEO - OM Lyso					GE-TIT. II			
MANPOWER		AVERAGE	ACCUM MANDAYS		AE-TIT. II			
FIXED PRICE								
COST PLUS FIXED FEE					CONST.	100		
PLANT FORCES					PF			
ARCHITECT-ENGINEER					CPFF			
DESIGN ENGINEERING OPERATION					FP			
GE FIELD ENGINEERING								

SCOPE, PURPOSE, STATUS & PROGRESS

This project 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.

Expansion of the justification for subject work is being prepared for submission to the AEC.

*Based on AEC authorization by June 1, 1962.

PROJ. NO.	TITLE	FUNDING
CAH-959	Graphite Machining Shop - 300 Area	62-k

AUTHORIZED FUNDS	DESIGN $	AEC $	COST & COMM. TO	$
$	CONST. $	GE $	ESTIMATED TOTAL COST	$ 105,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES		PERCENT COMPLETE			
	DESIGN 7-1-62*			DESIGN 10-15-62*		WT'D.	SCHED.	ACTUA
	CONST 12-15-62*	DIR. COMP. DATE		CONST. 7-1-63*	DESIGN	100		
ENGINEER					TITLE I			
FEO - OM Lyso					GE-TIT. II			
MANPOWER		AVERAGE	ACCUM MANDAYS		AE-TIT. II			
FIXED PRICE								
COST PLUS FIXED FEE					CONST.	100		
PLANT FORCES					PF			
ARCHITECT - ENGINEER					CPFF			
DESIGN ENGINEERING OPERATION					FP			
GE FIELD ENGINEERING								

SCOPE, PURPOSE, STATUS & PROGRESS

This project 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.

Additional justification has been requested for the proposed work.

*Based on AEC approval by June 1, 1962.

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SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. -- Hanford Laboratories

HW- 73514

DATE 4-30-62

PROJ.NO.	TITLE	FUNDING
CAH-963	Geological & Hydrological Wells - FY-1962	62-k

AUTHORIZED FUNDS $	DESIGN $	AEC $	COST & COMM. TO $
	CONST. $	GE $	ESTIMATED TOTAL COST $ 80,000

STARTING DATES		DATE AUTHORIZED	EST'D. COMPL. DATES	
	DESIGN 5-18-62*			DESIGN 6-1-62*
	CONST. 7-15-62*	DIR. COMP. DATE		CONST. 4-1-63*

ENGINEER

FEO - HE 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		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project involves the continued drilling of special hydrological research, test and monitoring wells.

Project proposal submitted to the Commission for approval April 23, 1962.

*Based on Commission Approval by 5-3-62.

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			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

1234568

TEST REACTOR AND AUXILIARIES OPERATION

APRIL 1962

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output was 1044 MWD for a plant efficiency of 49.7% and a total experimental time efficiency of 51.7%. Accumulated exposure through April 30, is 7261 MWD. Additional exposure information is as follows:

Maximum UO_2 exposure/element	2240 MWD/TU
Average UO_2 exposure/element	1518 MWD/TU
Maximum Pu-Al exposure/element	73.7 MWD
Average Pu-Al exposure/element	44.5 MWD
Maximum Moxtyl exposure/element	12.6 MWD
Average Moxtyl exposure/element	10.2 MWD

The ninth refueling was performed on April 2. Three new Pu-Al elements containing high exposure plutonium and three new elements containing a mixture of PuO_2 and UO_2 (Moxtyl) were charged in place of UO_2 elements. Additionally, an LX Pu-Al element and an additional Moxtyl element were charged in mid-month. The performances of the HX and the Moxtyl elements have been satisfactory, with no unusual operating problems.

The status of the various test elements at the end of April is shown below:

PRTR Test No.	Tube Location	Element Number	Description	Date Charged	Date Discharged	Accumulated MWD
5	1653	1501	UO_2-Tubular	11-3-61	--	60.3
10	1051	1082	UO_2-Hot Swage	11-3-61	--	35.0
10	1857	1067	UO_2-Vipac	11-3-61	--	38.9
13	1253	5092	Pu-Al Instrumented	12-3-61	--	54.7
13	1544	5093	Pu-Al Instrumented	12-3-61	--	47.3
13	1556	5095	Pu-Al Instrumented	1-17-62	4-18-62	41.3
13	1247	5051	Pu-Al Instrumented	1-17-62	2-7-62	13.1
13	1847	5052	Pu-Al Instrumented	1-17-62	4-2-62	30.7
14	1146	5096	Moxtyl-Swaged	4-2-62	--	12.6
14	1156	5097	Moxtyl-Swaged	4-2-62	--	10.6

The first of a series of gamma irradiations under PRTR Test Number 40 (Graphite Oxidation Tests) was completed.

D_2O and Helium losses were 1540 lb. and 189,400 scf, respectively.

There were nine reactor scrams during the month, five occurring at power levels of 60 MW or higher. Six were the result of erratic operation of the log flux instrumentation, two were caused by a faulty relay in the shutdown valve circuit, and one was the result of pressurizer high liquid level during a scram recovery.

There was one two-day scheduled outage at the beginning of the month during which Primary Pump No. 3 seal was replaced. Two unscheduled outages were caused by high helium and D_2O losses. The sources were found and corrected. On one of these outages, the primary system ion exchanger was replaced because of breakthrough.

Equipment Experience

Primary pump #3 was replaced because of seal failure. The pump had been in operation about three months. Pressurizer level switches were rewired because of insulation imbrittlement on the old wiring.

A series of repairs were made on and about a pressurizer relief valve. The ring flanges leaked and required remachining, then an original weld leaked and required replacement, and then the valve spring failed. The spring apparently failed during hydrostatic testing following the weld replacement.

A leak-through failure of the control valve on the primary coolant clean-up system required replacement of the stem and seat.

Replacement of the primary ion exchanger was made difficult by the presence of weld protrusions which interfered with operation of the internal cutters used to disengage the ion exchanger. Radiation levels (30 R 4" from the top of the ion exchanger) were lower than expected and permitted direct sight guidance during replacement work.

Programmed maintenance required 460 man-hours or 8.7% of available craft hours.

Improvement Work Status (Significant items)

Work Completed:

Installation of new helium unloading station valves.
Revised secondary seals on moderator, reflector and top and bottom shield pumps.
Additional shielding for storage basin wall at -12' -0" level.

Work Partially Completed:

Safety circuit ground and low voltage detector.
Outlet nozzle cap modification (now 50% complete),
Automatic condensate drain from HX-7.
Boiler feedwater auxiliary control valve.
Relocation of pressurizer level transmitters.
Modification of primary pump power cables.
Primary pump bowl vent line.
RTD terminal board modification.
Gas bottle storage facility modification.
Fueling vehicle hoist modification
Reactor core liquid level instrumentation.

Design Work Completed:

Reroute D_2O injection line.
Primary oxygen analyzer modification.
Install dual valves on helium stack lines.
Enlarge chemical feed system.
Decontamination facility.
Shim rod readout modification
13.8 KV switch installation.
Prototype replacement inlet gas bellows.
Chain Barricade.

Design Work Partially Completed:

Control room ventilation.
Additional fuel storage and examination layout.
Oil storage building.
Outlet nozzle bracing.
Boiler feed pump seals.
D_2O recovery system.

Process Engineering and Reactor Physics

A preliminary study was made of the hazards associated with the transition from the 3-zone UO_2-Pu-Al loading to a 2-zone UO_2-Moxtyl loading. It was concluded that the proposed change would be safer with respect to an increase in the negative temperature coefficient and would not change the delayed neutron fraction significantly.

PRTR Test Number 26 (Shim Rod Calibration) was performed involving calibration of one typical shim rod from each of the three rings.

PRTR Test Number 27 (Moderator Level Coefficient) indicated a moderator level coefficient of 0.53 mk/inch at 103 inches moderator level.

An attempt was made during the performance of PRTR Test Number 42 (Full Design Operation of the Steam Generator) to raise the steam generator pressure from its current operating pressure of 340 psig to full design pressure of 410 psig. The attempt was terminated at a pressure of 365 psig due to the primary coolant bulk outlet temperature reaching its maximum limit of 530 F and two process tube outlet temperatures approached their maximum limits of 540 F.

Procedures

Revised Operating Procedures Issued	13
Revised Operating Standards Issued	10
Temporary Deviations to Operating Standards Issued	5
Revised Process Specifications Accepted for Use	0
Maintenance Manuals Issued	1
Maintenance Procedures Issued	1

Drawing As-built status	April	Total
Approved for as-built	214	521
Ready for approval		31
In drafting		100
Voided		45

Personnel Training was as follows:

Qualification subjects	358	Man-Hours
Specifications, Standards, Procedures	165	
Fueling Vehicle	30	
Maintenance Craftsmen	100	
	653	Man-Hours

Status of Qualified Personnel at Month-End:

Qualified Reactor Engineers	10
Qualified Technicians	6
Qualified Technologists	18

Project Items

Second Communication System - 90% complete.
Fuel Element Examination Facility - Mechanical reassembly completed. Electrical repairs in progress.
Water Control Laboratory - All equipment received and installation started.

Plutonium Recycle Critical Facility

Project Exceptions

Safety rod cadmium plating was tested and found to be satisfactory. The rods were installed and drop tested satisfactorily. The Log-N channels triple trip unit was returned from the vendor and tested satisfactorily. Design test status is as follows:

Electrical	99%
Instrumentation	90%
Fuel Handling	20%
Moderator System	75%
Thimble Coolant	50%

Operation

Work continued on startup preparations. An orientation session was conducted for the IPD team which will conduct the audit on startup preparations.

Preliminary planning was started on the H_2O moderated second generation experimental program. Current efforts include scope, budget, and scheduling definitions.

PRCF orientation sessions were completed for 8 PRTR assigned personnel. A total of 40 PRCF man-hours was devoted to training.

Fuel Element Rupture Test Facility

Project Status (Project CAH-867)

Overall construction is estimated at 94% complete. CPFF work is 97% complete and the water plant is 88% complete. Minor design changes were initiated to alleviate potentially hazardous conditions. Design for shielding of the piping and heat exchangers is approximately 60% complete.

Operation

Work continued on the instrument manual visual training aids, the technical manual and valve lists.

GAS COOLED POWER REACTOR PROGRAM

Project Status (CAH-822)

The project is 91% complete. The vendor started fabrication of the heater. The blower vendor indicated satisfactory progress toward mid-May testing of the blowers.

Operation

Formal training of all PRTR operating personnel was initiated April 19. Some work continued on procedure preparation and manuals, etc.

TECHNICAL SHOPS

Total productive time for the period was 23,790 hours. This includes 15,593 hours performed in the Technical Shops, 5,960 hours assigned to Minor Construction, 1,419 hours assigned to off-site vendors, and 818 hours to other project shops. Total shop backlog is 22,582 hours, of which 70 percent is required in the current month with the remainder distributed over a three-month period. Overtime hours worked during the month was 6.4 percent (1,228 hours) of the total available hours.

Distribution of time was as follows:

	Man-Hours	% of Total
Fuels Preparation Department	5,206	21.89%
Irradiation Processing Department	2,109	8.86%
Chemical Processing Department	1,977	8.31%
Hanford Laboratories Operation	14,482	60.87%
Construction Engineering & Utilities Operation	16	.07%

Requests for emergency service increased, requiring a 6.4 percent overtime ratio compared to a 3.5 percent ratio for the previous period. A large percentage of the increase was due to shop work related to the 234-5 incident and to the start-up of the Purex process.

WD Richmond

WD Richmond:bk
May 10, 1962

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
R. J. Hennig	Plastic Bonding of Bricks and Blocks
I. L. Kirkelie, Jr.	Electrophoresis as a Means of Heavy Water Purification
L. C. Schmid	A Thickness Gauge
R. A. Walker	A System for Producing a Time Variable Transport Lag for an Analog Voltage (HWIR-1511)
R. C. Aungst	Roller Chain Extensometer for Tube Testing
Frank B. Quinlan	A Method of Preparing a Metallic Surface so that it will not Gall During a Cold Forming Operation (HWIR-1486)
Frank B. Quinlan and R. G. Wheeler	The Art of Bonding Iron to Zirconium or Zircaloy-2 by Hot Rolling and the Use of a Molybdenum Interface (HWIR-1489)
Frank B. Quinlan and R. G. Wheeler	The Art of Bonding Iron Alloys to Uranium with a Sound Metallurgical Bond (HWIR-1490)

Manager, Hanford Laboratories