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HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT

FEBRUARY, 1963

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HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT
FEBRUARY, 1963

Classification Cancelled
Declassified
By Authority of RLO CG 4
Hans Toffer 6-20-91
By Betty Zuchile
Date 6/20/91

Compiled by
Section Managers

March 15, 1963

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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TABLE I - HANFORD LABORATORIES FORCE REPORT

DATE: February 28, 1963

	At Beginning of Month		At Close of Month		
	Exempt	Salaried	Exempt	Salaried	Total
Chemical Laboratory	131	132	132	131	263
Reactor & Fuels Laboratory	186	173	186	171	357
Physics & Instruments Laboratory	95	66	96	68	164
Biology Laboratory	39	60	39	61	100
Applied Mathematics Operation	18	4	18	4	22
Radiation Protection Operation	43	92	44	88	132
Finance & Administration Operation	113	112	113	113	226
Programming Operation	11	3	11	3	14
General	3	3	3	4	7
Test Reactor & Auxiliaries Oper.	56	297	56	298	354
TOTAL	695	942	698	941	1639

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BUDGETS AND COSTS

February operating costs totaled $2,680,000, an increase of $325,000 from the previous month; fiscal year-to-date costs are $18,676,000, or 62% of the $30,239,000 control budget. February's cost of the Plutonium Recycle Program includes $109,000 for chemical processing of PRTR fuels. Hanford Laboratories' research and development costs for February compared with last month and the control budget are shown below:

(Dollars in thousands)	COST Current Month	Previous Month	FY To Date	Budget	% Spent
HL Programs					
02 Program	$ 83	$ 70	$ 590	$ 1 069	55
03 Program	27	4	66	175	38
04 Program	1 155	942	7 730	12 170	64
05 Program	124	113	804	1 373	59
06 Program	275	282	2 071	3 154	66
08 Program	11	2	20	100	20
	1 675	1 413	11 281	18 041	63
NRD Sponsored	172	152	536	1 270	42
IPD Sponsored	73	71	636	932	68
CPD Sponsored	118	106	960	1 421	68
FPD Sponsored			493	493	100
Total	$2 038	$1 742	$13 906	$22 157	63%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

Two fluted single tube N-Reactor fuel elements have successfully completed one cycle of irradiation in the ETR and will be recharged. Two irradiation tests of fluted N-inner elements are also continuing.

Radiometallurgy examination of prototypical N-outer components exposed to 1250 Mwd/ton reveal no detrimental features. Slight corrosion in a fuel support crevice was noted.

Improvements in hot heading techniques have permitted an improved design of the projection welded cap.

High temperature out-of-reactor loop tests of the braze layer of a "self brazed" closure revealed no evidence of corrosive attack.

Three thousand N-outer and 10,000 N-inner supports have been fabricated to supply emergency needs of NRD for fuel supports.

Tests simulating the wear of N-Reactor fuel supports on process tubes have shown a maximum wear of 1.3 mils after three series of tests. Tests were simulated by passing steel support shoes over a single path on the process tube 40 times, then autoclaving the tube, and repeating the process.

A charging test to evaluate the effect on N-Reactor fuel elements passing over the magazine-to-nozzle transition and nozzle-to-pressure-tube roll joint was conducted. Ten fuel charges were made with fuel of various lengths and support types. No cases of stripping of supports or support shoes was noted, but the leading edges of some "Arch" type fuel supports were damaged apparently as a result of passing over the tube-to-nozzle roll joint.

The effect of heat treatment on the size and dispersion of aluminum and iron compounds in uranium is being studied by metallographic techniques. UAl_2 particles in the 0.1 to 3 micron size range have been revealed. Techniques are being developed to reveal both the iron and aluminum constituents and to determine that the true structure is being revealed.

An optical readout system has been devised for measuring circumferential elongation of a tube specimen undergoing stress rupture test. The scale is read with an accuracy of ± 0.002-inch. On an N-Reactor tube section this represents about ± 0.02% of the circumference.

Experiments were conducted to determine if the addition of a large compressible fluid volume to the test system would affect "upstream" boiling burnout. It did not.

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The test in KER-1 to evaluate use of NH_4OH to regulate pH was terminated because of operating difficulties. The elements will be discharged at the next shutdown and a new test will be started. Development of thermocouple slugs is continuing.

N-Reactor instrumentation evaluation has been continued. A procedure for water hardness has been evaluated and shown to be satisfactory down to 10 ppb. The hydrazine procedure has been tested to determine effects of oxygen; there is some interference but no quantitative results are yet available.

Dilute decontaminant solutions suggested by Bettis for decontaminating Shippingport are ineffective in removing PRTR films. Moderate leakage into gaskets during decontamination, using the procedure developed for PRTR, does not affect the sealing ability of the gasket. Several modifications of the 4518 procedure appear effective in reducing amounts of precipitate.

The use of quachrom glucosate and low pH appears to be satisfactory for corrosion control in present reactors. Quantitative data on weight loss are not yet available. Marking inks appear to protect aluminum surfaces, with some inks being more effective. Corrosion rates of aluminum pieces in Zircaloy-2 tubes corrode at the same rate as do pieces in aluminum tubes.

In September 1962, 19 graphite cores were removed from D-Reactor for total stored energy (TSE) determinations. X-ray examination at that time disclosed extreme broadening of the 002 diffraction line, an indication (based on part correlation of line shift, crystal disorder and stored energy) of an increase in stored energy. This implication was not confirmed by recent National Bureau of Standards TSE determinations on five of these samples, as they showed only small changes in comparison with 1958 samples.

The pretreatment of uranium samples by anodic etching in nitric acid produced surfaces which were plateable in sulfate or sulfamage nickel plating baths. Plating adhesion is being studied.

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During startup of PRTR at low temperature and high flow, several tubes vibrated, giving an audible signal, noted only when Pump No. 2 was used. Water analyses at the same time showed a high zirconium concentration, > 10 ppb compared with 0–2 ppb observed normally, indicating fretting was occurring. A program will be followed during the next few PRTR operating cycles to determine validity of zirconium analyses as an indication of fretting.

Study of decontamination methods for use after plutonium fuel rupture continues in laboratory and out-of-reactor loop tests. Refractory fired PuO_2 resists chemical dissolution as expected.

The boiling-burnout tube power limit for PRTR Mark-I fuel elements at a coolant flow rate of 1.5 times the design value is predicted to be in excess of 2500 kw.

Design and fabrication of the second generation mechanical shim rod drive assembly was initiated. Installation of the shim rod environmental test facility is estimated 95% complete for mechanical work and 65% complete for electrical.

Tests of stressed stainless steel with 18% NaOH contamination in 275 C steam show rapid cracking. Additions of $KMnO_4$ to the NaOH contamination exhibits a mild inhibiting effect until reduced to MnO_2. Concentrations of 0.1 to 3% $KMnO_4$ in 18% NaOH were studied.

The corrosion of a small "dead leg" on the PRTR secondary water system was found to be primarily pitting corrosion of carbon steel. A systematic investigation of the cause and frequency of this corrosion is under way.

Accelerated out-of-reactor corrosion tests in steam of Zircaloy-2 coupons, representing different levels of cold work, have failed to show the reduced pickup of hydrogen in the vicinity of 10% cold work found after in-reactor exposure.

A number of new creep tests have been started under the BMI Zircaloy-2 creep testing program using specimens given a heat treat of 48 hours in vacuum at 400 C, analogous to process tube autoclave treatment, following a standard anneal at 750 C for 1/4 hour. After standard annealing only, strip tested at 550 F at a stress of 25,000 psi exhibits about 4% primary creep strain and failure occurs in a few hours. However, annealed and heat treated strip when tested at 550 F and 35,000 psi exhibits only about 0.4% primary creep strain and the stress rupture life is in excess of 1000 hours. Similar tests at higher stresses also show substantially lower primary creep strains and indicate that a 48-hour, 400 C heat treatment substantially increases the creep strength of annealed Zircaloy-2.

An in-reactor creep test at 20,000 psi stress and 350 C on 20% cold worked Zircaloy-2 has been in progress for 600 hours as part of a series to determine the stress dependency of creep during irradiation.

AISI 406 stainless steel tensile specimens have been prepared for irradiation in the ETR. Additional Zircaloy-2 specimens from N-Reactor process tubes, representing the various fabrication histories employed in their manufacture, are being machined and will also be irradiated in ETR.

An analytical model to evaluate the complex elastic energy distributions in a notched beam under load is being prepared using relaxation methods. To test the model, a fatigue fixture for the initiation of sharp defects in Zircaloy-2 is being built.

Cold-worked Zircaloy-2 undergoes substantial strength recovery (loss of strength) with annealing at 360 C after irradiation at 300 C in the ETR hot-water loop. Cold work recovery does not generally occur in unirradiated Zircaloy-2 below 450 C. Furthermore, specimens in the annealed condition before irradiation show only a small amount of radiation recovery as a result of postirradiation annealing at 360 C. Apparently,

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defects associated with cold work have been rearranged by radiation-induced defects to form a more thermally-unstable structure. Such an effect is substantiated by the increased in-reactor creep rate observed for cold-worked Zircaloy-2 upon reactor shutdown.

Procurement of materials for the coordinated Irradiation Effects on Reactor Structural Materials Program is continuing. One 10-ton heat of A302B which originally did not meet specifications was reheat treated to improve its ductility and is being shipped. A212B stock is also being shipped. Sheet and rod stock of columbium, columbium–1 wt% zirconium, tantalum, and tantalum–10 wt% tungsten refractory alloys, and rod stock of a 0.5 wt% titanium–0.08 wt% zirconium balance molybdenum (TZM) alloy have been received on site. Presently on order is a TZM sheet alloy and a tungsten–25 wt% rhenium alloy in both sheet and rod form.

A FORTRAN computer program has been written to facilitate the analysis of single crystal tensile tests. It provides for correction of elongation values for deformation of the loading system and computation of engineering stress, engineering strain, true stress, true strain, and resolved shear stress and glide strain on any crystallographic slip system.

In order to supplement the research program on molybdenum and to gain an understanding of radiation damage in other crystal systems, work has been initiated on parallel studies of nickel, a face-centered cubic metal, and rhenium (a close-packed hexagonal metal). Single crystals of these metals are being procured, with purities of 99.999% (Ni) and 99,995% (Re). Polycrystalline materials are also being obtained in the form of rod, foil, and wire.

Analysis is continuing on the tensile deformation of irradiated and unirradiated molybdenum single crystals containing various amounts of carbon. The results obtained to date suggest that the effect of increasing carbon content is a restriction of the ease with which cross-slip (i.e., transfer

of moving dislocations from one slip system to another) occurs, and that the effect of neutron irradiation is to restrict the extent of dislocation intersection.

X-ray lattice parameter and line broadening studies have been completed on nonirradiated and irradiated molybdenum foils. The results show the effect of purity very markedly. Very little change attributable to the neutron irradiation could be observed in the high purity foils while foils containing excess carbon showed greater changes in lattice parameter and line breadth after irradiation.

Quenching experiments have continued on molybdenum foils containing carbon. After quenching from temperatures as high as the melting point and subsequent thinning, no defects of the type observed in neutron bombarded molybdenum have been observed. Aging of quenched foils is currently in progress to determine whether quenched-in defects can cluster.

A series of nonirradiated U-Zr diffusion couples have been vacuum annealed at 650, 700, 800, and 950 C for 100 hours. These samples will serve as control samples for the study of irradiated U-Zr diffusion couples that were recently completed.

Processing for metallography, hardness and density of two thorium specimens irradiated at < 300 C to burnups of 0.18 and 0.92 at.%, respectively, has continued. Preliminary metallographic examination has revealed no irradiation damage.

Being determined is the size-distribution of second-phase particles in a series of heat-treated specimens from an uranium alloy containing Fe-Al additions. A series of micrographs have been analyzed. Also, a computer program normally employed in determining pore size distributions has been appropriately modified to handle second-phase particles. This information will be employed to select appropriate heat treatments for studying the influence of various types of second-phase particle size distributions on the irradiation behavior of uranium.

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Specimens recovered from two swelling capsules irradiated at 575 C to 0.15 at.% burnup are being processed for metallography and density. Preliminary density values indicate volume changes in the order of 30% which correlates with previous data. This volume change is a result of "growth" which leads to the development of microstresses between grains with resultant internal tearing. Light microscopy is partially completed and replicas have been prepared from two of the specimens.

A nondestructive gamma-scan test for determination of plutonium segregation in a mixed oxide fuel rod correlates well (qualitatively) with autoradiographs.

Extended surface, clip-on wear pads continue to perform satisfactorily on PRTR Mark-I fuel elements. Further repadding of PuO_2-UO_2 and selected UO_2 elements is planned. Stagnant water tests of fuel elements suspected of contaminating PRTR primary coolant indicate an Al-Pu element may have a cladding leak or surface contamination.

An experimental fuel element is being fabricated in which additional fissionable material can be added to the active fuel region during irradiation. Calculations have been made of the amount of low exposure plutonium needed to restore a spent PRTR mixed oxide fuel element to its initial reactivity value.

The simulated storage basin for experimental fuel element repair studies was completed and installed in one side of the pit used for remote fabrication studies.

Uniform plutonium distribution was found in UO_2–2.5 wt% PuO_2 densified by high energy impact.

Four types of UO_2 were impacted in a Dynapak machine with 4-inch-diameter, tool-steel punch to densities previously attained only with tungsten carbide punches at impact pressures of 300,000 psi or greater.

Sintering difficulties with ThO_2-PuO_2 pellets were eliminated by calcining pre-pressed powders in 1000 C air.

Recent improvements in autoradiography and mechanical sampling were used to remove microsamples of irradiated $(U, Pu)O_2$ for fission produce and plutonium analyses. Migration of plutonium and fission products during irradiation of MgO-PuO_2 and ZrO_2-PuO_2 was shown to be strongly dependent on mutual solubility and fuel temperature.

Eight irradiation test capsules containing comparison specimens of sintered UO_2, bicrystal UO_2, and tricrystal UO_2 were completed to further study effects of grain structure on fuel element performance. Data from other test capsules allowed estimation of the heat transfer rate through molten UO_2 (0.05 watts/cm^{-1} C^{-1}).

Special fabrications included completion of 200 Al-Pu rods for physics test in the PCTR and PRCF, sintering of approximately half the UO_2-PuO_2 pellets needed for PRTR flux monitor fuel clusters, and completion of eight Al-Pu coextruded rods for corrosion studies.

Material procurement and design of facilities for making the plutonium loading for EBWR are on schedule. Project funds were approved for initial phases of installation of vibrational compaction facilities.

Single-phase PuC and PuN samples are being readied for high exposure irradiation.

PuN-UN mixtures show complete, mutual solid solubility over the complete range of compositions.

Electrical resistivity measurements are being used to study phase changes in plutonium carbide. Preliminary results suggest that transformation of the ζ phase of PuC plus ϵ phase is not an isothermal transition as previously reported.

PuO_2-MgO mixtures show slight mutual solubility in the liquid phase. These mixtures are not compatible with tungsten at high temperatures, probably because of oxygen evolution.

UO_2, PuO_2, UN, and PuN mixed with aluminum nitride diluent are being evaluated as possible high temperature fuel materials for use with liquid metal coolants.

Reaction of O_2 with UN-W cermet was observed by reflection electron microscopy. Oxygen, introduced as a jet impinging on the specimen surface, produced the reaction without upsetting the high vacuum necessary to operate this equipment.

A method was developed to heat UO_2 specimens to over 1000 C in oxidizing atmospheres. This is an important step in studies of UO_2 properties under conditions where UO_2 will not lose oxygen.

Irradiated samples of UO_2 were explosively melted to study retention of fission products in rapidly melted and quenched UO_2.

An apparatus to measure the relative thermal conductivity of various types of UO_2 to the melting point was fabricated. This device will supplement studies on the thermal diffusivity apparatus.

An in-reactor capsule to permit measurement of the thermoelectric potential generated in a UO_2 fuel rod was readied for irradiation.

Uranium monosulfide shows an extremely rapid increase of hardness during irradiation.

Alpha-autoradiographs of UO_2 with better than 5 μ resolution were obtained.

Corrosion behavior of several superalloys has been determined in flowing, Grade A helium at 1900 F. The lead samples in the reaction tube oxidize much more heavily than the samples to the rear, illustrating the

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gettering properties of even the more oxidation-resistant superalloys at elevated temperatures. Some alloys (notably Haynes 25) are found to lose weight when exposed to the purified (gettered) helium.

Re-evaluation of the U^{238} cross sections to account properly for self-shielding reduced FSPPR enrichment requirements and reactivity coefficients using material volume fractions close to final design values. Results now indicate that required enrichment is near 20% Pu (13% fissile). Void coefficient is zero and the total negative flooding effect is -5.7% $\Delta k/k$. These values are not expected to be changed in the unsafe direction by core burnup. A preliminary estimate of breeding ratio is 1.1.

An initial transport theory calculation was completed for a core resembling the FSPPR. Preliminary comparisons indicate that diffusion theory calculations overestimate K_{eff} by about 2% $\Delta k/k$.

The rough draft of a document describing the Plutonium Fuel Spacecraft Reactor concept was completed.

A dome seal under test for possible AECL use was subjected to a total of 50 thermal cycles from 100 to 570 F at a constant pressure of 1500 psig. There was no evidence of leakage from the seal during these tests, but some unexpected deformation of the center of the dome and retaining lugs was observed.

In heat transfer studies preliminary analysis of coolant temperature data from the 6.3-foot-long, 19-rod-bundle test section with wire wrap rod spacers shows that interchannel mixing reduces the subchannel coolant temperature disparities about 80% from those which would result with no interchannel mixing.

2. Physics and Instruments

Additional data were obtained from an exponential pile experiment for improvement of buckling measurement accuracy on the N-Reactor lattice. Buckling values are useful in estimating the critical size of reactors.

Calculation of self-shielded cross sections for natural uranium rod and tube-in-tube lattices has pointed up a deficiency in the unresolved fission cross section of U^{235} as computed with code GROUSS. The unresolved resonance parameters are being examined.

Analog simulations of the N-Reactor pressure injection and bleed systems have been successfully developed and studied on the computer. Results to date have revealed the need for modificating one automatic controller. Presently planned computer runs are now complete; however, more detailed simulations are continuing to accurately evaluate the effect of abnormal water pressure and level transients, and to investigate the consequence of instrument failure, reactor scrams, and normal operating changes.

Experiments are being performed at D-Reactor to determine the relation of control rod position to outlet water temperature. The coefficients obtained from these tests will complete the data necessary for a 4-node, two-dimensional representation of that reactor.

A novel method of combining reactor flux level measurements with reactor period measurements so as to provide a single parameter indicative of reactor performance is being investigated. Data in this form are expected to be particularly useful for reactor startup operations. Feasibility of the concept has been established mathematically.

At the Critical Mass Laboratory, an 11.5-inch-diameter criticality sphere filled with plutonium nitrate solution was used to irradiate neutron detecting foils. This is the start of a program to measure the neutron flux distribution and the neutron energy spectrum in plutonium solution systems.

Final preparations are under way for criticality studies on PuO_2-plastic mixtures. The Remote Split-Table Machine is ready for installation in the second hood in the critical assembly room. The fuel is complete and is in storage at the laboratory.

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Theoretical critical mass studies included computations of criticality for Pu solutions in H_2O reflected spheres for comparison with measurements, development of scattering kernels for polystyrene, calculations of the buckling of partially filled spheres, and further work on subcritical interactions.

Nuclear safety consulting services were provided CPD regarding storage of plutonium bearing waste materials and reactivation of a hood used for dissolving PuO_2.

The fourth and fifth sessions of a course in Nuclear Safety for CPD engineers and other interested HAPO personnel were begun at the Critical Mass Laboratory. There are 50 people taking the course including five members of AEC-RLOO and one person from Euratom.

A unique contribution was made to scientific knowledge through successful measurement of the total cross section of Pm^{147}, a fission product isotope. Data from the measurement are still being processed.

Startup experiments began in the Plutonium Recycle Critical Facility. Neutron multiplication measurements were completed during the loading of 25 UO_2 fuel clusters.

PCTR measurements continued on PuAl fuel clusters in a 10-1/2-inch graphite lattice. Data were obtained with the lattice poisoned to $k_\infty = 1$. Work is under way on the unpoisoned lattice.

Success was achieved in reproducing from transport theory the neutron flux values measured in a graphite PCTR core pierced by a large copper bar. The agreement between theory and experiment was excellent. The theoretical calculation also produced interesting values for neutron temperatures in the copper and in the graphite. Typical values are 45 C in the graphite, 121 C at the surface of the copper bar, and 180 C at a depth of 0.3 cm into the bar.

The value of plutonium as an alternative fuel has been determined to be less dependent on isotopic composition and fueling strategy when lattice spacing and enrichment are simultaneously optimized than when they are not.

Fuel burnup in the EBWR is being computed to aid in planning the EBWR experimental program.

Burnup calculations for Pu-fueled compact water-moderated cores indicate that the Pu^{240} content will probably have to be higher than previously expected.

The project proposal for design funds for the High Temperature Lattice Test Reactor and a summary of the design criteria have been drafted.

In the extended lifetime neutron flux monitor program, samples for determination of neutron flux parameters are ready for irradiation in KE-Reactor.

The prototype multiparameter eddy current tester has successfully separated three test parameters on a composite test specimen consisting of brass and steel layers, each of different thicknesses. As test parameters were varied, probe spacing was read out on one meter, brass thickness on a second meter, and steel thickness on a third. Present tests provide accurate data separation for parameter variations up to approximately five mils.

Wide frequency band ultrasonic test equipment and 1/8-inch-diameter transducers are being applied to 0.017-inch-thick wall tubing under the USAEC-AECL Cooperative Program. Defect resolution of the system is capable of resolving the front and back surface echoes of longitudinal waves reverberating in the tube wall in a direction normal to the surface. The echoes are separated in time by only 0.2 microseconds.

Further analysis of the dispersion data collected over the past 3 years has provided a basis for comparison of climatic and terrain effects on the atmospheric dilution characteristics of the atmosphere. Dispersion measurements over level terrain were comparable at the three sites studied for similar

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meteorological conditions. However, measurements obtained over rough terrain averaged about 0.7 of those over flat terrain at comparable distances.

Work on the following radiological physics problems continued routinely: Whole body counting for P^{32}, assistance to RPO with their truck shadow shield, X-ray scintillation counting, cosmic ray measurements, calorimetry of Pm^{147}, and study of the heat amplifier.

Collection of urine samples was started from parts of Alaska not visited last summer with the shadow shield whole body counter; these samples will be counted for Cs^{137} to estimate the corresponding body burdens.

Information received from Swedish Hospital concerning the whole body counting experiments done there last November indicates that the results were of sufficient interest that they wish to build their own counter.

Maintenance work on the Van de Graaff positive ion accelerator was completed and it seems to be running well again.

A large area, 80 cm^2, scintillation detector for low energy betas was developed, assembled, and successfully used to aid in monitoring for Pm^{147} contamination. The instrument can detect uniform depositions of Pm^{147} of 1.6×10^{-5} $\mu c/cm^2$ compared to levels of about 3.2×10^{-4} $\mu c/cm^2$ detectable with a standard thin-mica-window GM detector.

3. Chemistry

In reactor effluent radioisotope reduction studies, additional data confirm that removal of parent material alone from process water is not sufficient to effect large reductions in radioisotope output in the effluent water. Coatings of some silicones, epoxy resins, and coumarone-indene resins are effective in reducing adsorption of arsenate ion on aluminum.

Shallow monitoring wells in the Purex 241-A tank farm which showed greater than background readings on December 28, 1962 were reprobed on

February 7, 1963 with the gamma scintillation probe. The only significant change was a slight increase in the reading in a well on the south side of the 106 tank.

A series of tests was completed simulating leaks in a direct-buried pipeline which will be used to transport fission product recovery solution from C-Farm to B-Plant. Results indicate that surface contamination and localized heating of the soil mass around a leak are more probable than ground water contamination.

Laboratory and engineering studies of the CSREX process continue to show promise for the extraction of fission product cesium, strontium, and rare earths from Purex acidic waste, and for the subsequent separation of cesium and strontium from the rare earths.

Culminating laboratory and semiworks development of a process for destruction of nitric acid in Purex 1WW with sugar, a plant test successfully demonstrated conditions as predicted. Laboratory studies of less expensive sugar concentrates (e.g., molasses) are promising.

A flowsheet for 0.3M trilaurylamine-Soltrol extraction of neptunium and plutonium from Purex 1WW was successively repeated in a laboratory demonstration. Batch extraction losses remained below 5% and stripping efficiencies were approximately 80%.

Analytical results verified that the 10,000 curies of promethium which were recently purified are chemically and radiochemically pure—all impurities below detection limits. Nuclear, physical, and chemical properties of promethium are being determined. The 12.9 grams of promethium oxide were packaged for shipment.

Results of scouting studies indicate that zeolites might be suitable for storage and/or shipping of recovered rare earth isotopes in addition to cesium and strontium.

In salt cycle studies of the precipitation of cerium dioxide (as a stand-in for plutonium dioxide), approximately 90% of the cerium was precipitated from a 2.5 LiCl-KCl mixture with a 60% O_2–40% Cl_2 sparge. A quartz filter recovered 70% of the precipitate.

The UO_2 products from engineering-scale LiCl-KCl salt baths were demonstrated to meet tentative ASA nuclear grade UO_2 density specifications.

Ruthenium volatilization was markedly reduced in a recent full-level spray calcination run in the High-Level Radiochemistry Facility. Less than 0.1% of the ruthenium volatilized when alkaline Purex 1WW was calcined, compared with 3 to 20% volatilization with acidic Purex 1WW.

Engineering studies of the operation of the 18-inch radiant-heat spray calciner indicate the capacity of the unit is increased 20% and without detrimental side effects by preheating the feed and using steam for atomization.

An experimental investigation undertaken to adapt the Dynapak process to fission product packaging is providing encouraging results.

4. Biology

Previously noted toxicity of the reactor tube cleansers, SULFAM-3 and BISULF-16, was shown to be probably due to their effect in reducing the pH of the raw water.

The present incidence of columnaris infection in older trout appears to be a result of retention of the organism from last summer's exposure and not due to re-infection from current river water exposure.

Yearling trout that were fed Zn^{65} and then killed after 94 days retained 19% of the dose. Highest concentrations of Zn^{65} were present in the gill filaments and gastrointestinal tract.

Three-tenths % of Sr^{85} titanate fed to young miniature swine was retained in the skeleton at 7 days postadministration. This compares to a 5 to 10% absorption and retention of strontium fed as the chloride.

The concentration of I^{131} in cows' milk is about the same as the concentration in blood. In sheep the concentration in milk may be five times the concentration in blood.

Thyroid I^{131} burdens in miniature goats are 3 to 4 times the daily intake. Newborn offspring had I^{131} burdens about one third those of the dams. The blood of the fetuses contained 2 to 3 times the I^{131} concentration of the dams' blood.

Pu^{239} administered to rats with a 75-fold molar excess of citrate was less acutely toxic than Pu^{239} administered with a 4-fold molar excess of citrate.

Plutonium injected subcutaneously in the foreleg of miniature swine was not effectively removed by intravenous DTPA treatment.

The uranium ore dust has been physically and radiochemically characterized and the first exposure of rats will begin this month.

In cooperation with Dr. Kelly of Industrial Medicine, bronchial lymph nodes were surgically removed from a dog. This rather drastic surgical technique is being examined to determine its feasibility in the event that harmful effects are noted from plutonium accumulation in bronchial lymph nodes.

The postirradiation response of the intestine with respect to bile salt absorption and metabolism was studied with C^{14}-labeled taurocholate in rats with biliary fistulae.

"Active transport" was shown to be involved in the absorption of I^{131} by plants, but not involved in the absorption of Rb^{86}.

Differences noted in the sensitivity of yeast strains to D_2O was related to their manner of oxidizing substrates. Loss of capacity to carry on aerobic metabolism was correlated with an increased sensitivity to D_2O.

Plankton sampling at the Vantage and Priest Rapids stations was stopped after collection of one year's data at these locations.

Radioiodine concentrations in deer, elk, and reindeer thyroids are decreasing with a longer half-time (25 to 30 days) than noted a year ago.

5. Programming

Computer codes used for fuel cycle calculations are undergoing extensive revision so that economic factors associated with the production of Cm^{242} and Cm^{244} in central station power reactors can be studied. The fuel cost code, QUICK, was altered to allow the assignment of separate sales prices for Pu^{239}, Pu^{240}, Pu^{241}, Pu^{242}, Am^{241}, Am^{243}, Cm^{242}, and Cm^{244}. Effective cross sections for the americium and curium isotopes are being provided by the Applied Physics group. In studies of plutonium recycle for power production, assumptions regarding the nonfission capture of neutrons in plutonium were made conservatively high to make the computed plutonium values conservatively low. When burning plutonium to make curium, however, it is necessary to eliminate this bias to avoid getting optimistically high values of curium production.

TECHNICAL AND OTHER SERVICES

Widespread contamination involving Pm^{147} occurred in the 325 Building. The spread of contamination apparently originated from analytical samples which were stored or used in rooms 303 and 309. Approximately one-third of the main floor of the 325 Building became contaminated before control of the contaminants could be established. The upper floor and basement areas were unaffected. Personnel surveys of approximately 200 employees in the building revealed 12 cases of detectable skin contamination but generally the activity was found to be low and the areas contaminated small. The maximum skin dose did not exceed 4.5 rads. Evaluation of bioassay samples submitted by 29 employees indicated that there were 14 cases of measurable deposition of Pm^{147}. The preliminary estimate of the

maximum internal deposition was < 1% of the maximum permissible body burden for Pm^{147}. Minor levels of contamination were found and removed from seven employees' homes. The event was categorized as a Class A incident by the AEC according to AEC Manual Chapter 0502.

There were no new plutonium deposition cases confirmed during the month. The total number of plutonium deposition cases that have occurred at Hanford is 316 of which 229 are currently employed.

A series of statistical analyses were performed on various problems associated with rail heights on self-support pieces to be charged in the K-Reactor.

In connection with the control of measurement biases in C-Basin, a comprehensive audit program was proposed. This program will enable the detection of observer and/or time differences.

An experiment was designed to determine optimum conditions of time, temperature, and pressure for placing end caps on fuel elements in the hot-die-sizing canning process.

A new computer program has been written to improve the estimates obtained from pencil dosimeter data by treating zero pencil readings as censored values.

The quantitative metallography program for estimating volume fraction and density of a spherical phase in a three-dimensional matrix from particle size distributions is being modified to handle large diameters which are being visually sized by the observer.

Coding and debugging of an extremely versatile computer program for obtaining numerical solutions to a simultaneous set of first order differential equations is virtually complete.

Completed was a study of the efficacy of compositing bioassay specimens to reduce the analytical work necessary to identify individuals with test results above a certain permissible level.

SUPPORTING FUNCTIONS

February output for PRTR was 752 Mwd for an experimental time efficiency of 70% and a plant efficiency of 50.5%. There were three operating periods during the month, all of which were terminated manually as a result of excessive helium and D_2O leakage.

Fuel exposure history at month-end was:

Maximum UO_2 exposure/element	2756 Mwd/ton_U
Average UO_2 exposure element	1864 Mwd/ton_U
Maximum Pu-Al exposure/element	77.3 Mwd
Average Pu-Al exposure/element	58.6 Mwd
Maximum Moxtyl exposure/element	39.4 Mwd (~ 788 Mwd/ton_U)
Average Moxtyl exposure/element	20.4 Mwd (~ 409 Mwd/ton_U)

PRTR's core loading on February 1, consisted of 24 UO_2 elements, 24 Pu-Al elements and 37 mixed-oxide elements. The core loading at the end of the month consisted of 22 UO_2 elements, 22 Pu-Al elements, and 41 mixed oxide elements.

Thirty-two Pu-Al fuel elements from PRTR, with an average Pu^{240} content of 23% for the 5.9 kg of plutonium, were shipped to Chemical Processing.

A total of 129.5 PRTR outage hours were charged to repair work. The major time consumers were process tube gasket replacement (102 hours) and repair of internal valve leakage on several helium system valves (15 hours).

Eight PRTR process tubes were examined. No significant damage was observed although there were many shallow marks of approximately 0.001-inch observed in areas where wire wraps could hit the process tube.

Studies of the effects of a DC power failure were completed during the month. The results of the DC power failure study showed that necessary manual actions can be taken to compensate for the inoperability of the electrical circuits.

The planned loading of 25 UO_2 elements in the PRCF was completed, and the desired data was generated. These initial loading steps proceeded very smoothly and yielded expected results with no problems of any significance.

The mechanical design test in the Fuel Element Rupture Test Facility was 50% complete. The total containment portion of the design tests was performed. The pressure transient and movement of piping were measured and found to be satisfactory. Operation of the system under automatic pH control with both single pass and recirculation was conducted. The system worked satisfactorily except that minor pH measuring system revisions will be required.

Total funds authorized the General Electric Company for the Gas-Cooled Loop Facility (GCLF) were increased to a total project authorization of $1,170,000 by AEC Work Authority CAH-822 (7), dated February 14, 1963. The project is 94% complete versus 94% scheduled as of the first of March.

Insulation for GCLF is now fully complete for the performance of design tests.

Following a thrust bearing failure on GCLF's Number 2 unit being tested by Bristol Siddeley, both blowers have been rebuilt and are now undergoing final performance tests. According to the latest information the Number 3 unit has run at conditions of 10,000 rpm 465 psia inlet pressure and temperatures up to 350 C with carbon dioxide. These conditions were not all achieved concurrently but do represent a marked improvement from past test results.

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Total productive time in Technical Shops Operation for the period was 21,565 hours. Distribution of time was as follows:

	Man-Hours	% of Total
N-Reactor Department	3 278	15.20
Irradiation Processing Department	3 853	17.87
Chemical Processing Department	827	3.83
Hanford Laboratories	13 607	63.10
Construction Engineering and Utilities Operation	0	0

Manpower utilization in Laboratory Maintenance Operation for February is summarized as follows:

	Hours	Hours
A. Shop Work		2574
B. Maintenance		8954
1. Preventive Maintenance	1800	
2. Emergency or Unscheduled Maintenance	2300	
3. Normal Scheduled Maintenance	4700	
4. Overtime	600	
C. R&D Assistance		9145

Total productive time realized was 20,673 hours.

Authorized funds for eight active projects in the Supporting Functions Category total $6,244,500. The total estimated cost of these projects is $7,689,000 of which $622,000 had been spent through January 31, 1963.

Heavy water physical inventory at the end of February indicated a loss of 2781 pounds valued at $38,500. Scrap amounting to approximately $600 was generated during the month. In addition to these losses, operating costs will be charged $10,657 for write-down of scrap heavy water on hand to absorb a SROO price change which was effective January 1, 1963.

The price for new heavy water was also reduced--to $19.94/pound, a reduction of $0.05/pound in the fund price. The new price includes $13.71 fund and $6.23 nonfund.

An allocation of $150,000 in 02 Program Equipment funds during the month brings the total FY 1963 authorization to $1,800,000.

An additional authorization of $100,000 was received for Project Whitney fabrication work. The total FY 1963 authorization is now $272,000.

Hanford Laboratories assumed, from Relations and Occupational Health Operation, responsibility for HAPO plant tours and for the operation of the Visitors Center in the Richland Community House.

Professional recruiting activity summarization:

	This Month				Open Offers at Month End
	Plant Visits	Offers Extended	Acceptances Received	Rejections Received	
Ph.D.	7	2	0	2	3
BS/MS (Direct Placement)		6	4	3	7
BS/MS (Program)		93	5	20	139

Eight Technical Graduate Program members were place on permanent assignment. Four additions to the Program were made; its current strength is 46.

Documents listing "Hanford Formal Research and Development Reports" and "Speech and Article Clearance Activity" for 1962 were issued for release through the AEC's TID and SA programs, respectively.

Manager, Hanford Laboratories

HM Parker:JEB:dph

[redacted]

REACTOR AND FUELS LABORATORY MONTHLY REPORT

FEBRUARY 1963

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 02 PROGRAM

1. Metallic Fuel Development

Fuel Irradiations. Two fluted single-tube N-Reactor size fuel elements completed one cycle of irradiation in the ETR M-3 pressurized loop facility. These Zircaloy-2 clad 0.95% enriched test elements operated with a peak power of approximately 130 kw/ft and a corresponding maximum metal temperature of 520 C. Exposure of approximately 200 MWD/T was obtained. Weight measurements indicated a volume increase of slightly over 0.2% had occurred. The elements were in excellent condition with no visible corrosion effects or distortion. The test was recharged in the reactor for continuation of the irradiation.

Two other irradiation tests of fluted elements continued. Three N-inner size elements continue their irradiation in cold water in the KE Reactor and one N-inner size element has completed five cycles of irradiation in the pressurized high-temperature P-7 loop of the ETR.

A second "N" outer fuel component of a 14 NAE charge that was irradiated to an average exposure of 1250 MWD/T is being examined in Radiometallurgy. As was the case with the first element examined, no unusual features or shortcomings have been observed in the examination of the Be-Zr eutectic brazed closures, the fuel, the cladding, or the fuel-to-clad bond. A slight buildup of corrosion product has been observed in a tight crevice between a spot-welded support and the fuel clad. No appreciable hydriding was observed in conjunction with the corrosion product. Examination of this support will be continued.

Modification of the measurement data analysis program to handle information from the basin measurement equipment is complete and been checked. The output of the program includes punch cards of summary data which will be included in a fuel element history file. Additional programs have been written which can order various fuel element data in terms of any one of the measured properties and determine correlation coefficients between any series of measured properties. These later programs are in the "debugging" stage.

Hot Headed Closure Studies. The hot-heading work for the hot-headed brazed closure was directed toward producing a thickened Zircaloy-2 inner shoulder on as-extruded NPR inner tubes. The fuel stock initially used had been previously beta heat treated, thus destroying the extrusion orientation in the uranium. This could have affected the thickening tendency of the Zircaloy-2 cladding. No great deviation was noted when as-extruded stock was utilized. However, results indicate that a slight die alteration will be necessary for best results.

Hot-heading tooling will be designed and fabricated to hot head NPR outer elements. Existing tooling was made for the old NPR outer size.

Due to recent improvements in hot-heading techniques, fuel element sections were received which had thicker cladding on the ends. Because the end cladding was thicker, the end of the elements could be machined to eliminate the radii on the OD and the ID corners which previously interfered with projection welds. The elimination of the radii allows the ring projection on the caps to be relocated nearer to the OD and ID of the element. The first cap designs welded to the improved hot-headed elements showed an OD and ID joint crevice of approximately 0.009 inch whereas previous welded elements show approximately 0.038 inch.

Although immersion plating techniques that are used to preplace thin films of copper on the caps were improved, the quality of the projection welds have decreased. This decrease in quality is attributed to the vapor or grit blasting which is one of the operations used in the immersion plating process. It is anticipated that alternative plating preparation techniques will restore the projection weld quality.

An evaluation procedure for the hot-headed projection welded closure has been prepared. This evaluation allows the complete testing of the hot-headed projection welded fuel on a comparative basis with the production brazed material. The evaluation includes quantitative mechanical property measurement, corrosion behavior, rupture and defect behavior, metallographic and radiation testing. Testing will begin as soon as sufficient material is available.

"Self-Brazed" Closure Development. Three specimens, including one closure which had been faced off on the end to remove the weld bead and lay bare the braze layer were placed in the T-7 (formerly known as ELMO-7) high temperature loop in K-East area in December. Since then, (8 weeks), these specimens have been exposed continuously to

environmental conditions simulating actual reactor operating conditions (except for nuclear effects). The most recent bi-weekly examination revealed no evidence of corrosive attack on either the weld-beads or the Cu-Ni-Zr braze layer. This indicates that the self-generated eutectic braze alloy is quite resistant to corrosion in hot pressurized water.

Specimens fitted with the self-brazed closure were subjected to the production ultrasonic bond test. Those processed strictly according to the recommended procedure were generally completely sound, while those representing various deviations from the recommended process showed more or less defective bonds. Selected specimens were subjected to thermal cycling between room temperature and 300 C. Others were cycled in the salt bath according to the normal beta heat treatment procedure.

Repeat testing by ultrasonics showed no significant change in bond quality as a result of the thermal cycling. Some of the specimens showing somewhat spotty bond indications were destructively examined for verification. The worst ultrasonic specimen was found to have only a small void in the bond and several micropores.

The current status of the development is as follows:

(a) Four completely closed 6" NPR inner test elements are mechanically ready for in-reactor testing. These had the closures processed prior to beta-heat-treatment for structure; they have now been beta-heat-treated; will be tested for bond quality; vapor blasted, etched, inspected for surface quality and dimensions; will have supports added; will be autoclaved and will then be ready for charging.

(b) Twelve 6" NPR inner test elements made from pre-beta-heat-treated stock are in process of assembly and preparation. Some of these are intended to be irradiated along with those of (a) above.

Analysis of Fuel Element End Caps. The difference in thermal expansions between the fuel and end cap material is a potential cause of fuel element failure. Shaping the fuel element end cap should reduce some of the stress effects resulting from the thermal expansion incompatibility thus minimizing this potential source of failure. A numerical method of stress analysis has been programmed; however, in initial calculations it has not been possible to get the machine solution to satisfy the semi-infinite boundary conditions. Further modifications of the numerical procedure will continue in order to make this solution satisfy the boundary conditions.

N-Reactor Fuel Support Development. Approximately 10,000 N-inner supports were supplied to NRD for use on N-fuels. Fabrication of inner supports by the laboratories will continue as required by NRD. The sheet material for the manufacture of these parts is fabricated in the laboratory using 3½" x 3/8" extruded sheet bar as the starting material.

Approximately 3000 N-outer fuel supports have been fabricated. Examination of the first "proof lot" of 1000 pieces by NRD inspection indicated a reject rate of 9% for cracks. Closer examination by MFDO indicated a crack reject rate of closer to 20%. Some forming variability was noted in different sheet stock, and variability from end to end within individual strips. Application of lubricant during forming eliminated the cracking problem. The supports are fabricated in two steps, forming and coining projections.

Fuel Support - Process Tube Wear Tests. Laboratory tests to study possible damage to the N-Reactor process tubes by charging of N-fuel elements are proceeding. A series of tests are being conducted to determine to what extent the reactor process tube is damaged by the repeated corrosion of the bare Zircaloy-2 that is exposed by scratching by the fuel element supports. The test cycle consists of passing steel support shoes 40 times over a single path on the process tube, then autoclaving the tube in 400 C, 1500 psi steam for 14 days to give a corrosion product buildup on the bare Zircaloy-2 equivalent to that expected to form in-reactor between the charge and discharge of the fuel elements. A third series of 40 passes was completed and the depth of the scratch measured. The total depth of the damaged area now measures from 0.0002 to 0.0013 inch. The maximum depth occurs near the downstream end of the tube in the area of a weld in the process tube section. Elsewhere along the tube the scratch depth is 0.0002-0.0010 inch. The process tube was returned to the autoclave for another 14-day cycle. Further tests will be run to determine the relationship of additional corrosion and charging cycles to the scratch depth.

Cladding Deformation Studies. Thirty-six NaK capsules containing a total of 94 Zircaloy-2 clad uranium rods are being irradiated in F-Reactor to provide data on the effects of cladding thickness uniformity, temperature, and exposure on the strain capabilities of Zircaloy-2 fuel cladding. Four of the capsules have thermocouples to measure the central uranium temperature. A high resistance has developed in one thermocouple which is attributed to a faulty connector on the reactor rear face. At the next outage, the connector will be replaced with a direct silver soldered joint. The other three thermocouples are recording satisfactorily, and agreement

between measured and calculated central uranium temperatures is excellent. The latest temperature and power data indicate that the cladding is currently operating approximately 10-15 C higher than desired. The exposure has now reached approximately 650 MWD/T. Goal exposure for nine of the capsules is approximately 1000 MWD/T and for the remaining 27 the exposure goal is 2000 MWD/T.

Heat Treatment of Zircaloy Clad Fuel. N-outer tube stock with the core material containing 640 ppm Al and 400 ppm Fe is being studied to determine the effects of heat treatment on the size and dispersion of UAl_2 and U_6Fe compounds. Treatments of 100, 200, and 500 hours at 400, 450, and 500 C were completed for samples previously quenched from the beta and gamma phases. These are currently being examined metallographically to determine the stability of compound size and dispersion at temperatures expected in-reactor. Selected samples of heat treated material representing the extreme conditions of compound particle size and distribution are being evaluated by electron microscopy. Initial examination was for material polished and cathodically etched prior to replication. This technique revealed UAl_2 particles in the 0.1 to 2-3 micron size range but did not show the U_6Fe compound in those specimens in which the compound had been observed optically after a citric acid inclusion etch. These structures were used, employing a Zeiss Particle Size Analyzer, to obtain data on size and volume distribution by techniques similar to those used for pore size analysis of irradiated uranium.

These data have been submitted for computer analysis. Two of the specimens were repolished and instead of cathodically etching, were etched using a phosphoric acid electrolytic bath. Replicas for the electron microscope were obtained for 1-, 2- and 3-minute etches and the results are currently being evaluated. This technique does retain the U_6Fe particle and appears to yield a smaller particle size for the UAl_2 compound. Additional work is required to be assured that the true structure is being observed in electron microscopy and is quite basic to the accurate evaluation of uranium with additives for fine compound dispersions.

Alloy Coextrusion Cladding Component Evaluation. The use of a harder and higher strength material than Zr-2 for the cladding component of the inner bore of the N-outer tube was attempted to determine the effect on clad thickness variation. A 7.5-inch diameter x 11.5-inch length as-cast billet containing 3 w/o Nb - 1 w/o Sn - balance Zr was extruded at 760 C (1400 F) at a 10.6:1 reduction to a 2.9" OD x 1.85" ID tube for machining to coextrusion billet components. The starting extrusion constant for the primary extrusion was 17.4 tsi. One N-outer tube coextrusion (ZT-27) was made with the core billet

specifically picked as a billet tester reject. An alloy inner clad component was used on the rear half and normal Zr-2 on the front half of the billet. Cladding thickness variation averaged 0.006" (0.005-0.009) for the alloy section and 0.005" (0.003-0.006) for the Zr-2 section, indicating no beneficial effect. This alloy component was machined from the as-extruded primary. Additional lengths have been given a 850 C, 2-hour, water quench heat treatment, increasing the hardness from R_c 20 to R_c 27, and are being machined to billet components for additional evaluation.

Zr-Be-U Corrosion Testing. Corrosion testing of the zirconium-beryllium system with beryllium contents up to 5% has been resumed. These tests take place in 360 C water and will extend up to 3000 hours. Both weight gain and changes in light reflectivity are recorded. A concurrent test is under way with zirconium-uranium alloys in 360 C water. Again, the weight gain (or loss) and light reflectivity is determined. A third series is under way, again in 360 C water, with zirconium-beryllium-uranium alloys. The same variables will be recorded. It is hoped that these same tests can be repeated in 360 C - pH 10 water for a comparison to the pH 7 tests.

Fuel Fouling Detector. An excellent seal between a Zircaloy-2 clad thermocouple and the Zircaloy-2 end cap of a fueled probe can be achieved by Be-Zr brazing. However, the quality of currently available Zircaloy-2 thermocouples is inadequate for in-reactor use due to extensive internal cracking in the Zircaloy sheath. Two test lots of Zircaloy-2 clad thermocouples have been ordered for evaluation purposes in an effort to obtain a high quality product. Mechanical seals including a Zircaloy-stainless Swagelok, a mechanical seal developed by BAPL, and a compression seal developed for N-Reactor instrument tubing are being evaluated for use with stainless steel clad thermocouples.

Zr-2 Thermocouples. The fabrication of Zr-2 sheathed thermocouples is in progress. Half-inch OD x 30-mil wall tubing is first free sunk to 200 mils inside diameter after which the tube will be drawn down over a piano wire mandrel. The final ID after drawing should be 0.187" at which time the thermocouple, insulated with crushable beads, is inserted. The final step will be to swage the Zr-2 tube onto the crushable beads, thereby forming a dense insulating core.

2. Corrosion and Water Quality Studies

Effect of Low pH and Quachrom Glucosate. First exposure of aluminum and steel alloy samples, and K_4N test elements, was completed in filtered water with 0.20 ppm Quachrom glucosate added as the corrosion

inhibitor. This testing is being conducted in 1706 Facilities SP-5 and SP-6, with control exposure in SP-2, SP-X, and SP-Y, using sodium dichromate inhibitor at pH 6.6. Non-fuel samples show uniform, shiny oxide surfaces, with less film present than on control sample surfaces. Both test elements and alloy samples are being processed for corrosion and film measurements.

Effect of Ink Markings on Corrosion and Film Absorption. It has been observed that aluminum samples marked with ink and exposed in the reactor appear to be protected by the ink. In some cases, the pigment is still apparent after several weeks' exposure; in other cases, the corrosion on the unmarked areas seems more pronounced. In a recently completed test, some dummy elements (spacers) were marked with four different types of ink and exposed in the reactor. In one test the pieces were exposed for 4 weeks in process water; in the other test the pieces were exposed in deionized water for 6 weeks. In all cases the ink appeared to protect the aluminum; the effect was more pronounced with the red and yellow inks than with the black ink.

Corrosion of Aluminum in Deionized Water. Tests were continued in the Single Pass KE Facilities to evaluate corrosion rates in deionized water. The galvanic effect of zirconium to aluminum was of little importance; elements (dummies) in a Zr-2 tube corroded at the same rate as elements in an aluminum tube (about 3.5 mils in 2.5 months). The corrosion was very uniform with no evidence of localized corrosion. Some measurements were made of radionuclides present in the film, but the results were quite variable, differing by one or two orders of magnitude.

Decontamination and Chemical Cleaning. The decontamination sequence used in PRTR and suggested for NPR was used to defilm TF-1 Loop after approximately 250 hours of prefilming. One test section gasket was loosened to allow leakage during the decontamination. Following decontamination inspection of the gasket revealed some leaching of the asbestos from the face of the gasket. The gasket was reseated and operation has continued since 2/11/63 at 1650 psi and 300 C without leakage through this gasket.

A single-step proprietary cleaner (Wyandotte 5231) was quite effective on PRTR piping coupons, giving decontamination factors of four in the laboratory. The solution is used at 10% concentration for 6 hours at 100 C. PRTR coupons are being used in these evaluations, because films on these coupons are the most difficult to remove.

A laboratory study was made to evaluate the effectiveness of the Bettis process suggested for cleaning the Shippingport reactor. The process includes treatment with 2% alkaline permanganate at 100 C for six hours followed by 1.2% ammonium citrate at 95 C for six hours. The decontamination of PRTR piping coupons was unsatisfactory, giving a decontamination factor of 1.6. If the dilute permanganate step is followed by a two-hour treatment at 85 C with a mixture of 5% ammonium citrate and 5% oxalic acid (T4518), the decontamination factor is about 8.5.

The mixture of oxalic and citric acids at the proper pH appears to be quite effective in laboratory studies and does not cause the precipitation problems noted with the 10% oxalic acid. In one procedure described in the literature, 0.2% ferric sulfate is added to the oxalic acid to prevent precipitation. This was tried in the laboratory with moderate success. The bulky film noted during treatment with oxalic acid was not present; however, a small amount of flocculent film did form.

Evaluation of NH_4OH in K-1. The present testing in KER-1 is designed to determine: (a) the best method for chemically cleaning the primary or secondary system of N-Reactor, and (b) the problems in substituting NH_4OH to regulate pH in the primary coolant. The present test involves chemical cleaning of the ex-reactor section, recirculation in the ex-reactor section at pH 10 with lithium hydroxide to form a protective magnetite film, charging of fuel elements, recirculation in-reactor at pH 10 with lithium hydroxide to check water quality, and finally conversion to pH 10 with ammonium hydroxide and operation in-reactor for the balance of the operating period. All operations had been accomplished up to conversion to ammonium hydroxide, but operating difficulties will force postponement of the test. Leaks in KER-1 occurred and it will be necessary to replace a valve. The present charge will be dumped at the next shutdown and the test will be started again.

Thermocouple Elements for K-1 Operation. During the test with NH_4OH the film buildup on the fuel elements will be followed, using thermocouple fuel elements. Very little progress has been achieved in the fabrication of a satisfactory element, principally because of the difficulty of effecting a leak-tight metallurgical bond between stainless steel clad thermocouples and Zr-2 fuel element cladding. During the past month several elements with mechanical seals were tested by thermocycling in TF-7. Two mechanical seals (English Revitts and S/S Navy Seal) were unsatisfactory; the Zr-2 Navy Seal developed a small leak. The Zr-2 Swagelock showed no evidence of any leakage. The evaluation is continuing.

Since all uranium-core thermocouple elements proved to be of doubtful value, a new thermocouple element was designed; it is an all stainless steel system using enriched UO_2 as the fuel material. Since S/S cladding is used, the thermocouples (3) are put in the cladding to increase the sensitivity. Two designs are being considered for the in-reactor. Since the literature is rather devoid of information on deposition of crud on Zr-2 and S/S, one element (during the in-reactor test) will have a Zr-2 jacket over the S/S clad and the other just the S/S clad. The heater elements will be N-Reactor inners so a reasonable correlation should be achieved.

The design basis of the elements was a 600,000 Btu/hr ft^2 heat flux at the surface and a constant heat flux. This was achieved by using thick S/S cladding, thus requiring a high enrichment. With the high enrichment, the expected burnup rate is about 5%/month. However, with a maximum burnup rate of 10%/month the heat flux will drop only 46,000 in two months and 11,000 in one month. The maximum drop in heat flux for two months is less than 10%.

Leakage of Dibasic Ammonium Citrate Into K-2 and K-4 During the Cleaning of K-1. During the cleaning of K-1 with inhibited dibasic ammonium citrate, on 1/29/63, some of the cleaning solution was inadvertently valved into the K-2 and K-4 loops while they were operating in-reactor with N-Reactor fuel elements. Since it is not impossible that a similar type incident could happen during N-Reactor operation, a special sampling program was immediately initiated. This will provide valuable information on clean-up rates, water quality control, and fuel element damage. K-2 was operating with N-Reactor inners and sleeves and K-4 with complete N-Reactor elements. This "test", although not planned, will provide valuable information on the introduction of decontamination solutions into a reactor during operation.

When the incident occurred, K-2 was operating on feed-and-bleed and received the largest amount of citrate contamination. K-4 was recirculating through the ion exchange clean-up system and thus received only minor citrate contamination.

The pH in K-2 decreased from 10.1 to 6.5 within 10 minutes and the specific resistance decreased to approximately 1300 ohm-cm. There was an immediate activity release in the loop, making sampling and analysis very difficult - some of the early water samples read as high as 300 mr/hr. The loop was put on maximum feed and bleed with high-purity deoxygenated pH 10 lithiated water. Within 24 hours the NH_3 was gone and the pH, specific resistance, iron and turbidity

were within normal operating limits. The total solids remained slightly high for approximately 100 hours but were decreasing steadily during that time. The pH in K-4 dropped to a low of 8.95 and the specific resistance decreased to 4120 ohm-cm. There was no appreciable activity release and therefore no radiation problem during initial sampling.

Instrumentation for N-Reactor. The magnesium versenate analytical procedure for water hardness measurement was adapted for use with a continuous wet chemical analyzer. This procedure is being considered as a possible method of detecting raw water leakage into the N-Reactor secondary coolant system. Laboratory tests indicate the procedure is sensitive to hardness concentrations as low as 10 ppb. Full scale sensitivity of the analyzer is within the 0-100 ppb concentration range. Reproducibility of the procedure is excellent although the accuracy is dependent on the preparation of accurate standards at low concentrations. At present, a 24-hour test is under way to determine the stability and reproducibility of the procedure during automatic operation.

The procedure for hydrazine analysis was modified to permit operation in the 0-10 ppm concentration range. This procedure is being used to measure hydrazine concentrations in the influent stream to a continuous dissolved oxygen analyzer. The purpose of this test is to determine whether hydrazine interferes in the operation of the oxygen analyzer; this is relevant since the analyzer is being considered for use in the N-Reactor secondary system where hydrazine will be present. Results to date indicate that hydrazine does interfere to some extent, although the quantitative effects are not completely known.

Studies are also in progress to determine the feasibility of developing very sensitive colorimetric analytical procedures for zirconium and calcium measurements with a continuous wet chemical analyzer. These procedures appear promising and are being actively investigated.

Effect of Heat Treatment on the Corrosion and Hydriding of Zircaloy-2. It was shown last month that corrosion rates in 400 C, 1500 psi steam were increased to some extent by increasing the heat treatment temperatures from 700 C to 950 C, but the most significant effect was found in the corrosion product hydrogen pickup. All of the Zircaloy-2 samples heat treated in the range of 700 C to 800 C increased in percent of corrosion product hydrogen pickup following the corrosion transition. The Zircaloy-2 samples heat treated at 850 C and higher have not shown any increase after corrosion transition.

A metallographic investigation of the heat treated and autoclaved samples has shown a definite difference in the distribution of intermetallics directly associated with heat treatment temperatures. Samples heat treated at 850 C and higher have a large concentration of second phase particles at the grain boundaries. The samples heat treated from 700 C to 800 C show virtually no intermetallic precipitation at the grain boundaries. The heat treating temperature at which this change appears to take place coincides with the temperature ranges at which the effect on corrosion product hydrogen pickup is noted. Hydrogen pickup in Zircaloy-2 following the corrosion transition may be affected by the distribution of the second phase material. This theory appears to be supported by results obtained on crystal samples of bar zirconium. The crystal bar zirconium samples, which do not contain second phase material, have shown no increase in post transition hydrogen pickup rates for any heat treatment temperatures studied.

Nickel Plating of Uranium. Anodic etching of uranium in 10% HNO_3 and nickel plating of the pieces in sulfate and sulfamate baths is being studied. The experiments have primarily been with thin uranium washers which expose mostly the end grain of the uranium. The pitting of the end grain and the subsequent mechanical adhesion of the nickel plate was found to be related to the anodic current density during the etch. The higher current densities (1000 amp/ft^2) produced the most uniformly roughened surface and hence the best mechanical adhesion. A single test with a small cylinder (mostly cylindrical surface exposed) indicated fair mechanical adhesion after etching at 100 amp/ft^2.

The room temperature anode current efficiency for several of the nickel sulfamate baths ranged between 5 and 20% depending on whether citrate, tartrate, or glycolate ions were used as a buffer. Ultrasonic agitation did not appreciably affect the anode efficiency but did increase cathode efficiency. A bath consisting of nickel sulfate (200 g/l) ammonium citrate (40 g/l) and ammonium chloride (5 g/l) gave excellent anode and cathode efficiencies at room temperature. The uniformity of plating at the base of micro pits on uranium is being investigated for the more promising baths.

3. Graphite Studies

Graphite Burnout Monitoring. Small graphite burnout monitors in channel 3580 at F-Reactor from August 26, 1962 to January 29, 1963, showed low oxidation rates. The highest rate was 1.5% per 1000 operating days (%/KOD) at a distance 72 inches into the graphite stack. A profile of the rates showed a sharp peak at this location and a broader one near the center line of about 0.5%/KOD.

[redacted] [redacted]

Monitors in channel 3066 at KE-Reactor from October 31, 1962 to January 8, 1963, showed one peak of 27%/KOD at about 80 inches, which is in the 0.130-inch coring zone. The maximum temperature occurs in this zone. In the same area monitors protected from the reactor gas by a 3/8-inch wall of graphite measured about 15%/KOD. This is a 7-fold increase over the highest rate previously found for protected monitors at KE-Reactor.

Monitors in three channels at C-Reactor from July 3, 1962 to January 20, 1963, were measured. In each channel there was one peak. The rates and locations were 8%/KOD at 120 inches in channel 1889, 8%/KOD at 80 inches in 1960, and 1%/KOD at 80 inches in 2780. Both the displacement of the peak in 1889 and the lower rates in 2780 are not unusual for C-Reactor. The displacement may be due to skewed temperature, but the lower rates in the higher channel is unexplainable; however, borescopic examinations of VSR channels at C-Reactor have confirmed that greater oxidation occurs in the lower regions of stack. Channel 2780 was not recharged because of a decision to retube it for production use.

N-Reactor Graphite Irradiations. The first second-generation capsule, H-4-2, in the series of long-term irradiations of N-Reactor graphite was removed from the GETR on January 22, after operating successfully for 104 effective full-power days. The sample exposure for the samples at the peak flux position is estimated to be 4.2×10^{21} nvt, $E > 0.18$ Mev, equivalent to about four N-Reactor exposure years.

During the disassembly at Vallecitos, it was found that the ends of the four samples in the top capsule position had been broken off. Length data from this group, which comprised two first and two second generation samples, will not be lost since the secondary measurement system based on accurately spaced drilled holes was not disturbed.

It was also found that the capsule had apparently received a sharp, shock-type blow some time after its manufacture, since the top six cooling rings were displaced downward about one-quarter inch from the position indicated by the final capsule radiograph. That the displacement happened prior to irradiation was shown by the position of the carbon deposits outlining the cooling rings relative to the rings themselves. How and when the displacement occurred could not be determined. The bullet nose and the capsule bottom surface showed no evidence of damage. The shipping box used was undamaged, and the personnel handling the capsule prior to and

during installation has no recollection of the capsule receiving a sharp blow. Fortunately, the displacement had no effect on capsule operation and did not damage the samples. All samples and flux monitors were recovered and returned to Hanford. Measurements are currently under way on both.

Some of the carbonaceous material typical to these capsules was recovered off the inner wall of the capsule shell. Analyses are under way to determine if some of this material may be carbon suboxide.

On January 22, the third second-generation capsule, H-6-2, in the series was installed without incident in the GETR. It is currently operating satisfactorily with all thermocouples functioning properly. Sample temperatures match those in previous capsules. The H-5-2 capsule continues to operate satisfactorily. The thermocouple which had apparently failed was also found to be broken inside the terminal box and has been repaired. Currently all nine thermocouples are functioning properly.

To aid in evaluation of flux-monitor data from these capsules, a two-dimensional transport-theory code, 2 DXY, is being set up to analyze the GETR neutron spectra. The results from this computation will be compared with the GNU-II spectra derived previously and with calculations being made at Vallecitos.

Stored Energy. In September 1962, 19 graphite cores were removed from D-Reactor for total stored energy (TSE) determinations. X-ray examination at that time disclosed extreme broadening of the 002 diffraction line, an indication (based on past correlation of line shift, crystal disorder and stored energy) of an increase in stored energy. This implication was not confirmed by recent National Bureau of Standards TSE determinations on five of these samples, as they showed only small changes in comparison with 1958 samples.

Whether these results imply an approach to steady-state or an interruption of stored energy build-up by annealing is unknown, but, hopefully, stored energy release spectra will tend to clarify this point. The principal significance of these preliminary results is that a hazardous stored energy condition probably does not exist in the reactors typified by D-Reactor.

Water Formation Over Graphite. The reaction rates of the reverse water gas reaction, $H_2 + CO \rightarrow C + H_2O$ are being measured to determine to what extent this source will provide water to oxidize Zircaloy process tube in a graphite moderator stack. Dry mixtures

of He - 2% H_2 - 1% CO were passed over two graphite bars in series at various flow rates and the increase in water concentration measured on an electrolytic hydrometer. The results show no measurable water formation at 600 C and below, although equilibrium calculations predict sizable water contents should have been found. At 700 C about 1 ppm of water was found. Above 700 C water formation rates increased rapidly reaching a peak at 900 C and then decreasing at 1000 C because of a lower equilibrium water content at higher temperatures. Additional data at other gas compositions will be required. However, these initial results indicate the reverse water gas reaction will not supply sufficient water to maintain normal oxidation of Zircaloy at graphite tube block temperatures below 700 C unless radiation accelerates the reaction kinetics. The results suggest an in-reactor experiment to measure the reaction kinetic may be necessary.

Resistance Measurement Capsule. An in-reactor capsule designed to measure electrical resistance between zirconium oxide film and graphite at 450 C failed to respond to controlled atmospheric conditions during preliminary laboratory testing. Instead of deoxidizing in dry helium gas, the zirconium samples oxidized beyond a point of reuse. This failure necessitated rebuilding the capsule. The new capsule has been assembled and is now being tested in the laboratory. Resistance measurements are showing satisfactory response patterns and will be compared to data obtained during in-reactor testing. Preliminary testing should be completed within another week.

A duplicate resistance measurement apparatus was assembled for testing in a laboratory tube furnace.

4. Process Tube Development

Stress Rupture Testing of Reactor Pressure Tubes. An extensometer is under development for measuring the circumferential strain of a tube specimen undergoing stress rupture test. It has employed an electronic readout system with limited success. As a replacement or auxiliary readout an optical system is now under evaluation. A chain link circumference gage with its steel scale calibrated in 0.010" increments is attached to the specimen inside the furnace and suitably illuminated. The image of this scale is read through a system of three mirrors using a Questar telescope placed outside the door of the concrete test cell. The total optical path is about 15 feet. Using a magnification of 40 X, the scale has easily been read with an accuracy of ± 0.002". On an N-Reactor tube section

this represents about ± 0.02% of the circumference being measured. The development of the optical system looks very promising.

This facility has continued to operate at capacity the past month. An initial test to determine the maximum temperature capability of a burst test furnace was made. A piece of 3-inch schedule 80 carbon steel pipe ruptured at 3500 psi when the temperature was elevated to 1100 F. Further tests are planned using stainless tubing to determine the maximum temperatures possible with the existing furnaces at supercritical water conditions.

5. Thermal Hydraulic Studies

N-Reactor Boiling Burnout Studies. Boiling burnout experiments were conducted in the heat transfer laboratory to determine whether a compressible fluid volume connected to the inlet of the burnout test section would affect the conditions for which "upstream burnout" have been obtained. A 10-gallon surge tank, partially filled with nitrogen gas, was connected by a tee to the inlet of the 12-foot long by 0.44 inch ID uniformly heated test section. Thermocouples spot welded to the outer surface of the tube at 1-foot intervals were used as burnout detectors. Runs were then made at 1500 psig with various coolant conditions, including those which had resulted in upstream boiling burnout in previous experiments without the compressible fluid volume.

A total of twenty-one boiling burnout points were obtained. Fifteen of these data points were obtained under conditions which had been investigated previously. Upstream burnout occurred at mass flow rates of 5,000,000 to 7,000,000 lb/hr-sq ft when inlet coolant conditions were near or above saturation conditions. Monitoring of flow rate and pressure drop by high speed response instruments showed that flow rate or pressure fluctuations were negligible and were not the cause of upstream burnout. Test results showed that the presence of the surge tank or compressible volume did not change the conditions under which upstream boiling burnout was obtained.

Six of the twenty-one data points were obtained with steam quality at the inlet of the burnout test section and represent a new region of investigation. Outlet steam qualities of about 65% were obtained with mass flow rates of 2,000,000 and 5,000,000 lbs/hr-sq ft. The data for the 2,000,000 lb/hr sq ft flow rate show that burnout heat flux continues to decrease slowly as the outlet quality increases from 20% to 60% and that upstream burnout conditions did not occur at this flow rate. For 5,000,000 lbs/hr sq ft, upstream burnout occurred for all outlet steam qualities above about 15% and the

burnout heat flux did not decrease with increase in outlet quality over the range investigated. The burnout heat flux for an outlet quality of 65% was about 30% greater than the burnout heat flux at an outlet quality of 15%. For this highest outlet quality, burnout occurred within three feet of the inlet end of the 12-foot long test section.

These experiments were conducted while using a recently installed coolant pre-heater which can provide quality steam at the inlet of burnout test sections. The use of this pre-heater required that the computer program for analysis of laboratory data be modified. Modification also included changes to allow use of punched paper tape for input of operating data to the computer.

Present Reactor Studies. Major modifications have been made to the low pressure heat transfer apparatus. These modifications included the installation of a deaerator, a demineralizer, and a new preheater to provide greater flexibility and control over the coolant conditions which are used in heat transfer studies with this apparatus.

Critical Flow Studies. A report, HW-76582, was written to describe the critical flow behavior of short tubes (L/D less than 6) carrying initially compressed water to a region of low pressure. The principal phenomenon studied was the two separate and distinct occurrences of choking in a short tube under certain flow conditions. One choking condition was observed experimentally and visually to occur in the region of the vena contracta near the inlet of the tube, and the other was observed to occur at the outlet of the tube. The first mentioned choking condition could be of principal interest in reactor flow systems since it could limit the flow through a pipe constriction even though the static pressure on both sides of the constriction is greater than saturation pressure. Choking at the outlet of the tube would be of interest in the analysis of coolant loss after piping rupture since many flow configurations resulting from ruptures and breaks can be idealized as short tubes. Included in the report are descriptions and discussions of the flow patterns observed visually through glass test sections. Computer calculations were used to evaluate the present data and compare it with empirical correlations of other investigators.

B. WEAPONS - 03 PROGRAM

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

C. REACTOR DEVELOPMENT - 04 PROGRAM

1. Plutonium Recycle Program

Fuels Development

Inspection of PRTR Fuel Elements. Three irradiated PRTR fuel elements were examined in the Fuel Element Examination Facility (FEEF) and two in the PRTR storage basin to assess their general condition and to attempt to locate cladding defects in elements which were suspected leakers or perhaps were contaminated externally with fissionable material. The general appearance of these irradiated elements was good and no serious cladding defects were found.

The center support stem was accidentally twisted from the gussets of the top end bracket of a UO_2-PuO_2 element (5120) during manipulation in the FEEF. Subsequent examination of the element in the basin revealed that the surface condition of the rods is good, that there are very few scratches and no loose wires. Careful examination of the broken gusset welds showed them to have been sound. Later destructive evaluation demonstrated that the tensile strength of the end brackets is more than three times design strength. The brackets were not designed to resist twisting, and only moderate torsional forces (35-70 foot pounds) were required to break the stems from the gussets. This relative weakness allows failure in a component not containing fuel, and maintains the rod bundle integrity.

Two other suspected leaky UO_2-PuO_2 elements (5162 and 5113) were examined in the FEEF and their general surface appearance also was good. No serious surface defects were detected. The clip-on, extended surface wear pads appear to be performing satisfactorily and only limited superficial wear has occurred. Some of the spot fusion welds on the spiral spacing wires appeared corroded, probably as a result of inadequate inert gas coverage during welding.

An Al-Pu element (5088), suspected to have a cladding leak or contamination, according to stagnant water tests conducted in the reactor, was examined in the basin. No defects were observed. The general appearance of the fuel rods is good. However, a loose spiral wire wrap on one fuel rod indicates mechanical core-clad interaction.

Stagnant water tests were performed on three elements in the relatively uncontaminated environment of a special stagnation tube in the PRTR storage basin. Two were leaker suspects as previously indicated by stagnant water tests in the PRTR and one was a

presumably sound element comparison standard. The tests do not differentiate between an element with a cladding leak or an element which has fissionable contaminants trapped on its surface. An analysis of water samples for several iodine isotopes indicated that only the Al-Pu element is suspect. This element will be further examined.

Re-padding UO_2 PRTR Fuel Elements. Extended surface, clip-on wear pads previously attached to several PuO_2-UO_2 fuel elements continue to perform well. A similar modification is planned for thirteen UO_2 fuel elements which will be kept in the reactor for extended life tests. The clip-on pads, as well as the remote assembly equipment, are being fabricated. The designs of the pads and tools are similar to those used for PuO_2-UO_2 fuel elements, with design changes made where necessary to accommodate different fuel element dimensions.

PRTR Fuel Element Fabrication. Five swage compacted, mixed oxide PRTR fuel assemblies were fabricated to meet February and March refueling needs.

An output of 100 pounds per day of UO_2 of a mix sized for swaged PRTR rods was achieved by adjusting screeners for more efficient operation and by using vibratory feeders for even flow of material to the screens. Use of the feeders also reduced manpower requirements.

A gamma scan test for nondestructive determination of plutonium segregation in a fuel rod continued to look promising. The gamma scan correlated with autoradiographs in all cases, and a rod can be checked in eight minutes by this method, versus 72 hours by autoradiography. Further testing is planned to determine the quantitative accuracy of gamma scanning.

PRTR Fuel Element Spacing Members. Tests of Zircaloy-2 wire used as rod spacing members on PRTR fuel elements indicate that a 90° bend does not cause circumferential cracks and has only a small effect on the surface imperfections normally observed.

Twenty-four wires were examined, including three wires randomly sampled from the batch of wire used in the 1000-series UO_2 fuel elements now in the PRTR. Twelve of the wires were bent 90° by the same method used in fuel element assembly. Visual examination at 40 X magnification revealed longitudinal pre-test microcracks less than 0.001 inch deep in 20 of the wires. Such cracks were widened only very slightly by the 90° bending. Photographs are being made.

High Energy Impaction of UO_2-PuO_2. Uniform plutonium distribution was found in UO_2 - 2.5 w/o PuO_2 formed by high velocity impact. Four randomly selected 10-gram specimens from one container were pulverized separately and divided into two parts. The resulting eight samples were analyzed for plutonium and uranium. Averaged results for each of the four particles are shown in the following table.

Plutonium and Uranium Distribution in UO_2-PuO_2

Sample	w/o Pu	w/o U
1	2.08	86.0
2	2.05	86.1
3	2.06	85.8
4	2.09	86.3

Average relative deviation between duplicates was 1.2%. Ceramographic examination of the impacted UO_2-PuO_2 revealed a uniform dispersion of PuO_2 particles in the UO_2 matrix.

Impaction of UO_2-PuO_2. Detailed design and cost studies were initiated for facilities to rapidly process impacted UO_2-PuO_2. System capacity is estimated to be 100 lbs/day. Crushing tests on impacted material using an adjustable pulverizer indicate 75% of the crushed material can be utilized for vibration compaction. The remaining material would be recycled. Previous methods gave only 50% first cycle utilization.

Remote Fabrication Studies. A manipulator system, consisting of two boom-crane mounted manipulators, was installed in the remote fuel element fabrication enclosure. Studies to determine the optimum T.V. camera placement and enclosure lighting are in progress.

Remote Welding Chamber. A small chamber for remote welding of end closures and end fittings of irradiated, rejuvenated fuel element rods was designed and is being constructed. The chamber will provide an inert atmosphere sufficient to assure good corrosion characteristics of the weld area. It will also enable rapid and accurate welds, with a minimum of adjustments between each weld.

Pre-Assembled 19-Rod Cluster. A new hanger assembly was designed and a prototype is being fabricated which will accommodate slight misalignment of cladding tubes in pre-assembled 19-rod cladding. This end hanger assembly, in conjunction with a new end cap concept, will permit a fuel element to be completely fabricated from the pre-assembled 19-rod cladding bundle.

Fuel Refurbishing Experiments. A water tank for experimental fuel element repair studies was completed and installed on one side of the remote fabrication facility.

Fuel Element Rejuvenation. Meleager calculations were made by R. I. Smith of PRTR Technical Planning Operation to determine the amount of low exposure plutonium needed to restore a spent mixed oxide fuel element to its initial reactivity value. Preliminary calculations show a minimum of 141 grams of low exposure plutonium per PRTR fuel element will be required for rejuvenation of a 5000 MWD/T_u exposed fuel element. The second irradiation life would be approximately 4400 MWD/T_u, for a total of 9400 MWD/T_u.

In-Reactor Enrichment Test Element. Components are being fabricated for an exploratory in-reactor test of an experimental fuel element to which additional fissionable material can be added during operation. The fuel element will contain a short section of hollow UO_2 powder which can be released into the active zone by electrical activation of a solenoid. This test is expected to reveal data on gas release and power increase occasioned by such a fuel element rejuvenation plan.

Irradiation of UO_2-PuO_2 Mixtures. Samples for analyses were removed from sharply defined zones of markedly different radioactivity levels observed on a cross-section of an irradiated $(U,Pu)O_2$ fuel capsule. An autoradiograph was prepared by a modified GE-APED method which employs a beta-sensitive, high resolution film plate. A micro-drilling unit, devised by Radiometallurgy Laboratory personnel, was used to obtain the 0.5 mgm analytical samples. The average exposure of the 0.5-inch diameter $(U,Pu)O_2$ sintered pellets (GEH-14-86) is 5600 MWD/T of $(U,Pu)O_2$ or 1.66×10^{20} fissions/cc. The samples from the zones of lowest activity ($\frac{1}{4}$-inch diameter, at center) and of highest activity (1/32-inch wide annular ring, $\frac{1}{4}$-inch ID) provided contact dose rates of 200 mrem/hr (including 15 mr/hr gamma) and 450 mrem/hr (including 100 mr/hr gamma), respectively. The samples were collected from 0.020-inch in diameter, 0.010-inch deep holes made with a diamond drill. Additional shielding is required for the equipment before additional fuel can be drilled.

Irradiation Performance of MgO-PuO_2 and ZrO_2-PuO_2 Fuels. High resolution autoradiographs have better defined the behavior of fission products and/or plutonia in irradiated MgO-PuO_2 and ZrO_2-PuO_2 fuel pellets. Autoradiographs made from transverse cross sections of irradiated MgO - 3.05 w/o PuO_2 pellets show that the fission products and/or the PuO_2 second phase migrated from the central region of translucent columnar grains. The translucent MgO

grains adjacent to the center void are larger than those more distant. Localized regions of high fission product concentration on the inner surface of the center void, shown by the autoradiograph, were also resolved on high magnification micrographs.

Limited migration occurred in the high temperature MgO - 13.5 w/o PuO_2 samples. However, the indicated maximum operating temperatures were lower than those of the MgO - 3.05 w/o PuO_2 samples mentioned above.

Autoradiographs were made from transverse cross-sections of irradiated ZrO_2 - 2.2 w/o PuO_2 and ZrO_2 - 10.4 w/o PuO_2 pellets. The microstructural features of the irradiated ZrO_2 - 2.2 w/o PuO_2 samples are different from those of the ZrO_2 - 10.4 w/o PuO_2 material although both were supposedly irradiated to produce approximately the same fuel temperatures. No center void or columnar grains were formed and the central region is free of cracks. Short radial cracks are found in the transition region of the fuel. Autoradiographs show that the fission products and/or PuO_2 are severely segregated into randomly distributed islands. Archive specimens of the sintered, solid solution fuel will be re-examined to determine if this unusual behavior occurred during irradiation or if it is an artifact remaining from fabrication.

The microstructural features of the irradiated ZrO_2 - 10.4 w/o PuO_2 fuel are different; center voids and columnar grains were formed and autoradiographs show that fission products and/or plutonia have migrated to preferred radial locations.

Examination of the Zircaloy cladding from a ZrO_2 - 10.4 w/o PuO_2 fuel capsule that failed during irradiation revealed severe hydriding and extensive cracking. In one transverse cross-section could be seen six cracks which completely penetrated the cladding. There was no apparent fuel-cladding reaction and the cause of failure is not yet known.

<u>ThO_2-PuO_2 Studies</u>. ThO_2-PuO_2 pellets were fabricated for several irradiation test capsules. During sintering an unexpected sensitivity to powder and die lubricants, presumably, was observed. Pellet surfaces spalled and cracked. Calcining pre-pressed powders in air at 1000 C eliminated the problem, and sound pellets are now being obtained.

<u>Phoenix Fuel Experiments</u>. Scoping was completed of a test to demonstrate the stability of PuO_2 cermets under the high burnup conditions anticipated for Phoenix fuels. In addition to demonstrating material

stability, the heat generating lifetimes of cermet fuel enriched with plutonium and U-235 will be compared.

Three fuels are included in the test: a high Pu-240, plutonium cermet, a low Pu-240 plutonium cermet, and a fully enriched UO_2 cermet. Matrices being evaluated include zirconium, stainless steel, and molybdenum. Seven-rod clusters, with each rod $\frac{1}{4}$" in diameter and 1-foot long, will contain cermets of approximately 18 volume percent (about 10 atom percent) PuO_2 or UO_2.

The three fuel clusters will be irradiated in series in a common basket. Thermocouples located in front of the basket, between each fuel cluster, and at the end of the basket will, together with the measured flow rate, provide a measure of the heat output of each fuel cluster, to be recorded as a function of exposure.

After irradiation (when the fissionable material is > 50% burned up), the rods will be sectioned, metallurgically examined and the isotopic composition of the remaining plutonium determined.

Corrosion and Water Quality Studies

Migration of Hydride in Zircaloy. Studies to determine the mechanism of charging Zr-2 with hydrogen beyond the solubility limit are continuing, with the maximum temperature increased from 300 C to 400 C. Samples uniformly coated with hydride are cycled daily between 400 C and room temperature, with 16 hours at temperature per cycle. Hydrogen analysis of the center of a sample cycled three times shows 250 ppm, in good agreement with the terminal solubility at that temperature. After 20 cycles, however, a hydrogen concentration of 400 ppm is found, indicating that a mechanism for charging Zr-2 with hydrogen beyond the accepted "terminal solubility" is operative.

Corrosion Cracking of Caustic Contaminated Stainless Steel. Caustic contamination of stainless steel surfaces exposed to steam can produce a variety of stress corrosion cracking. The present experiments examined the effects of contamination by mixtures of caustic and $KMnO_4$. The corrosion samples, 3/8" male flared tubing connectors, were immersed in the contaminating solution, connector assembled, removed from solution, wiped off, tightened. After exposure to a slow flow of 275 C steam at atmospheric pressure the connectors were removed, disassembled and examined for cracks. Selected samples were metallographically examined. The contaminating solutions were 18% NaOH, 18% NaOH plus 0.1%, 0.5%, 1.0%, 2.0%, 3.0% $KMnO_4$ and 18% NaOH plus 3% $KMnO_4$ reduced to MnO_2 with H_2O_2. The fittings were type 316 stainless steel and the flared tubing type 304. The fittings

were type 316 stainless steel and the flared tubing type 304. The fittings trapped an estimated 0.2 ml of solution.

In general, the $KMnO_4$ acted as a very mild inhibitor; however, all the contaminated fittings were cracked in less than 219 hours. Observation of the fittings during exposure showed the fittings contaminated with caustic, caustic plus MnO_2 and caustic plus 0.1%, 0.5% $KMnO_4$ all cracked in less than a day. The caustic plus 1% $KMnO_4$ showed cracks in one day while the 2% and 3% $KMnO_4$ did not show cracks for several days. The location of cracks seemed to be primarily associated with stressed areas of the nut, flare backup sleeve and flared tubing. All cracks were transgranular and resembled those produced by chloride.

Pitting Corrosion of Mild Steel Pipe. Pitting corrosion of a small dead leg exposed to PRTR secondary system water was investigated. Pits had perforated a 135-mil pipe wall in one place and nearly penetrated it in several others. Analyses of the corrosion product and metallurgical examination of the pits are in progress. Radiography of the pipe adjacent to the failed piece as well as several other dead legs will be used to determine the prevalence of such pitting.

Zirconium Concentration Measurements in PRTR Primary Coolant. The program to routinely monitor the primary coolant for zirconium was continued; the results are significant since they may have pinpointed the cause of some of the zirconium fretting corrosion observed previously.

On 2/9 the reactor was not operating, but the coolant was being recirculated with one primary pump on low speed. The zirconium concentration in the coolant at this time was 1.3 ppb. On 2/10 through 2/12 the reactor still was not operating, but the coolant was being recirculated at full-flow, using pumps 2 and 3. During this time (2/10 and 2/11), the zirconium concentration increased sharply to greater than 10 ppb and remained at this value for two days; the exact concentration during this period is not known but is presently being measured. On 2/11 it was observed that several process tubes were vibrating enough to produce an audible signal. On 2/12 the coolant was recirculated at full-flow using pumps 1 and 3 (rather than 2 and 3), and the zirconium concentration began decreasing. The concentration on 2/12 was 0.56 ppb. On 2/13 the reactor resumed operation with the same primary pumps on line and the zirconium concentration decreased further to less than 0.2 ppb. Similar conditions existed on 2/14, and the results indicated no detectable zirconium in the coolant.

These measurements indicate that the process tube vibrations observed on 2/11 were accompanied by zirconium fretting corrosion of the tubes and/or fuel elements. It is possible that the cause of the vibration was the pump combination (2 and 3) used at this time. A cooperative program has been started to monitor the zirconium concentrations in the coolant under various operating conditions. The program is designed to determine: (a) how effectively the fretting corrosion can be monitored by following the concentration of zirconium in the coolant; and (b) what effect the operating variables (pump combinations, temperature, water quality, oxygen concentration, etc.) have on fretting corrosion.

Rupture and Decontamination Tests in IRP. The seventh test was run in the IRP with a ½" $MgO-PuO_2$ fuel rod section from the PRTR ruptured fuel element. The fuel section required 21 hours at 300 C to dissolve the core. Activity present in the filter at the end of the run was 10 r/hr. A solution composed of 1 M H_2SO_4 and 0.1 M $KMnO_4$ was used for the dissolution step. In laboratory tests in the 308 Building using fired PuO_2 this solution was found to dissolve more PuO_2 than any of the others tested. It was not considered for use in the PRTR because of the high carbon steel corrosion rate.

At 20 C the filter activity rapidly dropped to 6.0 r/hr after five minutes. During an additional 15 minutes, the filter activity began increasing to 6.4 r/hr. The temperature was raised to 50 C, and the filter activity continued increasing to 6.6 r/hr after 30 minutes. The temperature was then raised to 80 C, and the filter activity rose to 6.9 r/hr after 40 minutes. Two rinses of the loop decreased the filter to 6.4 r/hr.

A peracetic-peroxide-oxalic solution lowered the activity to 4.0 r/hr after one hour at 80 C. Alkalines permanganate for one hour at 105 C followed by Turco 4518 and ammonium citrate for two hours at 85 C lowered the filter activity to 2.0 r/hr. Repeated application of the AP and Turco 4518 and ammonium citrate solutions did not lower the filter activity.

Radiation Surveys. A complete radiation survey of PRTR is being made each time the reactor goes down. Recently, two leaky fuel elements were present in the PRTR core. Post-shutdown radiation levels were much higher than normal. After the leakers were removed, the radiation levels were decreased.

Fretting Corrosion. The fretting studies in TF-7 on the PRTR fuel element with detachable end bracket supports were terminated after three additional weeks of operation with induced vibration. Two

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weeks of operation was at 35 psi and one week at 19 psi air input to the vibrator. Total exposure at discharge was 89 days. A slight amount of oxide was found to be removed from a few individual rod wire wraps and the end bracket supports but no measurable metal loss was associated with these areas. A 360° support area fuel element has been charged.

Thermal Hydraulic Studies

Boiling Burnout Limits for Increased Tube Powers for PRTR. Calculations were made, based on laboratory heat transfer experiments, to determine the boiling burnout limit of a PRTR Mark I fuel element with a coolant flow rate 50% greater than the design value. The calculations assumed the worst case for all process and design variables for which a quantitative degree of variation could be assigned. The boiling burnout heat flux obtained with electrically heated test sections was divided by 1.85 to provide a safety factor to account for variations and errors which cannot be quantitatively described. Under these conditions, the tube power limit would be about 2500 kw; boiling burnout would occur about two feet from the downstream end of the fuel element; the heat flux at the boiling burnout location would be about 600,000 B/hr-sq ft; and the maximum heat flux at the center of the fuel element would be about 660,000 B/hr-sq ft. The calculations were made in support of a program to increase the PRTR tube power limit to 1800 kw.

Reactor Components Development

BMI Creep Testing Program. Provisions are being made to extend the Zircaloy-2 creep testing program at the Battelle Memorial Institute. This extension will carry the creep testing work through April 1963, and will provide some short term test data concerning: (1) the effect of a heat treatment analogous to an autoclave treatment, and (2) the effect of grain size on the creep behavior of annealed and 15% cold worked strip. Some data concerning the effect of a simulated autoclave heat treatment, 48 hours in vacuum at 750 F, on the creep strength of annealed material tested at 550 F, were obtained and these data indicate that this heat treatment substantially improves the creep strength. In the as-annealed condition, strip material with a grain size of 40-50 microns was tested at 550 F and about 25,000 psi. It exhibited about 4% primary creep strain and failed in a few hours. Identical material, except for a 48-hour heat treatment in vacuum at 750 F, sustained a creep stress of 35,000 psi at 550 F and exhibited a primary creep strain of about ½% and the stress rupture life is expected to substantially exceed 1000 hours. By comparison, pressure creep test at 34,800 psi and 550 F on auto-

claved PRTR tubing produced about 0.9% creep strain in comparable test periods. These data strongly suggest that annealed Zr-2 is artificially age-hardenable and that much of the differences in the strengths of tube and strip material heretofore observed may be in part attributable to age-hardening effects rather than preferred orientation textures alone. Further test data are needed to confirm these early observations.

In-Reactor Monitoring. Final checkout tests runs on the gas gap and tube ID measuring instruments were completed. Although the analysis of these data is not complete, it appears that the instruments are satisfactory. Upon attaching the dial gage, which is used for fretting corrosion measurements, it was found that there is interference with illumination of the tube ID surface which lies within the field of view of the borescope. Several preliminary experiments with the use of reflectors produced some improvement. At this time, however, an adequate solution to this problem has not been obtained. Further experiments are planned. If these are unsuccessful, it may be necessary to obtain a new objective lens section which is constructed with a different type of lens system.

Post-Irradiation Evaluation of PRTR Pressure Tubes. Replicas were made of the corrosion marks in the Zr-2 pressure tube that had been removed from PRTR channel 1356 in June 1962. In-reactor measurements had indicated a maximum depth of 26 mils. Depth measurements on the replica were made with a microscope and showed depth of the corrosion mark to vary from 26 to 27.5 mils over the one-inch length where the in-reactor measurement was taken.

Second Generation Mechanical Shim Rod for PRTR. Detailed design of the transmission housings was completed and these three parts have been fabricated. Design and fabrication of the remaining driving head parts was also started.

The Selsyn position readout equipment has not been delivered as scheduled. No further significant progress has been made in finding a capable vendor for the rotary limit switch assemblies.

Shim Rod Environmental Test Facility. Fabrication and installation of the shim rod environmental test facility was resumed. The piping is estimated 95% complete. Electrical work is estimated 65% complete.

EDEL-I Renovation. Last month's report stated that the vibration of the recirculation pump motor and adjusto speed drive was excessive. Contacting the vendor disclosed that the unit had been bolted to a

solid concrete foundation for their vibration check. This type of mounting tends to restrict the vibration and explains why the vibration exceeded factory results when the unit was mounted on the vertical pump. The motor and adjustable speed drive had been rebalanced by Tech Shops and are now being installed on the pump. The new coupling has been received and installed.

Radiographing of the field welds has been completed and installation of the new code approved safety relief valve is 50% complete.

Fretting Corrosion Investigation. Design, procurement, and fabrication of eddy current vibration instrumentation by Physical Measurements Operation continued. An oscillographic recorder to be used with various monitoring instruments is on order. Design of a prototypical PRTR pressure tube and jumper installation for the EDEL-I test pit is 70% complete.

PRTR Rupture Loop Components. The rupture loop in-reactor test section has been removed from the EDEL-I test pit and crated ready for shipment to the PRTR.

A code review of EDEL-II disclosed that two differential pressure transmitters in the facility were rated at 1500 psi although the facility is rated at 3000 psi. Since the rupture loop tests are run at 2100 psig, these cells have been replaced with 6000 psi rated cells and testing of the Grayloc coupling will now be resumed.

Cleanup work to complete the discharge tools and equipment is finished and this material is ready to be shipped to the reactor.

PRTR Pressure Tube Seals. Agreement has been reached to design and install test facilities for PRTR cap and nozzle-to-tube gasket seals in EDEL-I. A testing and development program for these seals is planned to be run in conjunction with operation of EDEL-I in the fretting corrosion investigation.

PRTR Gas Loop Components. The gamma heating test assembly is complete to the point of installing the shield plug, hanger, and thermocouples. All design drawings have been completed. Items yet to be fabricated include a mandrel for removing the test section liner tube and two wrenches for dome seal installation.

Hazards Analysis

Plutonium Recycle Test Reactor. A document discussing the PRTR primary system transients during a total power outage with free convection and boiling of coolant was published as HW-76313. The

document, based on calculations discussed in previous monthly reports, presented the information for review of the General Electric Technological Hazards Council at the end of the month.

Plutonium Recycle Critical Facility. A new set of three-group cross sections has been obtained for use in lattice calculations to arrive at reactivity coefficients for H_2O moderated PRCF experiments. These were calculated using the GAM-1 and TEMPEST codes. Water to fuel ratios of 2.5, 2.0, 1.5 and 1.0 were selected. Fuel sizes which include the present 0.5-inch rod and the smaller EBWR rod were evaluated.

A supplement to the PRCF Final Safeguards Analysis was published.

2. Plutonium Ceramic Fuel Research

High Burnup Irradiations of Plutonium Compounds. Irradiation samples of single phase PuC and PuN are being prepared by grinding arc melted buttons to thin plates 0.25-inch square and 0.060-inch thick. These samples will be irradiated in the MTR to goal exposures between 5 and 50 a/o burnup of plutonium.

PuN-UN Phase Studies. X-ray diffraction analyses of helium sintered PuN-UN pellets show complete solid solubility between the components. The PuN specimen exhibited a 0.25% lattice expansion, perhaps resulting from some PuO_{2-x}-PuN solid solution. Pure UN sintered under identical conditions reacted with oxygen to form UO_2 as an impurity phase. Random deviations from Vegard's Law observed at several compositions are believed to be a result of oxygen contamination.

PuO_2-Carbon Reactions. An 80-gram batch of a two-phase mixture of α-Pu_2O_3 and α'-Pu_2O_3 was prepared by carbon reduction of PuO_2. The original PuO_2-carbon mixture was pressed into pellets and reacted at about 1550 C for 10 hours in flowing helium. The reacted pellets were crushed, re-pelleted, and re-sintered at 1675 ± 50 C for 10 hours in flowing helium.

Electrical Resistivity of Plutonium Carbides. Electrical resistivity measurements are being used to follow phase transitions in the plutonium-carbon system. The room temperature resistivity of a 39.9 a/o carbon sample was 267 x 10^{-6} ohm-cm. Between room temperature and 500 C the temperature coefficient of resistivity was slightly negative, about -0.02 x 10^{-6} ohm-cm/C. An additional 20% decrease in resistivity occurred between 500 and 550 C. According to the phase diagram by Mulford, et al, this abrupt change

corresponds to the $\zeta \longrightarrow PuC + \varepsilon$ phase change, which is indicated as an isothermal reaction. However, the rate of resistivity change is temperature dependent, indicating that this may not be an isothermal transition.

The kinetics of the phase change are now under study.

Studies of PuO_2-MgO Mixtures. Metallographic examinations of PuO_2-MgO specimens melted on a tungsten ribbon show evidence of slight mutual liquid solubility between PuO_2 and MgO. Dendrites of either component were seen in several of the melted specimens.

The system PuO_2-MgO is not compatible with tungsten, as evidenced by the frequent failures of the tungsten ribbon during the melting studies. The tungsten was probably attacked by oxygen evolved from PuO_2 and MgO at high temperatures.

Aluminum Nitride-Uranium/Plutonium Ceramic Mixtures. Aluminum nitride, with only 5 w/o silicon carbide present as a solid solution, was obtained as a -325 mesh powder. This material does not melt but decomposes near 2300 C, and reportedly resists attack by molten metals. Because of its refractory properties and low cross section, this material is of interest as a matrix for uranium or plutonium ceramics.

Compacts of 20 v/o AlN with UO_2 and with UN were prepared both by sintering in hydrogen for 12 hours at 1650 C and by high energy impaction. Examination of treated specimens is in progress. Preparation of AlN (50 v/o) with PuO_2 and with PuN is also in progress.

3. Ceramic (Uranium) Fuels Research

In-Reactor Thermal Conductivity Studies. Eight irradiation test capsules were assembled and sent to NRTS for further comparison of the thermal conductivity and irradiation performance of very large grain and small grain UO_2 at high rod power. The test is expected to provide additional insight into the cause and results of fuel structure changes that occur during irradiation. The test capsules contain various geometrical combinations of sintered pellets and bi- and tri-crystal pellets ground from chunks of fused UO_2.

Molten UO_2 Studies. The heat transfer rate through molten UO_2 was estimated to be 0.05 watt/cm °C. This value is considerably less than the thermal conductivity of single crystal UO_2 near the melting temperature. Theoretical considerations indicate that this should be the case.

The heat transfer rate value was estimated from data obtained during post-irradiation examination of UO_2 specimens containing tungsten marker wires.

Observation of Gas-Solid Reactions by Electron Microscopy. Reactions of UN-W cermets with O_2 were examined by a new technique using reflection electron microscopy. The gas was introduced as a low pressure jet impinging on the specimen surface in the microscope. Careful pressure control allowed continuous observation without interference with the electron beam. Movies and detailed micrographs of the reaction zone showed extensive oxidation of the UN particles in the cermet at temperatures near 1000 C. Tungsten oxidation was relatively slow and the specimen remained coherent during the reaction. This technique is applicable to a broad range of specimens, gases, and experimental conditions.

Release of Fission Products from UO_2 During Transient Heating. A lightly irradiated sample of UO_2 was explosively melted by passing a high current through it. Autoradiographic and radiochemical analyses are being used to determine the relocation of fission products. This technique may be useful in studying the migration and release of fission products under conditions that might be experienced during severe reactor transients. Preliminary autoradiographic results indicate that most of the fission products were retained in the rapidly melted and quenched UO_2.

Resistance Heating of UO_2. UO_2 specimens were resistance heated to the melting point in various atmospheres to develop methods for high temperature annealing. Anneals in air were performed by preheating a UO_2 sintered plate to approximately 1000 C in flowing helium and then changing the atmosphere supply to air. Color motion pictures show anticipated oxidation as well as effects attributable to the higher electrical and lower thermal conductivity of UO_{2+x}. Specimens did, however, maintain their structural integrity at temperatures approaching the melting point for times up to four minutes.

Impaction of UO_2. Densities previously obtained only by using tungsten carbide punches at pressures of 300,000 psi or greater were achieved with a four-inch diameter tool steel punch, using four different starting materials.

Starting Material	Impacted Density (% TD)
Sintered scrap, pulverized and roasted to $UO_{2.01}$	98.8
Sintered scrap (not pulverized), $UO_{2.00}$	99.0
Micronized powder, $UO_{2.09}$	98.0
Ceramic grade powder, $UO_{2.06}$	98.8

These high densities indicate that optimum punch material and design will lessen the need for relatively expensive tungsten carbide punches.

High Temperature Thermal Conductivity of UO_2. Thermal conductivity of UO_2 single crystals relative to polycrystalline UO_2 will be measured above 1500 C using radial heat flow methods. Relative conductivity is determined by comparing the surface temperature of suitably shielded single crystal and polycrystalline pellets as they are internally heated by a central tungsten wire.

Twelve specimens each of single crystal and sintered UO_2 were prepared for these studies from high purity standards. These are cylindrical shells, $\frac{1}{2}$ inch high x $\frac{1}{4}$ inch OD x 1/8 inch ID (with each dimension held to ± 0.0001).

In-Reactor Measurement of Thermoelectric Potential in UO_2. A capsule was fabricated to permit measurement of the thermoelectric potential gradient in a cylindrical UO_2 fuel element during irradiation. Laboratory studies have shown that the large temperature gradients in a cylindrical fuel rod, combined with the semi-conducting properties of the UO_2, create a significant center-to-edge potential in the UO_2. However, effects of radiation on this potential are not known.

The in-reactor experiment consists of a fuel rod containing central and peripheral tungsten electrodes connected to external leads. Potentials generated by the various junctions in the capsule were measured so that these effects can be separated from the thermoelectric emf.

Irradiation of Uranium Monosulfide. A low neutron exposure was found to have a profound effect on the microhardness of uranium monosulfide ($US_{0.98}$). The hardness of the samples after two sequential

irradiations is listed below:

Exposure	Knoop Hardness	% Increase
0	197^{+24}_{-35}	0
$3x10^{16}$ fissions/cc	341^{+40}_{-38}	73
$6x10^{16}$ fissions/cc	386^{+32}_{-36}	95

The lattice expansion rate is also very high - 0.09% after the first exposure.

High Resolution Autoradiographs. Autoradiographs providing exceptional resolution were prepared of irradiated UO_2 crystals and UO_2-W cermets. An alpha-sensitive emulsion is applied as a single-grain thick layer to the specimen surface, then fixed and developed while still in place. The emulsion replicates the surface so that, after stripping, it can be shadowed with metal to bring out surface detail in relation to alpha tracks. Origins of alpha tracks can be determined within less than 5 μ error by this method.

4. Basic Swelling Program

Irradiation Program. Two general swelling capsules were discharged this month after reaching their goal exposure. Two new capsules were charged. The control temperature on each capsule now being irradiated is 625 C regardless of reactor operating conditions. One new capsule is about 15% complete. The new capsule will contain four specimens of uranium in lieu of the normal three specimens per capsule as in the previous irradiations. Two specimens were made from high purity uranium. The third specimen contains additions of aluminum and iron and the fourth uranium specimen contains iron and silicon additions.

All specimens have been machined in the form of split, hollow, thin-walled cylinders as in most of the previous irradiations. A fifth thermocouple has been added to the capsule to provide a complete thermal history of each specimen during irradiation. One of the center specimens will be equipped with two thermocouples located 180° apart.

Post-Irradiation Examination. The uranium specimens that were recovered from two capsules irradiated to 0.15 a/o B.U. at 575 C are being processed for metallography and density. One capsule contained natural uranium while the other contained uranium enriched to 1.44 a/o U-235. The external appearance of the 0.030-inch wall,

split tubular specimens indicated that less damage had been accumulated in the enriched specimens, although all samples exhibited growth. Preliminary density measurements indicated volume increases of from 30 to 40% on the enriched specimens and of about 55% on the natural uranium specimens. Metallographic data will be available shortly.

The irradiated dual enriched, Zircaloy-2 clad uranium specimens that were previously examined after respective post-irradiation anneals at 650, 700, 800, and 950 C for 100 hours were reannealed at 600 C for 1000 hours. It will be recalled that two burnup levels were represented: one set was irradiated to 0.2 w/o B.U. in the shell and 0.01 a/o B.U. in the core while the second set was irradiated to 0.4 a/o B.U. in the shell and 0.02 a/o B.U. in the core. The purpose of the 600 C anneal was to precipitate all gas into pores that may have diffused during the high temperature anneals. The high burnup specimen previously annealed at 950 C is being processed for metallography to determine if a gradient in porosity exists on either side of the uranium-uranium interface. If a gradient is observed, other specimens will be examined.

Non-irradiated control specimens of Zircaloy-2 clad dual enriched uranium have been annealed at 650, 700, 800, and 950 C for 100 hours. The primary purpose of these anneals is to examine the uranium-Zircaloy interdiffusion and compare the non-irradiated diffusion data with that obtained from the irradiated samples. Metallography is in progress.

Thorium. The density values of the three thorium specimens for which hardness and metallography were reported last month are as follows:

Sample No.	Condition	Density (grams per cc)
CT-9	Unirradiated control	11.67 ± 0.01
FX	Irradiated to 0.085 a/o B.U. at ~ 200 C	11.73 ± 0.03
JX	Irradiated to 0.095 a/o B.U. at ~ 200 C and then vacuum annealed at 750 C for 100 hours	11.74 ± 0.04

A slight amount (~ 0.5 a/o) of densification has apparently occurred as a result of the irradiation, probably due to transmutation. The density values correlate with the absence of porosity that was observed metallographically.

[redacted]

Two other thorium samples irradiated to 0.18 and 0.92 a/o B.U. have been polished and etched for metallography. Examination with the light microscope failed to reveal microstructural evidence of irradiation damage. Replicas have been prepared and will be examined in the electron microscope shortly. One of the specimens is cracked but it is possible that the cracking occurred prior to irradiation during fabrication of the specimen. Hardness and density measurements will be made upon the completion of metallography.

Thorium-Uranium Alloys. Four NaK filled noninstrumented capsules were opened and the thorium-uranium tensile specimens contained therein were recovered. These were irradiated at < 300 C to burnups from 0.5 to 1.0 a/o and included alloys of thorium of 1.0, 4.0, and 5.4 a/o uranium. Room temperature tensile testing will be performed and the broken ends will be processed for metallography, hardness and density; post-irradiation annealing is then contemplated followed by reprocessing as above. It is expected that valuable insight into the effect of crystal structure on irradiation damage will be obtained, particularly as part of these samples are solid solutions and others are two-phase.

Restrained Irradiations. In order to gain insight into the influence of restraint on the swelling of uranium, Zircaloy-2 clad uranium rods with selected uranium temperatures, cladding thicknesses, and exposure are being irradiated in NaK filled temperature monitored capsules. Density measurements, chemical burnup determinations, and metallographic examination has been completed on nine unalloyed and three U-2 w/o Zr, Zircaloy-2 clad fuel rods. Density of the unclad samples, volume average temperature of the uranium, and atom percent burnup were determined.

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Fuel Sample	Fuel Composition	Calc. Vol. Avg. U Temp. °C	a/o Burnup (Chemical)	% Density Decrease $\frac{\rho_i - \rho_f}{\rho_f}$
TD 45	Unalloyed	580	0.18	2.7
TD 42	Uranium	580	0.17	2.7
TB 36		580	0.18	3.3
BB 212	Unalloyed	565	0.20	3.5
BB 31	Uranium	565	0.20	2.3
BB33		565	0.20	3.0
TC 23	Unalloyed	580	0.20	2.3
TC 25	Uranium	580	0.19	3.1
TC 27		580	0.21	3.3
244	U-2 w/o Zr	520	0.20	7.7
154	U-2 w/o Zr	520	0.19	7.7
1124	U-2 w/o Zr	520	0.20	8.4

ρ_i - Pre-irradiation density
ρ_f - As-irradiated density.

Calculating from the burnup, it was found that the volume average uranium temperature could run as much as 40 C higher than those shown above which were calculated by a different method. This would place some portions of the fuel center in the beta phase. The certainty of these temperatures cannot be definitely established but based on the metallography to be described later, the temperatures given in the table are more likely those that prevailed during the final period of the irradiation. The density decreases in the unalloyed samples is less than has been experienced in other Zircaloy-2 clad samples. The greater density decrease of the U-2 w/o Zr alloy fuel compared with the unalloyed fuel is consistent with previous experience. Macro examination of four of the fuel samples TC 23, TC 27, 244, and 1124 showed a structure of columnar grains radiating from the fuel center in a zone extending approximately three-fourths of the fuel radius. However, at higher magnification the columnar grains were found to extend even farther, passing into a structure of more equiaxed grains near the edges. The outer edges of this structure are severely distorted by the irradiation and no grain boundaries are visible. The lack of a definite interface which in other irradiations has marked the boundary between the alpha and beta phases is evidence in support of the calculated temperatures given in the table. Comparison of the pre- and post-irradiation metallography of the U-2 w/o Zr alloy reveals that the delta phase coarsened during the irradiation into

a thicker network in the center of the fuel. Progressing towards the edge, the second phase is finer and more uniformly distributed as it was before irradiation. Out-of-reactor control pieces with temperature gradients existing as in the irradiated fuel are not available yet. It is not possible as of now to make any generalizations on the relative effects of temperature and irradiation on changes of the delta phase distributions. Nor is it possible to determine what role the change in microstructure may have had on the observed volume changes.

Electron photomicrographs show fission gas pores up to about one micron in size in the matrix regions. Surrounding much of the delta phase network in the center of the fuel are regions of the matrix approximately three microns wide in which the size and distribution of the second phase appears to change significantly. These narrow matrix regions contain fewer gas pores but their size is 10 to 20 times the size of the pores in the unaffected matrix area. At approximately three-fourths radius, electron photomicrographs show a similar effect but the depletion of the smaller gas pores is not as complete. Further studies of both irradiated and non-irradiated samples of the U-2 w/o Zr are planned.

5. Irradiation Damage to Reactor Metals

Alloy Selection. Procurement of materials to be used as test specimens for the Irradiation Effects on Reactor Structural Materials Program is continuing. Bids have been received for quantities of 406 SS, Inconel 600, Inconel X-750, AM 355, 304 SS, and 348 SS. Inspection reports indicate that the heat of A302B pressure vessel steel has been heat treated and now meets specification. Shipment of this alloy and the A212B alloys is expected by February 26, 1963.

Four graphite boats containing Charpy impact specimens of Hastelloy X-280 and type AISI 406 alloy and tensile specimens of R-27, R-41, R-235, Inconel 718, Inconel 625, and TD nickel were delivered for irradiation. These materials will be irradiated in a 1400 F gaseous environment to exposures of 1.2×10^{20} nvt. Similar tensile specimens of R-27, R-235, Inconel 718, Inconel 625, and TD nickel have also been sent to the ETR for irradiation in the G-7 hot water loop facility.

Procurement of samples of a number of typical refractory metal alloys has been started. Some materials arrived on site this past month. Sheet and rod stock of columbium, columbium-1 w/o zirconium, tantalum, tantalum-10 w/o tungsten, and TZM rod alloy was received. Still to arrive are TZM sheet alloy and tungsten-25 w/o rhenium sheet and rod.

In-Reactor Measurements of Mechanical Properties. The first test of a series of tests is in progress for measuring the stress dependency of Zircaloy-2 creep under irradiation. The test parameters are: 350 C test temperature, 20,000 psi stress, on 20% cold worked Zircaloy-2. The test has accumulated 600 hours of operation during which time two reactor outages have occurred. Although neither the in-reactor test nor the ex-reactor control test has accumulated enough time to establish precise rates, it appears that the creep curves are following the same general pattern established with the series of creep tests at 30,000 psi stress. The creep rate during irradiation at 350 C and 20,000 psi stress has been tentatively established at 1.7×10^{-6} in/in hr. The in-reactor rate during the first outage exhibits an increase from the above rate. These in-reactor rates cannot yet be compared with out-of-reactor rates due to equipment difficulties with the ex-reactor test. All systems in the in-reactor creep capsule are operating within established limits and no difficulties have been encountered with the new heating elements.

Development of a differential control system for the temperature control of the in-reactor creep capsule heaters is continuing. The system uses one of the three thermocouples on the creep specimen to establish the desired temperature. The other two thermocouples are referenced to the control thermocouple and the output difference between the control couple and the reference couple is amplified and used to control the temperature at the reference couple. This differential control system automatically holds all temperatures the same as the one thermocouple on control. The extra controllers now used to control each thermocouple will be eliminated by this system as will the tendency for these separate controllers to buck each other and create temperature oscillations. The input signal to the differential control system will be 0 to 400 microvolts to affect temperature control within one degree Centigrade. As such, the first stage differential amplifier must have extremely high sensitivity and stability. One amplifier tested meets requirements with a stabilized and guarded circuit capable of full scale response to as little as a three microvolt input. A second stage of the system provides voltage to the heater section nearest to the referenced thermocouple. The second stage consists of an amplifier and integrating feedback networks to hold the heater power at a constant value when the error signal is zero. Complete design of the differential control system requires an analysis of both the open loop transfer function and the closed loop control mode on a mocked-up creep capsule. The mock-up has been completed and amplifiers and integrators are being evaluated for the control system.

Preliminary techniques for tensile testing irradiated and un-irradiated heating element wires were developed during the month. Tensile testing is being done to establish the criteria for selection of heating element materials in creep capsules and to determine the cause of failure of previous elements during reactor service. Wire tensile specimens, 3 inches long and 20 mil in diameter, were selected because of the irradiation capsule length and the similarity of wire diameter to heating elements presently being used. The specimen materials include Ni-Cr-Fe-Al, Ni-Cr and 406 stainless steel wires. During testing, power is supplied to the insulated grips and heating is accomplished by current flow through the specimen. Trial wire temperature measurements using small thermocouples proved to be quite inaccurate and led to erratic temperature control. This was a result of considerable heat flow through the thermocouples away from the small diameter wire specimens. To circumvent the temperature measuring and control problem, power density was chosen as an alternate parameter to correlate tensile data. Power density is measured as watts of heat expended for a given surface area of heating element wire in a given environment - in this case, still air. Values of power densities in the range of those encountered in-reactor will be used in those tests. This parameter eliminates the need for thermocouples and simplifies control. Power is manually controlled throughout the duration of the test using a precision wattmeter. Tensile specimen grips were modified to include electrical insulation and to accommodate the wire specimens. Inserts were made with corrugated serrations on the gripping surface perpendicular to the tensile axis of the wire specimen. It is anticipated that this will eliminate specimen slippage. During technique development, it was observed that a change in specimen watt density did give a measurable change in wire properties.

Pre-Post-Irradiation Measurements. The purpose of this program is to investigate the combined effects of irradiation and reactor environment on the mechanical properties of structural materials. Special attention will be given to the determination of mechanical property changes produced in metals by irradiation at elevated temperatures.

During the month, tensile tests were conducted on 48 Zr-2 specimens irradiated at 540 F in the ETR, G-7 hot-water loop. The specimens represented the 0, 20, and 40% cold worked conditions and received estimated exposures of 8×10^{20} and 2.5×10^{21} nvt. Prior to testing, some of the specimens were annealed at 360 C for 10 hours in vacuum. Specimens in the annealed condition prior to irradiation

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showed only a small amount of radiation recovery as a result of post-irradiation annealing. Specimens in the cold-worked condition before irradiation, however, underwent substantial recovery with post-irradiation annealing. For example, the ultimate strength of 40% cold-worked Zircaloy-2 irradiated to 2.5×10^{21} nvt decreased from 128,000 to 102,000 psi with annealing. The corresponding values for this cold-work in the unirradiated condition before and after annealing at 360 C are 109,000 and 103,000 psi, respectively. Substantial cold-work recovery in unirradiated Zircaloy-2 generally occurs above 450 C, indicating that the defect structure of the cold-worked specimens has been altered by irradiation. Apparently, defects associated with cold work have been rearranged by radiation-induced defects to form a more thermally-unstable structure. In this state, overheating above design operating temperatures can be particularly harmful to the strength of cold-worked reactor components. Furthermore, these results would predict an increase in the in-reactor creep rate for cold worked Zircaloy-2 upon reactor shut down. Upon irradiation, creep resistance due to cold work would be reduced from that initially present, and the creep resistance due to neutron damage would become progressively dominant. Consequently, creep during reactor shutdown would be controlled by the rate of damage recovery.

Over 100 flat tensile specimens, fabricated from AISI 406 stainless steel, have been characterized and loaded in quadrant baskets for insertion in the ETR and ex-reactor hot water loops.

A fixture is being built to initiate crack-like flaws in beams by the use of half-cycle bending fatigue. The insertion of an initially dead sharp crack in a material is essential to closely approximating crack propagation situations. Such situations will enable the study of crack arrest behavior.

Notched bend specimens are being machined from Zircaloy-2 plate cold worked 20%. Bending of these notched beams under controlled temperature conditions will establish the crack arrest - crack propagation transition temperature for this material. Also, considerable information from a fracture mechanics standpoint will be obtained both before and after irradiation. Knowledge of applied elastic energy, its distribution in a notched specimen, and its redistribution in the presence of a propagating crack is necessary for these studies. Since available mathematical models for these analyses are intractable, relaxation methods are being applied to analyze the states of stress imposed during the bending of notched beams.

Damage Mechanisms. The objective of this program is to establish the nature of the interaction between defects present prior to irradiation and those produced during irradiation with emphasis on the role played by interstitial impurities. The investigation is presently concerned with high purity iron and its low carbon and nitrogen alloys.

The evaluation of density measurements by the pycnometric method was continued. The objective of these measurements is to confirm the presence of clustered defects observed in electron transmission microscopy experiments by comparing the measured and x-ray densities of irradiated samples. Such clustered defects would lead to different experimental values of density by an x-ray density determination and direct determination. Experiments on 10-gram samples have improved the precision considerably over that obtained on 3-gram samples. Five determinations of the density of a bar of cold worked iron gave a value of 7.8827 ± 0.0030 g/cm^3 while five additional determinations after a one-hour anneal at 200 C in vacuum yielded a value of 7.8822 ± 0.0030 g/cm^3.

A series of isochronal anneals of cold worked high purity iron has been completed to 800 C. Recovery was followed by hardness measurements. The major recovery step to this temperature occurred between 200 and 300 C and accounted for approximately 40% of the recovery. The balance of the hardening due to cold work recovered in a series of ill defined steps of lesser magnitude below 700 C.

Zircaloy Corrosion and Hydriding. An accelerated out-of-reactor corrosion test of Zircaloy-2 coupons representing different levels of cold work was completed during the report period. The purpose was to confirm, ex-reactor, indications reported previously of an effect of work history on hydrogen pickup for coupons exposed to the ETR flux. Specimens at each level of cold work and cut from the same plate stock material as used for the in-reactor studies were exposed to steam at 1500 psig and 425 C for 140 hours in a static non-refreshed autoclave. Two sets of duplicate autoclaved specimens were exposed in the high temperature test loop so that they went into the accelerated autoclave test with an initial oxide film of about 11 mg/dm^2 of oxygen weight gain. One set of duplicates was exposed in the as-etched condition.

Hydrogen pickup values resulting from this investigation showed no trends as a function of work history of the plate material, although final weight gains and total hydrogen content in the plate were at levels comparable to those found for the in-reactor exposure. The absence of readily observable trends, out-of-reactor, may indicate

that the apparent in-reactor cold work effects are not real or that a subtle mechanism requiring both work history and irradiation is at work.

Additional weight gain data available from 24 Zr-2 coupons (quadrants 63 and 64) irradiated in the G-7 loop of the ETR for about 28 days at an estimated neutron flux of 1.6×10^{14} nv continue to show average corrosion rates, an order of magnitude greater than for exposure out-of-reactor. Although these coupons were irradiated at a higher flux intensity by a factor of about four than any of those analyzed previously, the fact that weight gains are less than 10% higher than the maximum found for lower flux exposure for a shorter time lend further support to a saturation effect of flux in-reactor.

Examination of photomicrographs of the grain structure of six pairs of Zr-2 coupons in the neutron-irradiated and unirradiated condition show an apparent reduction in grain size of the irradiated metal. The coupons compared were cut from the same plate stock and had been subjected to essentially the same thermal history in- and out-of-reactor. Annealed, 10 and 40% cold work conditions and two different plate stocks were represented. In no case was a larger grain size found in the irradiated specimen. An indication of grain size reduction in zirconium following reactor irradiation at elevated temperature has been reported by other investigators (Russian). The grain size differences were not so great as to rule out the possibility that similar variations could be found for specimens taken from different parts of a single plate.

Superalloy Corrosion in High-Temperature Helium. Two tests have been completed which illustrate some of the possible corrosive effects of flowing Grade A helium on superalloys. Tests were conducted at 1900 F, with a linear gas flow rate of about 2 in/sec. Each test was of about 10 days duration. The helium employed has a minimum purity of 99.9%.

In the first test, a total of 14 samples of various materials were exposed simultaneously in the reaction chamber to 180 ft^3 of helium. The oxidation of the lead samples purified the gas to the extent that the rear samples did not show nearly the weight gain, and in some cases (notably Haynes 25), show weight losses. In the second test, a single row of Haynes 25 coupons was exposed to a total of 190 ft^3 of helium. The lead samples showed weight gains and the rear samples showed weight losses, though all samples were oxidized. The oxide films were easily scraped off, and oxide was apparently lost from all samples during the run due to sloughing.

Oxidation of Nickel in High Temperature Oxygen. The oxidation kinetics of Inco 270, a high-purity nickel (99.97 Ni + 0.02% C, trace Cr), were compared with similar data for TD nickel. Sheet specimens of Inco 270 were oxidized in pure oxygen (25 mm pressure) at temperatures of 700, 800, 900 and 1000 C. The oxidation kinetics can be approximately described by a parabolic rate law. A plot of weight gain squared versus time in all cases yields a line with a slowly but constantly diminishing slope. The Arrhenius plot of the first 200 minutes of oxidation yielded an activation energy of 32 k Cal/mole. The previously reported irregularity in the temperature dependence of TD nickel oxidation is not observed. Weight gains for Inco 270 were significantly greater than for TD nickel at equivalent time and temperature. Wlodek proposes that the oxidation resistance of TD nickel can be attributed to the high purity of the nickel matrix. It is possible that the impurities in Inco 270 are sufficient to alter the oxidation kinetics significantly.

Irradiation Damage to Inconel. A section of the Inconel pressure tube from DR-1 gas loop was examined metallographically. The sample was taken from the shield zone of the reactor, several feet away from the transverse break that had occurred in the reactor. The operating temperature at this point varied from 600 to 700 F. The examination revealed a large amount of precipitate at the grain boundaries. This precipitate was present across the entire tube wall. None of the other samples examined to date from this tube or a companion tube has shown this type of precipitate.

Gas Loop Development. As prospective structural materials for the model loop, Hastelloy X and C, Haynes Alloy 25, and stainless steels 316 and 347 have been butt welded in pipe form to themselves and to each other by the tungsten-electrode, inert-gas process. Test specimens cut from 24 butt welded pipes are in various stages of preparation or testing. Examination of the macro and micro structure of the majority of the samples show the welds to be physically sound in confirmation of the previously reported results of radiographic and fluorescent penetrant testing. Analysis of the microstructure is under way.

Initial room temperature bend test of Hastelloy X welded to Haynes Alloy 25 and Hastelloy C welded to stainless steel 347 indicates these welds to be ductile. However, further testing is being held in abeyance until changes are made in specimen geometry and in the testing equipment. Room temperature tensile tests on these same metal combinations indicated a weld efficiency of about 90%. The first stress rupture test at 1800 F of Hastelloy X welded to Haynes Alloy 25 is under way.

The selection of materials for the test section is in progress. The pressure bearing member or outer shell that would be exposed to irradiation will be of stainless steel 347 because there has been extensive experience with the effect of neutron fluxes on the properties of this material. The stainless pressure tube will be protected from the high gas temperatures by insulating and lining materials. Four insulating materials are under survey: the refractory oxides, oxidation resistant carbides (SiC), different forms of carbon or graphite, and metal honeycomb or cloth structures. Those commercial products having the best combination of low density (for low gamma heating), low thermal conductivity, and structural capacity at elevated temperatures are being ordered for test and evaluation.

6. Gas Cooled Reactor Studies

Hot Capsule Temperature Calculations. Last month calculated temperatures were reported for the graphite irradiated in a helium atmosphere in "hot capsules" in the ETR and MTR. Additional calculations have been made to see how temperature is expected to vary as a function of gas atmosphere (helium at 1 atm., nitrogen at 1 atm., and vacuum), gamma heating rate (2 through 20 watts/gram), and aluminum surface emissivity (0.2 and 0.3). The peak temperatures calculated for the specimens in the three atmospheres are shown in the following table as a function of gamma heating rate for an aluminum emissivity of 0.2.

Calculated Maximum Temperatures of Hot Capsules, °C

Gamma heating rate, watts/g	Atmosphere		
	Helium	Nitrogen	Vacuum
2	240	570	760
4	400	800	960
6	550	960	1100
8	670	1090	1220
10	770	1190	1300
12	870	1270	1380
14	960	1350	1450
16	1040	1420	1510
18	1120	1480	1570
20	1180	1540	1620

Some capsules after irradiation show blackening of the inside aluminum wall. Since this is likely to increase the emissivity of the

wall above 0.2 (which pertains to slightly oxidized aluminum), additional calculations were made using 0.3. The temperatures calculated were lowered about 100 C. This change is considered the maximum possible due to surface blackening.

Previously, samples have sometimes been irradiated in vacuum. Since graphite is known to out-gas on heating, temperatures calculated for a nitrogen atmosphere should be considered as more closely approximating the temperature of irradiation. The values for the temperature of the graphite at one atmosphere nitrogen pressure are probably reasonably valid for pressures down to a few millimeters.

Speer Research and Development Contract. Recent studies have indicated that a relationship may exist between dimensional change during processing of graphite and the radiation-induced contraction. To better define this relationship a series of samples will be graphitized by the Speer Carbon Company at rates ranging from 50 C/hr to 30,000 C/hr. Previous work conducted by Speer indicates that the resulting volume changes during processing will vary from a small volume contraction to a volume growth of approximately 15%.

The influence of the degree of binder graphitization will be studied in another series prepared with pre-graphitized filler coke and heat treatment of the binder-filler mix to seven final temperatures ranging from 700 C to 2900 C.

The final sample set will be prepared by combining processing techniques and raw materials which are known to produce high shrinkage during processing. The purpose of this set will be to determine the extent of reduction in radiation-induced contraction that can be achieved by this approach.

EGCR Graphite Irradiation. The fifth capsule, H-3-5, in the long-term series of irradiations of EGCR graphite continues to operate normally in the GETR. It has been irradiated for 65 effective days and is scheduled for removal the last week in February.

Graphite Compression Test. The five 300-psi compressive stress boats and the 100-psi tensile boat could not be discharged in December due to unavailability of monitoring personnel at the K-Reactors. An attempt to discharge these sample boats in January was halted after removal of the tensile boat and three of the 300-psi compression boats. Contamination of the exterior surface of the boats by deposition of radioactive materials while in the 2C test hole at KW-Reactor resulted in high activity of the boats and forced discontinuance of the discharge. Discharge of the remaining two boats and

recharge of the five 150-psi compression boats is planned for mid-February.

Hot-Test-Hole Irradiation Facilities. Three test holes, 3A, 3C and 3D, in KE-Reactor are being converted for high-temperature graphite irradiations of the type now conducted in the 2B and 2C hot test holes in KW-Reactor. An additional feature being incorporated will permit monitoring of graphite temperatures in a dummy boat located in a position normally occupied by sample boats. Thermocouple leads will extend to the inner rod room where they will not interfere with normal sample charge-discharge operations. The added facilities will be utilized for irradiating tensile and compressive strength specimens, creep capsules, and large graphite samples for study of sample-size effects.

7. Graphite Radiation Damage Studies

Boronated Graphite Irradiations. Capsule No. 5 containing six samples of graphite with 6 to 9% boron was irradiated in the Hanford Snout II facility. The exposure was estimated to be in the range 2 to 2.4 x 10^{19} nvt (E > 0.18 Mev) and the temperature of irradiation was 590 C. The results of the irradiation are shown in the following table.

Irradiation of Boronated Graphite

Material	Orientation	Nominal Wt. % B	ΔL/L, %	ΔW/W, %
NCC "Black"	Transverse	5	+ 0.007	- 0.003
NCC "Black"	Longitudinal	5	- 0.011	+ 0.071
NCC "Grey"	Transverse	5	- 0.010	+ 0.117
NCC "Grey"	Longitudinal	5	- 0.018	- 0.125
NCC "Grey"	Transverse	7	- 0.008	+ 0.019
GLCC	Transverse	7	+ 0.110(?)	+ 1.66

In the above table, all samples except the GLCC sample were made by introducing the boron as B_4C during the processing. The GLCC sample contains boron as sodium tetraborate and, therefore, is not directly comparable with the other samples. In the case of the samples of graphite containing B_4C, two varieties were tested - "black" and "grey". The "black" material contains B_4C as discrete particles, whereas the "grey" material incorporates the B_4C more homogeneously in the matrix. The base graphite for NCC samples is similar to TSX. As seen in the table, the "grey" material shows slightly more damage (contraction) than does the "black", probably as a direct

result of the closer proximity of the graphite lattice to boron fission events. More data are necessary to determine the exact significance of the effect. As with non-boronated graphite, the transverse samples either expanded slightly or contracted less than the corresponding parallel sample.

The sodium tetraborate sample was found to be coated with a white substance which probably affected the length measurement. This behavior has been noted in previous irradiations at 30 C. The standard deviation of measurement was six times the standard deviation for the other samples. Also, a large positive weight change was observed, which is consistent with previous irradiation experiments.

Additional work is being performed to experimentally determine the heat generation of boronated graphite in a known neutron flux.

Radiation Effects in Lampblack-Based Graphites. Lampblack-based graphite samples originally processed to 3000 C and heat-treated to that temperature after cutting to sample size, have completed two cycles of irradiation at 650 C and subsequent anneal at 3000 C. The maximum exposure of 2580 MWD/AT_K produced a linear contraction of 0.27%. A total of six observations indicated that each anneal removed 19 ± 2% of the contraction accumulated in the sample since the previous 3000 C heat treatment. The present observations extend data to higher exposures and include the effect of annealing after the second irradiation. There seems no doubt that high-temperature radiation-induced contraction in this material consists of annealable and unannealable components.

Samples which had twice been irradiated and annealed were of the same length or shorter than samples irradiated to the same total exposure with no intermediate anneal. The results indicate that annealing may increase both the annealable and unannealable contraction during subsequent irradiations. Further irradiations are scheduled for these samples.

Annealing of Irradiated Graphite. A sensitive, high-temperature dilatometer is being developed to study the annealing kinetics of radiation damage in graphite. It is expected that the kinetics of graphite irradiated at high temperature will give a better insight into the mechanism of radiation-induced contraction. The equipment is presently being tested on a CSF graphite sample that had been irradiated near room temperature to an exposure of 2530 MWD/AT. The radiation-induced expansion transverse to the extrusion axis was 1.14%. Annealing was carried out in 100 C intervals in an argon

atmosphere, and length changes were measured at the annealing temperature. Oxidation of the graphite, if any, appeared to have no effect on the length changes. The results, expressed as the fraction of the expansion annealed in each temperature interval were:

Annealing Temp., C	Fraction of Damage Annealed
300	0.022
400	0.116
500	0.130
600	0.135
700	0.167
800	0.230
900	0.135
	0.935

Application of the Primak[1] approximation for step annealing shows a rate of property change at an activation energy above 90 kcal/g-atom. This value is in the range of the activation energies necessary to cause motion of vacancies in the graphite lattice.[2] A plot of ln $\Delta L/L$ vs ln time for most temperature intervals, interestingly enough, yields nearly straight lines, each with a slope of 2/5, which conforms to the Kingery-Berg model[3] of sintering by diffusion of vacancies through either lattice or grain boundary paths. Although these observations may be fortuitous, they suggest that annealing occurs by migration of vacancies, perhaps to interstitials and crystallite surfaces.

(1) W. Primak, J. Appl. Physics, 31, 1524-1533 (1960).
(2) W. Kingery, Kinetics of High Temperature Processes, pp. 63 (1959).
(3) R.L. Coble, J.E. Burke, Sintering in Ceramics, GE Report No. 62-RL-3124M (1962).

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8. Aluminum Corrosion and Alloy Development

Dynamic Corrosion of Aluminum in a Non-Isothermal Loop. It has been shown (HW-61089) that the solubility of aluminum corrosion product plays a role in determining the corrosion rate. The CMO non-isothermal loop, which operates with an adjustable temperature differential, has the capability for controlling the level of dissolved corrosion product. Water saturated at a high temperature is cooled, precipitating out corrosion product. The water is then recirculated past the test specimens in a state of unsaturation.

Data from the first 10-day run in the loop are available. The loop was operated at 330 C, a linear flow rate of 25 fps, an aluminum area to loop volume ratio of 1.25 in^2/gal and a relatively small ΔT (8 C). In the high flow section of the loop, penetrations on X-8001 aluminum coupons were 1.39 mils in 10 days. By comparison, penetrations were 0.9 mil in an isothermal loop operated at 330 C, 25 fps, with an aluminum surface area to loop volume ratio of 23.4 in^2/gal. Ratios of aluminum area to refreshment rate were 12.5 and 8.3 in^2/gph for the isothermal and non-isothermal loops, respectively.

Loop aging effects on aluminum corrosion rate have been observed by other investigators upon startup of a new loop. The presence of this effect must be investigated and the degree of reproducibility established before significance can be placed on the higher corrosion rate observed in the non-isothermal loop.

The C-1 Loop was moved to the 100-H Shops for modification. All prints were completed.

Aluminum oxide films from coupons exposed in the H-1 Loop have been analyzed by infrared transmission techniques. The films were found to be composed of boehmite and pseudo beohmite.

9. Metallic Fuel Development

Evaluation of Th Alloy Coextrusions. The two 1.75-inch OD x 1.05-inch ID Zr-2 clad coextrusions containing Th-2.5 w/o U (93.2% enriched)-1 w/o Zr core material produced January 18 are being processed for irradiation test element fabrication. Specimens were obtained from the front, middle, and rear of each tube for examination in the extruded and heat treated (1 hour at 750 C vacuum, furnace cool) conditions. Over-all dimensions, cladding thickness, hardness, and

metallographic examinations are summarized below.

Tube Number	Dimension (Mid-section) 129	130
Outside Dia. w/copper	1.785 inch	1.785 inch
Inside Dia. w/copper	1.050 "	1.050 "
Outside Dia. over Zr-2 clad	1.759 "	1.757 "
Inside Dia. over Zr-2 clad	1.058 "	1.058 "
OD Clad thickness	0.0285±0.001	0.0277±0.0006
ID Clad thickness	0.0275±0.0005	0.0280±0.0005

As-extruded hardness varied from R_f 75-78 at the front to R_f 71 at the rear for both extrusions. This is in good agreement with values found on the pilot coextrusions. The decrease in hardness toward the rear represents an increase in the amount of recrystallization. After heat treating for one hour at 750 C, the hardness of both extrusions was reduced to R_f 59-61, also in good agreement with that determined for the pilot material.

Both coextrusions were found to be uniformly metallurgically bonded at the clad-core interface. The bond of tube #130 appeared slightly cleaner than that of tube #129. The billet for #130 was degreased only following machining and #129 was degreased and chemically pickled and a slight surface film was observed following pickling. The core material was only partially recrystallized following coextrusion at 760 C (1400 F) as was observed in the pilot extrusions. Subsequent heat treatment for one hour at 750 C produced complete recrystallization and uniform growth of the clad-core bond to approximately 0.007 inch thickness. The major impurity, ThO_2, was present in approximately the same quantity as in the pilot material. The uranium phase after heat treatment appeared to have slightly larger particle size than was observed previously. An unidentified phase was observed following heat treatment that is of the same size as the ThO_2. It is softer than the ThO_2 but harder than the alloy matrix. Its appearance is similar to that reported for a nickel containing compound as a thorium impurity. Both OD and ID Zr-2 clad appear fine grained and of uniform structure from front to rear after heat treatment. Six fuel element sections were cut from the rear half of each tube for further processing to fuel elements for ETR loop irradiation.

Chemical analysis of the ingots (#129 and #130) from which the enriched thorium $2\frac{1}{2}$ U-1% Sr alloys were coextruded have been received from Oregon Metallurgical Corporation. These are listed in the following table. Duplicate samples have been sent to the Hanford laboratories, the results of which are not yet available.

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Chemical Analysis of Enriched Th-U-Zr Ingots and Extrusions

Ingot Number	U	Zr	C	O	N	H	Cu	Fe	Ni
	Weight %								
129 Top Center	2.42	1.02	0.022	0.18	0.040	0.0006	0.008	0.030	0.032
Top Edge	2.48	*	0.020						
Bottom Center	2.48	1.10	0.022						
Bottom Edge	2.55	*	0.023						
130 Top Center	2.49	1.15	0.022	0.23	0.038	0.0007	0.012	0.029	0.038
Top Edge	2.46	1.07	0.023						
Bottom Center	2.54	*	0.027						
Bottom Edge	2.55	*	0.022						
Extrusion									
129 Front	2.40	*							
Middle	2.38	*							
Rear	2.45	*							
130 Front	2.44	*							
Middle	2.41	*							
Rear	2.43	*							

*Determination to be reported.

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Most encouraging result is the good uranium distribution in the front middle and rear samples of the coextruded tube stock.

Two 30-pound ingots of the above composition using natural uranium have been through the primary melting. Second melt electrodes are currently being prepared. Zirconium clad rod stock will be fabricated from these ingots for autoclave defect testing and filming.

Brazing of Zircaloy to Thorium. Brazing tests are being conducted on the prototypic KSE-3-size thorium-2.5% uranium-1% Zr alloy fuels clad with Zircaloy-2. Optimum time-temperature-pressure cycles have been established for the brazing of this geometry fuel. Since the end-caps extend above the cladding, any excess braze material spills over the side and runs down the outside of the cladding instead of flowing onto the top of the cap. In order to prevent the overflow of the braze, the volume of the recess and the volume of the braze material must be kept uniform. Since chemical milling leaves very non-uniform recesses, it has been necessary to finish the fuels to final depth by close tolerance machining. The machining is followed by degreasing, wire brushing, and cathodic etching. Redeposition occurring during cathodic etching interferes with adequate wetting. For that reason, brazes have been tried on a variety of wire-brushed surfaces to determine if bonding can be accomplished between the thorium and the braze without cathodic cleaning. Metallographic examination has not been completed on these brazes.

Refractory Metal Fabrication. Several pure molybdenum pieces of tubular shape have been successfully fabricated in the 306 Building by powder metallurgy techniques. The tubes, 1.6" OD x 0.84" ID x 2" long, were isostatically compacted from pure molybdenum powder at 30 tsi in re-usable rubber molds in a 3-inch diameter extrusion container. The green density was 85% of theoretical. These compacts were then sintered in hydrogen at 1600 C for 16 hours resulting in a density of 94% of theoretical. The room temperature ductility of the as-sintered material was demonstrated by the cold upset of a sample to more than 50% reduction with only minor edge cracking. Both hot extrusion and hot swaging will be tried to reduce these sintered compacts to thin wall tubing. Sintered molybdenum cylinders have been successfully reduced in area 50% by hot swaging over a mandrel. Mandrels with improved hot strength are now being fabricated.

A 0.75-inch diameter pure molybdenum powder compact was sintered in hydrogen at 1600 C for $7\frac{1}{2}$ hours to 85% of theoretical density, and subsequently heated to 2000 C in vacuum for one hour. This treatment resulted in an increase in density from 85% to 92% of theoretical and indicates that equipment for higher sintering temperatures is desirable.

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

19-Rod Bundle Mixing Studies. The irregular shape of the flow area of a 19-rod bundle fuel element allows an imbalance between heat transfer area and coolant flow area and results in coolant temperature disparities among the various coolant sub-channels. These temperature disparities in turn have a bearing on the boiling burnout heat flux, on the heat transfer coefficients, and on the two phase pressure drop characteristics of the fuel element. The degree of temperature or enthalpy disparity depends, in part, upon the cross channel or interchannel mixing which occurs.

Hand calculations were performed using some data from a 6.3-foot long electrically heated 19-rod bundle fuel element with 0.050-inch spacing between rods as maintained by wire wraps on 12 rods. The data show that, over the 6.3-foot length, the average increase in the enthalpy of the coolant in sub-channels within the bundle is about 1.4 times the enthalpy increase of the coolant in the sub-channels between the bundle and the pressure tube wall. If no interchannel mixing had occurred, the enthalpy increase ratio should be about 3.0. The degree of mixing which would result in the 1.4 enthalpy increase ratio requires a cross flow between the inner and the outer portions of the bundle of 30% to 40% of the total axial flow or about 2.3 to 3.0% of the axial flow for one of the 12 sub-channels between the outer ring of rods and the pressure tube. This compares well with results of tests reported in TDSI-31, "Coolant Mixing in 19-Rod Element Fuel Bundles," by Howieson & McPherson. They reported a cross flow per sub-channel of 1.6% to 2.7% of axial flow for a similar type of rod bundle.

The hand calculations also show that the average heat transfer coefficients in the rod bundle range from about 1500 B/hr-ft^2-F at a flow rate of 500,000 lbs/hr-sq ft to about 6,000 B/hr-ft^2-F at a flow rate of 5,000,000 lbs/hr-sq ft. A computer program is being written to perform these calculations by machine and allow examination of all the available data.

19-Rod Bundle Two-Phase Pressure Drop. The analysis was continued for the two-phase pressure drop data obtained with an electrically heated 19-rod bundle test section, 6.3 feet long with 0.050-inch spacing between rods. It was determined that the two-phase pressure drop for the rod bundle assembly can be correlated quite well by using a pseudo-quality which is defined in terms of the bulk enthalpy at which local boiling begins, rather than in terms of the coolant saturation enthalpy.

Two-Phase Pressure Drop Studies. Initial equipment operation is under way to study the two-phase pressure drop characteristics of various piping configurations. At report time, operation had been confined to obtaining thermocouple calibration data, heat loss data, and some liquid phase pressure drop data.

Dome Seal Type Nozzle Closures. The seal was reinstalled in the test nozzle following the 116-hr steady state test. A total of 50 thermal cycles at constant pressure was then run. The temperature was cycled from 100 F to 570 F while the pressure was maintained at 1500 psi.

During the night and on weekends the dome was exposed to operating conditions of 570 F and 1500 psi. As a result, a total of 170 additional hours of steady state testing was accumulated. The flow of water through the test assembly was about 30 gpm.

The same collection arrangement of a sealed enclosure around the test closure with a water-cooled exit for leakage was used. There was no trace of leakage from the closure at any time during cycling or steady state conditions.

The pre-loading nut on the dome stem was again tightened to 100 ft-lb at the start of the test. The disassembly torque at the end of the test was zero. Force required to remove the seal from the nozzle was 400 lbs. Examination of the seal assembly revealed that each of the six 304 stainless steel retaining lugs had yielded an estimated 0.025 to 0.050 inch. Viewed from the side, strain indications were observed on these lugs at the shear cross section. The center of the dome had been deformed by 0.032-inch causing the dome center to have a slightly larger radius. The seal ring, nozzle seal surface, other bearing surfaces on the seal and nozzle assembly, and other parts were undamaged.

There is an apparent discrepancy between the previously made stress calculations and the presently observed deformation at the dome center. This dome deflection is of concern and will be closely investigated. At this time, however, it is believed that continuation of the tests using the present dome is desirable. The retaining concept for holding the dome and seal ring in the nozzle is of only secondary concern for these tests. It is believed that design of an adequate retaining mechanism can be easily accomplished, and that failure of the present lugs is of concern only as it affects the seal, dome, or seal ring. Replacement with lugs of higher strength is planned prior to the third test phase.

11. Advanced Reactor Concept Studies

Fast Supercritical Pressure Power Reactor. Calculations were completed to revise the 18-group U-238 capture cross sections in the resonance energy range. Self-shielding corrections were obtained from the GAM-I code which utilizes the methods of Adler, et al. As anticipated, substantial reductions in both enrichment and reactivity coefficients were obtained when the revised cross sections were used in core calculations. Material volume fractions were also adjusted to reflect the most recent fuel element design. Results obtained now indicate an enrichment near 20% Pu (13% fissile). An investigation of the initial reactivity coefficients show a zero void coefficient and a total negative flooding effect of -5.7% $\Delta k/k$.

Additional calculations were completed to estimate the effects of fuel burnup. A high initial conversion ratio results from the depleted uranium adjacent to the ZrH_x moderator regions. An initial value of about 1.4 was obtained, not including the axial blanket. This value will decrease with fuel exposure, but present calculations indicate that a net breeding gain should still result. A further consequence of this core design is a rising initial reactivity characteristic. This is due to the relatively high importance of plutonium adjacent to the moderator segments in the core. Problems arising from a shift in the core power distribution with exposure are now being evaluated.

A transport theory calculation was completed to confirm diffusion theory results for this core design concept. The core model used was a two region laminated neutron spectrum core similar to that of the FSPPR. Although a complete comparison has not yet been made, it appears that the diffusion theory results from K_{eff} are high by about 2% $\Delta k/k$.

A triangular fuel element was selected as the most practical configuration to use in the 3-region FSPPR core. The initial triangular fuel element contains a 0.3" thickness of ZrO_2 between the fuel and the fuel element cladding. A gas cooling annulus of 0.3" is left between the fuel elements. Using the thermal SPPR thermodynamic cycle, the resulting heat fluxes and mass flow rates yield FSPPR tube wall temperatures on the order of 1500 F and pressure drops about 1/10 of the values found for SPPR. The tube wall temperature of 1500 F (without hot channel factors) is too high. This will be alleviated by changing the tube size and maximum heat flux to increase pressure drop and decrease the tube wall temperature. In order to improve element design, a selective parametric study is being made of the effects of several design variables. These

variables include coolant mass flow per unit area, core length, and coolant tube diameter. The effect of these variables on heat flux, tube wall temperature, fuel temperature, and pressure drop is being determined. A core design will then be selected using the values of the above parameters which yield an apparent economic design.

Batch fueling of FSPPR appears technically feasible based on (1) the probability of economically achieving about the same average power density (and thus same average burnup rate) in each core region, and (2) the prediction of fairly small reactivity change with burnup. Batch fueling appears desirable since it would minimize shutdown frequency and total shutdown time and permit assembly of the total core-blanket-moderator-flux trap complex out-of-reactor while cold-and-clean. This would permit easier and less costly design of components and ease above-core piping and control system layout.

Plutonium Fuel Spacecraft Reactor. A rough draft of a document describing the original PFSR concept was completed.

Investigations are continuing on the application of the "laminated neutron spectrum core" concept to PFSR cores. It appears that, in a scheme combining a moderating zone with a burnable poison (as boron), variation of the moderating efficiency of the zone may provide an even more potent control mechanism than variation in poison concentration.

D. DIVISION OF RESEARCH - 05 PROGRAM

1. Radiation Effects on Metals

This program is aimed at establishing the combined effect of impurities and neutron irradiation on the properties and structure of specific metals, and to deduce from thermally activated recovery processes how the damage state can be altered. Studies on irradiated single crystal and polycrystalline specimens of molybdenum containing carbon as intentional impurity are continuing.

Single Crystal Molybdenum. Studies of deformation modes in single-crystal molybdenum are continuing. Unirradiated crystals deform by slip in the <111> direction as observed previously. After irradiation the slip appears to be confined to either {110} or {112} planes. In a majority of the irradiated crystals the tensile-axis movements are so small that the determination of poles of the resulting arcs is difficult. In several cases an alternation of slip directions is observed, from one < 111 > direction to another. In

others, one portion of the crystal may slip in one direction while another portion may slip in a second direction. These situations occur when the tensile axis is located equidistant from two possible slip directions. All results can be explained by the above slip systems, namely {110} < 111 > and {112} < 111 >. Future experiments will be designed to more precisely define the deformation mechanism in the region of uniform elongation.

Lattice parameters of irradiated and unirradiated molybdenum single crystals are being determined. A gradient in lattice parameter values has been shown to exist in the electron-beam zone-refined single crystals, the highest value of lattice parameter being observed at the center of the rod and a lower value at the surface. Typical values are 3.1471 A at the surface, 3.1473 A at a point midway between surface and center, and 3.147 A at the rod center. Crystals previously examined are being re-examined to determine if this gradient is present in all samples, and if so, whether there is a dependence on carbon content and irradiation history.

Computation of resolved shear stress-resolved shear strain values for all molybdenum single crystals tested to date has been made on the 7090 computer, utilizing a FORTRAN program. Input to the computer consists of load-elongation data, angular relationships between the crystal axis and slip system components, initial diameter of the crystal specimen, and the effective spring constant of the loading device. Computations of engineering stress and strain, true stress and strain, and resolved shear stress are made according to the normal relationships. Resolved shear strain is computed according to the equation:

$$\gamma = \frac{1}{\cos\phi}\left[\left(\frac{L_i}{L_o}\right) - \sin^2\lambda_o\right]^{\frac{1}{2}} - \frac{\cos\lambda_o}{\cos\phi}$$

where ϕ_o is the initial angle between the crystal axis and the vector normal to the slip plane, λ_o is the initial angle between the crystal axis and the slip direction, L_i is the instantaneous gage length and L_o the original gage length. Elongation values are corrected by subtracting elongation of the loading device (i.e., in the grips, pull rods, and load cell coupling) from the observed elongation. Since only approximately a third of the specimens available have been tested, it is difficult to clearly establish the particular effects of orientation and carbon content. There seems to be an effect, due to carbon, of increasing the slope

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of the Stage III portion (cross-slip region) of the flow curve, a sufficient number of orientations have not been examined for confirmation of this observation. The effect of neutron irradiation, on the other hand, is very well defined. Without regard for crystal orientation or carbon content, irradiation results in (1) the development of an extensive easy glide region, which increases with greater exposure; (2) the development of a definite Stage II (linear hardening) region, the slope of which decreases with increasing exposure; (3) a decrease in slope of the Stage III region; and (4) an increase in the critical resolved shear stress for slip. These results suggest that (1) deformation is restricted in the irradiated crystals to slip on one family of slip planes, such as 101 , rather than a combination of slip on planes of the type 101 , 112 and 123 ; and (2) the first dislocations to move through the lattice in the irradiated crystals do so at a very low velocity and sweep out "channels" which have a relatively low concentration of defects. Subsequent dislocations moving through these channels can then move more easily.

Further testing of molybdenum crystals has been suspended pending completion of development testing on techniques for photoengraving calibrated grids on the gage length of crystal specimens.

Polycrystalline Molybdenum. X-ray diffraction studies on irradiated and unirradiated molybdenum foils representing four controlled carbon levels have been completed. The carbon levels studied were (a) less than 10 ppm, (b) 10-20 ppm, (c) 100-200 ppm, and (d) 400-500 ppm. The lattice parameters of the unirradiated foils differed by less than 0.01% for the four carbon levels. After irradiation to 10^{19} nvt ($E > 1$ Mev), the lattice parameter of the high purity material was unchanged within experimental error. However, the 10-20 ppm C and 100-200 ppm C foils showed a lattice expansion of 0.03% while the 400-500 ppm material showed a 0.04% expansion. Post-irradiation annealing reduced the lattice parameter, with the greater changes occurring in the samples with the higher carbon content. The x-ray line breadths showed parallel results. The effect of carbon content on the amount of irradiation-induced damage is shown very clearly by these results.

Molybdenum foils are currently being utilized in radiation damage and recovery studies. In order to identify the defect structures observed in irradiated foils, attempts have been made to quench in defects in similar non-irradiated molybdenum stock. For quenching, a partial cylindrical tantalum susceptor, large with respect to the volume of the specimen, was spot-welded directly to the foil assembly and this assembly was heated by induction. Quenching occurred after

power shut-off since the tantalum is at a much lower temperature than the molybdenum. Foils of high purity molybdenum and molybdenum containing carbon were then aged in vacuum at 200 and 600 C for two hours. Subsequent thinning revealed no defect structures by transmission electron microscopy. The absence of defects in quenched and aged molybdenum may be explained in two ways: (1) the energy of vacancy formation is so high that only few vacancies form in the material at high temperatures, or (2) a stacking fault produced by the collapse of a disc of vacancies on a (110) plane would be removed by a dislocation reaction of the type

$$\frac{a}{2}[110] + \frac{a}{2}[001] \longrightarrow \frac{a}{2}[111].$$

Another technique for producing defect structures in molybdenum is being investigated. Foils of high purity molybdenum and foils containing 400 ppm C were annealed at 2000 C for 22 minutes and cooled slowly. These foils will be thinned to a thickness of 2000 A, placed adjacent to a source of fission fragments in an evacuated capsule and irradiated in the reactor. After bombardment with fission fragments the foils will be studied by electron microscopy and x-ray diffraction techniques.

In order to supplement the study of radiation damage in molybdenum, parallel investigations have been initiated on metals of other crystal systems. Initial studies will involve nickel, a face-centered cubic metal, and rhenium, a close-packed hexagonal metal. Single crystals of high purity nickel (99.999%) have been obtained and single crystals of high purity rhenium (99.995%) are on order. Polycrystalline nickel and rhenium are also being obtained in the form of rod, foil, and wire.

A number of new items of specialized equipment are in various stages of design or fabrication. Included are: (1) an electrolytic machining device for producing reduced sections on single crystal tensile specimens without increasing the density of dislocations in the crystal; (2) a chamber for quenching electrical resistivity specimens, in which the quenching, measurements of resistivity, and intermediate annealing may be performed without disturbing the specimen; (3) a vacuum furnace to be used for quenching nickel specimens from a temperature of 1450 C; and (4) a torsion pendulum for low frequency internal friction measurements at temperatures from -269 C to 1000 C.

Operational tests are now being performed on the differential calorimeter which has been constructed to measure the energy released from

cold worked and irradiated specimens. Molybdenum specimens are being prepared by a series of upsetting and swaging operations designed to yield specimens from the original polycrystalline rod stock with the required geometry and several levels of cold work.

E. CUSTOMER WORK

1. Radiometallurgy Examinations

The examination has been completed on the four hole failures from D and DR-Reactors. All four failures were caused by groove pitting of the spire. In the enriched elements the pitting had occurred in deep localized grooves, but in the normal uranium elements pitting was general over most of the spire surface. Similar, but less severe, corrosion had also occurred on the outer can walls.

Photomacrographs of cross-sections of three sludge samples from Purex waste storage tanks were photographed in color.

The apparent cause of failure of a hot press element was penetration of the cladding by erosion at the upstream end of two of the supports which lay in an area that is operated at high temperatures.

Examination of three production elements with numbers etched in the cladding by preferential attack was completed this month. Metallography of the cladding revealed no evidence as to the cause of the preferential attack.

Twelve production elements goal irradiated and cooled for five days were defilmed along with four dummies from the same tube, and the solutions were submitted for analysis.

Other work is or will be reported by the respective research and development component served.

2. Equipment Projects

Pulse Annealing Furnace. Final calibration of the pulse annealing furnace was completed. A 50% improvement in sample heating rate was made possible by a modification to the sample holder. Maximum heating rate is now about 90 C per minute at a vacuum of 1×10^{-4} mm of mercury. Moving pictures were completed on a remote operation of the furnace.

Micro Sampling Equipment. Final testing of the micro sampling equipment was started by drilling and collecting samples from a

5600 MWD/T UO_2-PuO_2 fuel mixture containing 4.13% PuO_2. Seven samples were removed and individually collected by drilling 0.020-inch diameter by 0.010-inch deep holes at selected radial positions on the cross-section of a 9/16-inch diameter fuel rod. The drillings from each hole were collected by a vacuum probe on a ½-inch diameter milipore filter disc. The collector and drill were changed for each sample to minimize cross contamination of the samples. Zones of high fission product concentrations produced in-reactor had previously been located by autoradiographs of the sample. This information was used to determine the radial positions for micro sampling. This test suggested several equipment modifications and improvements in handling methods which will be incorporated before the micro sampling equipment is available for routine use as examination tool. These changes should be completed within two weeks.

Over-bore Plug Drill. The first over-bore hole has been successfully drilled in "D" Cell. The original hole was enlarged from 7½ inches in diameter to 9 inches in diameter which will match other large holes in the latest "F" and "G" Cells. The ability to do this kind of work without dismantling the cell not only represents a monetary economy but will greatly facilitate installation of larger equipment in the existing cells.

Manipulators E-1. A light-duty, extended-reach manipulator has been installed in "D" Cell for testing. The manipulator reaches two rows of plugs higher than the Hanford type manipulator, thereby making these plugs available for equipment mounting.

X-Ray Diffraction Equipment. The modifications to the x-ray diffraction equipment are approximately 90% complete. The remaining work consists of electrical rewiring, extending the mechanical controls, and stacking the lead for the biological shielding. The equipment is presently being aligned. The new horizontal x-ray tube had been operated for short periods of time at 50 kv and 35 ma with no apparent over-heating.

Prototype Pneumatic Carrier System E-115. Specifications and vendor information drawings of the pneumatic carrier system have been completed.

2. Metallography Laboratories

Thermocouples embedded in the Zircaloy caps of fuel elements and used to check the temperature of the brazing furnace have given ambiguous temperatures. One of the thermocouples was metallographically examined after two cycles in the furnace. The

examination showed that the tip of the thermocouple well and the thermocouple bead had been dissolved by the brazing material which led to the erroneous temperature readings. Another method was suggested which would allow some loss of thermocouple wire by solution in the braze yet maintain good contact with the Zircaloy cap.

A dispersion hardened nickel sample containing 2% ThO_2 in a matrix of pure nickel was received for ThO_2 particle size determination. Examination with the electron microscope disclosed that a chemically etched surface became pitted, thus making observation and measurement of ThO_2 impossible. A vacuum cathodic etching procedure was developed which exposed the sub-micron size ThO_2 particles without pitting.

A metallographic examination of an N-Reactor carbon steel boiler tube failure disclosed that the failure was caused by corrosion of the exterior surface. No evidence of internal corrosion of the tube was found. Extensive pitting and corrosion of the exterior surface of the tubing below the failure was evident. Pits in the original 0.125-inch thick tube were up to 0.060-inch deep and the residual wall thickness at the area of failure was only 0.013 to 0.015 inch thick. Analysis of the scale taken from the area below the failure showed a high sulphate content which could be indicative of sulphurous and/or sulphuric acid attack of the carbon steel tube.

Other work is or will be reported in connection with the respective research and development programs served.

3. N-Reactor Design Testing

N-Reactor Charging Machine

Modifications. The modified plug conveyor guide rails have been fabricated and installed. The new guide rails are heavier and stronger than the old ones, and in addition, a guide rail is now used on both sides of each conveyor to increase functional reliability.

The new end thrust stabilizer rod anchors have been fabricated and installed. These new anchors bring the line of action of the rods closer to the center of moments of the truck frames which should help prevent rotation of the truck frames.

Minor electrical modifications were made. Installation of the television and communication equipment was started and is estimated 80% complete.

Testing. The nozzle-process-tube assembly which was originally received for use in Design Test 22, Part A, was found to be in unsatisfactory condition and was replaced with an acceptable assembly. All equipment was assembled and inspected, and the test was started. Ten complete charges of fuel elements were made consisting of the following types and quantities of fuel elements:

Test 1 - 18 elements, 24" long with suitcase handle-type supports.
Test 2 - 18 elements, 24" long with arch-type supports.
Test 3 - 18 elements, 24" long with suitcase handle-type supports.
Test 4 - 18 elements, 24" long with arch-type supports.
Test 5 - 36 elements, 12" long with arch-type supports.
Test 6 - 36 elements, 12" long with arch-type supports.
Test 7 - 18 elements, 24" long with suitcase handle-type supports.
Test 8 - 25 elements, 18" long with suitcase handle-type supports.
Test 9 - 18 elements, 24" long with suitcase handle-type supports.
Test 10- 18 elements, 24" long with suitcase handle-type supports.

Tests 9 and 10 used shoes made of softer material than those used previously.

In none of the tests were any supports or shoes stripped from the fuel elements. Some marking of the leading edges of the arch-type supports was observed which apparently resulted from passing over the roll joint transition. Detailed evaluation of fuel support condition after charging was carried out by NRD personnel. Apparently the damage to the arch-type support was such as to eliminate it from further consideration for use at this time. As a result, it is understood that additional charging tests of this type are desired to be run on slightly modified suitcase handle-type supports. The original Design Test 22-A called for a total of 10 charges.

The magazine and nozzle process tube combination were inspected visually after each test and no damage to either was apparent. (Only a 2-foot length of process tube is used in this test assembly).

Design Test 22, Part B, is now being activated and scope design of the test facility has been completed.

Design Test 17, "Noise Level of Components," was run and a report issued.

N-Reactor Fuel Handling Tongs

Machine work on detail parts for the multiple tongs which will lift N-Reactor fuel from the pallet to the magazine loader has been started.

N-Reactor Fuel Magazine Loader

Design Test #23, "Magazine Loader," has been received and approved. Arrangements for installation of this machine in 314 Building are complete. We are awaiting delivery from the fabricator.

Magazine Support

A slightly modified nozzle adapter insert was designed and made. This insert provides proper alignment of the magazine to nozzle.

4. Special Plutonium Fabrications

High Exposure Plutonium-Aluminum Fuel for Physics Tests. Two hundred, 3-foot long, Al-Pu rods were fabricated for physics tests in the PCTR and PRCF. This completes 650 of the 1000 rods needed. Billets were cast for another 250 rods.

High Exposure PuO_2-UO_2 Fuel for PRTR Physics Tests. Sufficient UO_2-PuO_2 pellets were sintered to fill 10 of the 18 rods needed for six PRTR flux monitor fuel clusters. Three rods were loaded.

Low Exposure PuO_2-UO_2 Fuel for PRTR Physics Tests. This work was postponed to permit completion of the flux monitor pellets described above.

High Exposure PuO_2-UO_2 Fuel for PCTR. Impacted PuO_2-UO_2 will replace pellets in these rods.

High Exposure Plutonium-Aluminum Fuel for Corrosion Tests. Eight of the forty 5-inch long Al-Pu fuel rods required for corrosion tests in the C-1 Loop were fabricated. These were the first plutonium-bearing elements to be fabricated by cutting long coextruded rods into pieces, counterboring, and completing end closures without plutonium contamination in the end cap weld.

EBWR Plutonium Fuel Elements. The accelerated schedule set in December for this work was maintained. Orders for Zircaloy tubing and arc fused uranium dioxide were placed. Initial deliveries are expected in April, well ahead of schedule. Eight of the 60 kilograms of Pu were burned to PuO_2. Detailed design of the fuel rods was established, with the exception of length. Extra length will be required for a fission gas reservoir if a pending program decision is made to increase goal exposure to 50,000 MWD/T.

A mockup of the vibrational compaction system was started in the 325 Building to test some of the design concepts. Initial tests to determine the optimum blend of powder sizes for the EBWR elements indicated the blend used for PRTR rods is satisfactory, despite the smaller diameter of EBWR rods (3/8 inch versus ½ inch).

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PHYSICS AND INSTRUMENTS LABORATORY

MONTHLY REPORT

FEBRUARY 1963

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

N-Reactor Experiments

The manuscript of the NPR lattice parameters document (HW-72696) is essentially complete. Work has recently been directed towards a complete review of all measurements in order to obtain the most consistent set of lattice parameters. It appears that the measurements of resonance escape probability may now be made consistent with the other parameters.

Vertical traverse data were obtained in the exponential mockup with water-cooled enriched fuel. These data will decrease the experimental uncertainty in the material buckling.

Optimization of Re-tubed Lattices

Horizontal traverses have been measured in the "C" exponential pile with CVIN (overbore) fuel. The measurements were taken with the lattice unflooded and also flooded.

Lattice Parameter Measurements on Large Diameter Fuels

An article entitled "Lattice Parameter Measurements for Solid and Concentric Tube Uranium Fuel in Graphite Lattices" has been completed for submission to Nuclear Science and Engineering.

Computational Programming Services

Programs KERNEL and SUMMIT were submitted to the Special Procedures Library. The following were added to the Fortran Library: The improved version of WTBX/RTBX; ORTHO2, the subroutine version of the polynomial fitting code ORTHO; GORGO, the subroutine which uses ORTHO2 and computes ordinary polynomial coefficients; and a main program which utilizes GORGO.

A deficiency in the manner in which SUMMIT handles computed kernels was demonstrated when a partially completed run was interrupted and removed after an hour's computing time. Although the kernels which were computed

were available as printed output, they were not available as punched output. The code has now been changed to punch kernels when they are calculated. In the future, if an interruption occurs, the calculations can be resumed with nominal repetition. As the kernels which are punched in these interrupted cases are not in the proper form for further processing, an auxiliary code SUMDUM has been written to read them and punch them in proper form. For the interrupted run above, a rerun was avoided by keypunching the kernels from the printed output.

The calculation of self-shielded cross sections for natural uranium rod and tube-in-tube lattices has pointed up a deficiency in the unresolved fission cross section of U-235 as computed by GROUSS. Although the effect on the self-shielding factors is small since there is little U-235 self-shielding in the unresolved range, experiment correlation has been delayed while the unresolved resonance parameters are being examined.

Instrumentation

Tests evaluating portions of the gamma energy spectrometers being used in the NPR Fuel Rupture Monitoring System were observed at GE-APED. Detail planning for further evaluation of the prototype unit under Design Test 1133 has been started.

Procurement of NPR Fuel Rupture Monitoring equipment for the PRTR test loop is progressing. A multichannel analyzer, typewriter, paper tape punch and reader, and an oscilloscope have been received. Bids were reviewed and an order placed for the scintillation detectors. Specifications for the flow controller and transmitter system were revised to eliminate possible safety hazards associated with the use of glass in contaminated pressure systems.

Instrumentation requirements for N-Reactor startup were formulated with NRD and IPD personnel. Instrumentation tests were started to provide basic data for selection of new counters, preamplifiers, and amplifiers. Preliminary startup instrumentation cost estimates were made on the basis of experiences at the older reactors.

Further calculations were made regarding the use of static laboratory data to evaluate the inherent error in reactor period instrument indications due to non-logarithmic amplifier response.

An invention disclosure was filed on a novel method of combining reactor flux level and period measurements into a single parameter which is expected to be useful in reactor startup operations. Feasibility was established mathematically and the concept is being further investigated.

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System Studies

A multipoint recorder was calibrated, tested, and installed at 100-D reactor to measure reactor outlet temperature changes as a function of rod motions. Of eight thermocouples planned for the experiment, only two have been connected to date. A production test authorizing small, planned rod movements for the purpose of obtaining reactor coupling coefficients has been submitted.

The simulation diagram for a 4-node, two-dimensional representation of 100-D reactor is complete except for the coupling coefficients which are being obtained from these rod position-outlet temperature measurements. Further simulation testing to be done in connection with automatic reactor control studies is planned for mid-March.

Computer runs of the NPR pressurizer-injection system were demonstrated to members of the AEC and NRD to familiarize them with the system and to demonstrate the application of computing techniques to the study of dynamic systems. Although the results of this particular program will be primarily utilized by Reactor Design Analysis Operation, NRD, the information gained will also be useful in establishing proper input functions for the more detailed simulation of the pressurizer system now in preparation. Studies of this system indicate that more detailed simulations may provide better evaluations of abnormal water pressure and level transients.

Several of the more complicated NPR gas composition functions were successfully simulated on the GEDA computer. Eleven functions are ready for setup on the machines as soon as they become available. A detailed analog simulation diagram has been completed, revised, and patched onto the EASE and GEDA patchboards. Preliminary run criteria for investigations have been submitted by Reactor and Fuels Research group.

Use of a miniature analog magnetic tape recorder and an associated analog-to-digital converter with tape punch was suggested to Research and Engineering, IPD, as a means of handling reactor process tube thickness data. The scheme requires extra portable equipment but eliminates the need for costly tape recorders.

SEPARATIONS

Experiments With Plutonium Solutions

Criticality experiments were continued with concentrated plutonium nitrate solutions in the 11.5-inch diameter stainless steel sphere (measured volume = 12.95 liters, wall thickness - 0.049-inches). This vessel was

especially designed with an off center mounting of the control and safety rods so that a re-entrant tube could be inserted along the axial diameter of the unit to facilitate flux measurements. Flux measurements will provide a check on the computed spatial distribution of neutrons in these plutonium solution systems. The first foil exposures for neutron flux measurements were made in February; a series of cadmium covered $\frac{1}{4}$-inch diameter gold foils, and copper foils were activated in the re-entrant tube during a 30-minute exposure in the critical assembly.

These initial experiments are being carried out to check experimental procedures and find the most applicable foil materials available for spatial distribution and spectral index measurements. Lutetium as well as gold and copper foils are being irradiated in the re-entrant tube to provide information on the power level and exposure times necessary for suitable foil activations. The concentration of Pu in the solution being used in the critical assembly is about 220 g Pu/ℓ at an acid molarity of 1.2. Due to analytical problems, the final results of the chemical analyses for the solutions have not been received.

The system was critical with 12.32 liters without the re-entrant tube, and with 12.45 liters with re-entrant tube inserted.

Experiments With Plutonium Oxide-Plastic Mixtures

Work is continuing toward the completion and operation of the Remote Split-Table Machine. The RSTM is now being readied for installation within the second hood of the critical assembly room.

The Critical Mass Laboratory, "Nuclear Safety, Operational Limitations, Instructions, and Emergency Procedures Manual," has been revised to include the experiments with the PuO_2-plastic fuels.

The plutonium-oxide-polystyrene fuel blocks have been transported to the Critical Mass Laboratory and are stored in a critically safe array in the assembly room.

Theoretical Analysis of Plutonium Solution Criticality in Water Reflected 14-Inch and 11.5-Inch Spheres

In connection with the correlation of critical mass experiments with theory, further effort was made to compute criticality for some of the solutions as measured in the laboratory. As reported in the previous monthly report, HW-76315 B, the calculated results are all within three percent of the measured critical volumes with the exception of Experiment No. 1142113 where a difference of $\sim$ 10% in critical volume is observed.

In this particular experiment the solution contained (water reflected 14-inch sphere) 44.5 g Pu/ℓ (4.6% Pu^{240}) with 386 g NO_3/ℓ. The calculations were repeated for the experiment to eliminate possible errors in the original calculation; the results of the second calculation being identical to that for the original calculation.

Another experiment involved a slightly higher plutonium concentration (47.5 g Pu/ℓ with 462 g NO_3/ℓ). The calculated results for this experiment were within three percent of the experimental critical volume. A plot of the water reflected 14-inch sphere experimental data shows that the results obtained for Experiment No. 1142113 are not quite consistent with the other data, indicating a possible difficulty in the experimental or chemical analysis.

A theoretical extrapolation was also made to determine the concentration of plutonium in water that would be critical in the 14-inch water reflected sphere, i.e., zero nitrate. The resulting critical concentration is 28.3 ± .7 g/ℓ (Pu^{240} = 4.6%).

Scattering Kernels for Polystyrene

In order to analyze critical experiments with PuO_2-polystyrene compacts, the scattering kernels for polystyrene need to be known. Through the use of Raman and infrared spectra for polystyrene, the scattering kernels were generated by adapting the water kernel code KERNEL to this use (this procedure has been used successfully for polyethylene). The accuracy of these figures is not known since there are no known polystyrene scattering cross section data available; however, spectral sensitive foil activations planned for the PuO_2-polystyrene compacts will provide some measure as to their accuracy.

Buckling of Partially Filled Spheres

The solution of buckling and flux distribution for a sphere 90% filled was obtained with the partial difference code. An attempt to get more memory space by eliminating one-third of the memory used in the Fortran library CROUT subroutine for solution of simultaneous equations has led to an elusive coding error.

Subcritical Interactions

Stability of SOLAN was improved by including a limiting function for the probability of neutron transfer between cylindrical vessels which nearly touch. The previously used numerical function, while adequate at larger separations, diverged as the separation vanished.

Some simple interaction functions were tested against the more exact SOLAN expressions. For the calculation of interactions between vessels of unequal size, it was found that errors could be reduced $\sim 5\%$ by using the central point solid angle for interaction from the smaller to the larger vessel, and averaging the 1/4 and 3/4 point solid angles for interaction from the larger to the smaller vessel.

It was found possible to eliminate the need for iteration on k_{eff} and surface albedo by multiplying the interaction probabilities by the ratio of radial to total buckling. Also, a better approximation for shadowing of neutrons by a third component spaced between the two interacting units was found and is being applied to the 92 element design problem for Critical Mass.

INTERSET in final form will use coordinate and height data for each element to set up the interaction matrix and solve for the eigenvalue albedo on an infinite cylinder of like characteristics. This albedo may then be applied as a boundary condition in a code such as HFN to calculate k_{eff}.

Instrumentation

Two Keithley Model 711 millivot discriminators were used in a new high-low trip controller for the reactor safety circuit at the Critical Mass Laboratory. The supposedly isolated input of the 711 discriminators proved to be quite sensitive to a-c pickup. Additional work will be necessary before the controller can be used.

Consulting Services on Nuclear Safety - Criticality Hazards

Nuclear Safety in CPD

A nuclear safety specification proposed by 234-5 Building Technology for storing plutonium bearing waste materials in large arrays was reviewed and comments submitted to CPD. The waste materials consist of plastics, cardboard, rags, rubber gloves, metal tools and pipe, glass, etc. The important characteristic of these waste materials is that hydrogen content per unit volume is very low, so the plutonium is undermoderated.

The waste is to be stored in boxes which are 18 x 18 x 24½-in. high. Nuclear safety will be assured by:

a) Limiting the amount of Pu per box to 400 g
b) Limiting the amount of hydrogeneous material per box to the equivalent of 35 lb. of water, and
c) Limiting the plutonium concentration in each box to less than 10 g/ℓ.

The stacking of boxes two high is permitted if the boxes contain 200 g or less of plutonium.

Participation on the Recuplex Deactivation Committee continued. Most of the effort this month was directed towards reviewing the hazards of reactivating Hood 42, which is used for dissolving PuO_2. Hood 42 is located in Room 221 and has two or three piping connections to Recuplex vessels, so has been shut down since the K-9 incident in April 1962.

Activation of the hood will permit the recovery of plutonium from recycle skulls and reject buttons, which will have a significant effect on plutonium production. It is planned to isolate Hood 42 from Recuplex by disconnecting and blanking off the pipe connections and then place the hood in operating condition.

Nuclear Safety - Training and Education

The fourth and fifth sessions of a course in nuclear safety for the benefit of CPD engineers and other interested HAPO personnel were begun at the Critical Mass Laboratory. The course consists of a series of about 15 lectures, together with the showing of several films, on the subject of criticality and nuclear safety. There are a total of 50 persons currently enrolled in the two sessions, with personnel from both CPD and HLO, as well as five members of the AEC-RLOO and one person from Euratom attending lectures.

NEUTRON CROSS SECTION PROGRAM

Scattering Law Measurements for Light Water at Elevated Temperatures

Measurements preliminary to the measurement of inelastic slow-neutron scattering from water at elevated temperatures were terminated during the month due to failure of the BF_3 neutron-detection system. A He^3 proportional counter was obtained as a replacement detector. Necessary modifications to the detector shielding assembly for this counter are almost completed.

The results of the computer program DOUBRAGG have been used to choose sets of values of initial and final neutron energies for inelastic scattering measurements using the crystal spectrometer. These energies are chosen to avoid complicating reflections which occur in the crystal monochromators.

Fast-Neutron Cross Sections

Measurements were made of the total cross section of Pm^{147} for neutrons of 2.5 to 15-Mev energy. The small amount of Pm which was available required

the use of a sample only 3/8 inch in diameter. Measurements were also made on a Pb sample of the same diameter to test the validity of the measurement. Measurements were also made to determine that the radioactivity of the sample did not cause appreciable systematic error in the cross-section measurement. The data from these measurements have not been processed as yet.

A new 5-inch diameter liquid scintillation detector was used during the month along with a new transistorized pulse-shape discrimination circuit. All components operated satisfactorily.

Rotating-Crystal Spectrometer

Development work was continued on the problem of phasing two rotating rotors for application to the measurement of slow-neutron inelastic scattering by time-of-flight. An electronic power amplifier had insufficient power to drive two rotors simultaneously. Phase stabilities of the order of 0.25 degree were obtained with 60 cps line voltage and 400 cps power from a motor-generator set.

Instrumentation

Equipment for the 6144-channel time-of-flight analyzer has been ordered. The additional logic modules for the system are expected to arrive within a week and debugging will then be possible. Wiring is about 90% complete.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE

High Exposure PuAl Lattice Studies

The measurements in the PCTR on the three cell by three cell array of 19 rod clusters on a $10\frac{1}{2}$ inch pitch have been completed for the poisoned case. The foil activation data are now being analyzed with the aid of the APDAC computer program. For the unpoisoned case, the cadmium gold ratios in the central cell and surrounding buffer cells have been matched. Foil activation traverses have been made using both bare and cadmium covered lutetium foils.

Approach to Critical Measurements in H_2O for PuAl Fuel

Experiments similar to the ones performed on the 1.8 w/o PuAl fuel rods are planned for 2.0 w/o PuAl with 16% Pu-240 content. Five hundred and fifty of the 1,000 fuel rods expected have been received. The set of templates with 1027 holes spaced 0.66" center to center was returned to the shops for

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additional holes to allow 1,000 fuel rods to be loaded in a cylindrical pattern.

The Critical Facility of the PRP

a) PRCF Startup

Twenty-five UO_2 fuel clusters have been loaded into the PRCF in four steps of 7, 6, 6, and 6 clusters each. For each loading multiplication data were obtained for moderator heights of 60, 70, 85 and 105 inches. The inverse multiplication for 25 UO_2 fuel clusters in the central fuel channels and a moderator height of 105 inches is between 0.33 and 0.665. This large spread in values is due in part to the asymmetric neutron source in the moderator resulting from the irradiated fuel stored in the reactor cell. The spread is also due to counter placement and the influence of fuel loadings.

The effect of the irradiated fuel on the horizontal flux in the reactor vessel has been observed for each fuel loading. The extreme asymmetry observed with no fuel in the vessel decreased rapidly with each addition of fuel.

Work on startup was suspended February 13, 1963, pending receipt of authorization to load PuAl fuel clusters. Special authorization is required because of the confinement problem of the cell gas.

b) Irradiated Fuel Measurements

The 1/4" long Co-Al monitors from the first PuAl fuel element which was irradiated in the PRTR as part of a burnup experiment have been received. Nine of the fourteen monitors have been removed from their Zircaloy jackets. Eight were removed with less than a 1/32" loss in length. The ninth monitor stuck to the jacket and attempts to free it have not been successful. Attempts of sawing and filing the jacket to free the monitor have resulted in the loss of about 40% of the monitor.

Equipment to count the monitors has been assembled and tested. The detectors being used in the tests are a BF_3 ion chamber which is normally used as a neutron counter and an air ion chamber from a Cutie-Pie (CP) radiation instrument. The signal from the detector is amplified by a DC amplifier and then recorded automatically by a digital voltmeter and typewriter unit. By recording the data automatically errors caused by operator judgment in reading a meter are removed. The stability of the two types of detectors was the same, but the CP chamber is $\sim$ 50% more efficient than the BF_3 chamber.

c) Plans for Future Work

The final draft of Supplement 1 to the final safeguards analysis has been reviewed. The information in the supplement will be submitted to the Atomic Energy Commission for approval to conduct experiments in the PRCF with D_2O moderator using a variety of core loadings and irradiated fuel elements.

The design modifications of the PRCF needed to do H_2O experiments have been established. At present the modifications consist of the following:

1) Fabrication of three safety-rod, control-rod clusters
2) Construction of three safety sheets
3) Construction of an external enclosure.

The safety rod, control rod clusters are to have fuel element followers so that the safety rods will be replaced by fuel when they are withdrawn. Because of the small lattice spacing involved for some of the experiments the plans are to use four rods covered with cadmium as a single cluster. In some experiments which require finer control, one of the shutter-type rods used now could be put into the reflector as a regulating rod.

The safety sheets would be used in the reflector and would have followers made of a material having nuclear properties similar to H_2O. An important problem which will have to be solved is that the sheets will have to be placed at different positions for different sized cores in order to be effective.

The enclosure will be four feet in height and will be used to support the safety rod and sheet mechanisms and the lattice templates. The enclosure will also help to keep foreign material out of the H_2O. The enclosure and attached templates will be fabricated such that they can be assembled outside of the tank and inserted as a unit into the tank and onto the existing outer template. It will be possible to put insulation around the tank for temperature coefficient measurements.

The ratio of the delayed neutron fraction to the prompt neutron lifetime β/ℓ is a basic quantity in the equations that are used to determine the amount of reactivity that can be added to a critical reactor before control by delayed neutrons is lost. This ratio varies with different fuel loadings and can vary considerably when mixtures of two fissionable materials (U-235 and Pu-239) are used as fuel. Thus each fuel loading will require a knowledge of β/ℓ, either by measurement or by cal-

culations or both. Three methods have been used to measure this ratio: from the neutron decay after a rapid insertion of poison, from the analysis of the statistical fluctuations of the neutron flux while the reactor power level is held constant, and from the damping of forced power oscillations as the oscillations are swept through the appropriate frequencies. Experiments utilizing the first two methods are being planned for the PRCF startup. The third method may be available later. The values of β/ℓ derived from the first two methods can be compared. The times needed to analyze the data will also be compared. Specifications have been written for additional equipment that will collect the data in such a manner that it can be processed by the IBM computer with a minimum of handling.

Spectrum Measurements

Several types of nickel for use as foils have been received for evaluation purposes. The nickel foils will be used as thermal neutron or resonance neutron detectors (n,γ reaction) and as a fast neutron detector (n,p reaction).

The use of the (n,γ) reaction is complicated by the production of Mn-56 from a small amount of manganese present as an impurity. The Mn-56 is produced in appreciable amounts compared to the Ni-65 from only a 1% impurity. The problem is complicated further by the fact that the half lives of Mn-56 and Ni-65 differ by only 0.02 hrs. (2.58 h for Mn-56 and 2.56 h for Ni-65).

The use of the (n,p) reaction is simpler and an irradiation in the TTR as a function of position has been successfully completed. Relative activities were determined by counting the gamma rays emitted between 400 and 900 kev. The differential counting was done in order to reduce the counting room background by a factor of 10. This is important because the small cross section ($\sim$400 mb) and low neutron population (neutrons with energy > than $\sim$3 Mev) result in a small reaction rate.

Transport Analysis of Neutron Rethermalization

A transport theory study of the neutron flux in the PCTR with a large diameter copper bar in the center has been completed. Cross sections for 21 energy groups were obtained from the programs GAM, SUMMIT, and SPECTRE. Upscatter was included for the 16 groups below 0.68 ev. The neutron distribution in space, energy, and angle was calculated with HAPO Program S. Sensor cross sections were included to calculate copper activity and cadmium ratio and the effective neutron temperature as a function of space and angle. The calculated copper activity matched the measured values within

1% in the graphite. The calculated activity was about 10% low at the center of the copper bar. This discrepancy is not surprising since the effective resonance integral for copper in copper is not known. Furthermore a small uncertainty in the cross section can cause a large error in the flux because the flux drops a factor of 25 in the copper bar. Typical calculated neutron temperatures were 45°C at the peak of the flux (35 cm from the center of the reactor), 121°C at the surface of the bar and 180°C at a position 0.318 cm. from the surface. The neutron temperature as a function of angle at the surface of the copper was 86°C straight in, 116°C parallel to the surface, and ~207°C straight out. Typical calculated cadmium ratios for copper (30-mil pins) were 70.3 at the peak of the flux and 59.5 at the surface of the bar. The experimental values were 78 and 58, respectively.

These results indicate that any measurement of angular distribution near a "gray" absorber must include at least two detectors with different cross section shapes in order to separate the energy and angular dependence of the flux and activation rates.

Theoretical Scattering Kernels for Water

Program EWGUS which calculates scattering kernels for water with first order corrections for finite energy widths is now operating. Preliminary comparison with experiment indicates that the hindered rotational level in water is too broad to be adequately handled by a first order width correction. Consequently, work has been started to produce the second order width correction as well as the first order correction. Inclusion of the second order correction adds only slightly to the complexity of the theory.

Fuel Cycle Analysis for Programming Operation

A document is in preparation summarizing the results of some 100,000 fuel cost calculations. The study optimizes the fuel cost on several parameters, including, enrichment (U-235 and low, medium, and high exposure Pu), fuel clad (stainless steel, Hastelloy, and zirconium) and lattice spacing. Both batch and graded cycles are examined.

The study indicates that the value of plutonium as an alternative fuel is less dependent on isotopic composition and fueling strategy when lattice spacing and enrichment are simultaneously optimized than when they are not.

EBWR Program Planning

ALTHAEA code is being used to compute one-dimensional reactor statics and fuel burnup in the EBWR. Some useful burnup results have been obtained in essentially zero-dimensional analyses. The results from the burnup studies

will be used in planning the EBWR experimental program.

The isotopic composition during burnup of a PuO_2-UO_2 EBWR fuel rod was computed. In this case the rod initially contains 2.5 w/o PuO_2 (with 8% Pu-240), and resides in a position at which the average H_2O coolant void fraction is 0.30. Since this position is an interior one in a large region of like material, the burnup results should correspond nearly to those obtained in a zero-dimensional analysis. The computed reactivity history does not, however, agree with the results obtained with the CYCLE II code at Argonne National Laboratory.

Phoenix Fuels for Compact, Water Moderated Reactors

Physics statics calculations for a variety of plutonium-fueled, compact, water moderated cores are continuing. Burnout calculations of the so-called Mark-I core (Zr-H_2O composition, M/W ratio equals unity) are underway. The initial results indicate that the Pu-240 content will probably have to be higher than had originally been believed. The presence of some Pu-241 in the initial plutonium composition may also turn out to be helpful.

Beginning-of-life calculations for various Mark-II cores (SS-H_2O composition, M/W = .25) have been completed and burnout calculations for selected Mark-II cores are also being started.

Code Development

CALX

Initialization subroutine SETZ has been debugged. All available options have been run successfully. Economics tape writing subroutine ECOUT has been coded and compiled. This complete rewriting was made necessary by an extensive revision of the QUICK input format. Some logical errors in the cross section subroutine SIGMA, possibly permitting a false average flux convergence, were discovered and are being corrected.

CALX data tape preparation code SIGMA-3C was modified by adding subroutine ECON which correlates information needed later by subroutine ECOUT in CALX.

GAM

The GAM subroutine INTERP has been altered to obtain more accurate values of the integral $J(k, \xi)$ for $1.0 < \xi < \infty$. This integral is used in calculating resonance integrals. To extend the tabulated ranges, an auxiliary table has been inserted in one copy of the GAM library tape. The auxiliary

table, composed of coefficients of a polynomial in $1/\xi^2$ fitted to appropriate existing tabulated values of ξ at each tabular entry of k, was generated by a new program, VOLTX. Both VOLTX and the revised INTERP are apparently debugged.

RBU Cross Sections

Some errors and inconsistencies have been detected in the RBU Basic Library. An error in Pu-242 cross section has been corrected. The Pu-241 fission cross section initially entered on the tape underpredicts the measured value of the infinitely dilute fission resonance integral by a factor ~ 2. Since the resolved resonances contribute only about half of the value, a recipe must be used to match the experimental value.

Contrary to the accepted opinion that the cross sections for U-235 are acceptably resolved, an evaluation of the data indicates that regardless of which measured values are used (i.e., experimentally determined resonance parameters or values of η one has to fudge to make experimental fission and capture integrals match the experiments. Program "GROUSS" is being utilized to determine which parameters best match experiment.

RBU

The major modifications to the Monte Carlo portion of RBU are completed and inserted. A number of questions concerning cross section averaging and subsequent use of the average values in the diffusion and burnup calculations need resolution prior to a problem test.

PRTR Burnup Experiments

Analysis of the test fuel element 5075 is essentially complete and an abstract was submitted for presentation at the ANS Meeting in June, 1963.

The first of the UO_2 instrumented fuel elements has reached a goal exposure of 46 megawatt days and is now cooling in preparation for detailed analysis.

Mass Spectrometry

The heavy-element mass spectrometer was used to provide isotopic analyses in support of Plutonium Recycle Program studies. Analyses were provided on seven samples from Pu-Al-Ni fuel element No. 5051 and on three samples from No. 5095.

Instrumentation and Systems Studies

Detailed instrumentation information assistance was provided in support of the FRPP project. Specifications were prepared for ion chambers, electrometers, power supplies, recorders and radiation area monitors. Information was also provided for an air filter counter for Pu-239. Loan of a commercial picoammeter was made from a vendor for testing and evaluation.

The detailed cost estimate and a portion of the drawings were completed for the three solid state scintillation liquid effluent monitors for use at PRTR. Final model printed circuit boards were completed and tested in the experimental prototype. The experimental prototype will be revised to conform to final design for use as a spare or reference unit. The three final model instruments will be fabricated as direct copies of the revised prototype.

Specific drawings are nearing completion for the underwater gamma scanning system for use with PRTR fuel elements. A general cost estimate was started regarding the system and requirements were established for the necessary electronic instrumentation.

Scoping plans were made regarding developmental work with Helium-3 neutron counters; one counter was obtained on loan from IPD.

Two new neutron-sensitive ionization chambers to be used in a forthcoming PRTR noise analysis test were tested in the Thermal Test Reactor during the month. Analysis of the data yielded a zero power transfer function which agreed with a previous test within the limits of experimental error. Both chambers were used, separately and in parallel, with good results.

HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

Rough drafts of the Project Proposal for design funds and a Summary of the Design Criteria have been completed.

NEUTRON FLUX MONITORS

Samples necessary for the determination of the neutron spectral parameter r, neutron temperature T, and neutron flux ø have been completed and are ready for irradiation in KE reactor. The actual irradiation tests have been delayed pending production test approval.

Individual weights of a new set of cobalt samples have been accurately determined. Samples have been segregated into matched pairs in order to

improve the reliability of experimental data from which the value of r will be determined.

Components required for experimental evaluation of the Borcn-11 beta current neutron flux monitor concept have been ordered. Experimental apparatus have been designed and a requistion for off-site fabrication prepared.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

Initial evaluation tests of the prototype multiparameter eddy current tester confirm its ability to separate three test parameters over a small range of parameter variations. Evaluation tests are being continued to determine detailed performance characteristics. The behavior of several inductively coupled eddy current diffusion models were studied to clarify the nature of the diffusion of pulsed currents. Techniques are being established for fabricating specific types of irregularities in cast Woods metal test specimens for use in a study of eddy current signals due to various types of irregularities.

The first prototype version of the multi-parameter eddy current tester was completed. The wiring was checked and several connections were changed in order to achieve proper operation. The tester is now undergoing evaluation tests. These tests have indicated reasonably good stability, but better stability may be required for certain high sensitivity applications. Two-layered test specimens of brass on steel have been used. The tester has been able to read probe spacing out on one meter, brass thickness on a second meter, and steel thickness on a third meter. The separation of these three parameters has been very good over a limited range. As expected, if the change of parameters is too great, the separation of these parameters becomes poor. However, parameter separation was good and the readout was essentially linear for probe spacing of zero to 11.5×10^{-3} inches, for brass thickness of 1×10^{-3} to 4×10^{-3} inches, and for steel thickness of 1×10^{-3} to 5×10^{-3} inches.

Work on the eddy current diffusion tests is continuing. The analog computer has verified the similarity of currents in a five-mesh inductively coupled circuit to measured values of eddy currents at different depths in a metal. Other simplified models were studied by synthesizing these models on the computer. Also, the effect of progressively coupling more meshes in the model was studied.

The use of cast Woods metal eddy current test specimens having embedded irregularities made of plastic, fiber board or metal is being explored. Both solid cylindrical and tubular test specimens have been made, using

glass test tubes for forms. Phenolic inserts are used to position the center test tube and to hold the irregularities in place during the casting operation. Using this technique cracks or voids within the tubing wall may be simulated. Uniformity of the resulting casting needs to be improved as some porosity is observed in the finished specimens.

An eddy current tubing tester has been modified for use as a uranium oxide thickness tester for use in indicating the thickness of oxide in the range of 0 to 3 mils on the internal surface of extruded uranium cores.

Heat Transfer Testing

Tests were made on end closures of four N-fuel elements. Variations in infrared signal during the initial tests were caused by surface irregularities. Test pieces containing known defects are being fabricated for use in determining the optimum scanning speed and heating power for heat transfer testing N-fuel end closures.

Zirconium Hydride Detection

Thirty-six typical samples of N-reactor process tubes have been obtained from the N-Reactor Department for use in evaluating nondestructive zirconium hydride methods presently under development. Hydrided regions typical of those expected in the field are to be produced in the samples when initial tests are complete. A device for displaying signals on the complex voltage plane was developed and incorporated in the laboratory eddy current hydride detection equipment. Improvements were made in the probe coil and associated circuitry.

Tests on hydrided tubing samples using the complex voltage plane signals display confirmed earlier conclusions that changes in eddy current signals, which are detected when approaching hydrided areas, are due to factors other than probe-to-surface spacing changes. This method of signal display should expedite interpretation of signals obtained when testing process tube samples.

N-reactor process tubes will normally have 18, 30, or 35 percent residual cold work. In addition, tubes containing different levels of cold work have had different heat treatment histories. Samples of each of the three types of tubes have been obtained, and tests are being made to determine the effect of tube history on the eddy current signals.

Fundamental Ultrasonic Studies

Preparations for experimental examination of attenuation and boundary wave effects for liquid-liquid boundaries continued. The acoustic and chemical properties of pairs of liquids were compared to determine suitable conditions for experimentally checking the analytical results. Kerosene and glycerine are believed to be sufficiently immiscible to form a good liquid-liquid boundary which will have desirable acoustic qualities. Since both of these liquids have appreciable attenuation, the analytic studies were extended to include the influence of both attenuation coefficients. The fabrication of equipment for experimental studies is about 90% complete. A previously initiated digital computer program for refining the solutions of the Lamb wave frequency equation was completed. The program provides continuous tabulated phase velocity and frequency-thickness data for the first twelve Lamb modes.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

The first phase of the program has been oriented toward the development of ultrasonic test procedures and written specifications for the inspection of fuel sheath tubing with wall thickness in the range of 0.030 inches. This work is now essentially complete. Second phase efforts are now in progress with the emphasis shifted to 0.017 mill wall tubing. As in the earlier work, objectives are to develop simple, reliable instrumentation capable of production line operation in tube manufacturing plants, and to develop complete supporting test specifications and unified test procedures. In addition, the theory of wave propagation in tubular geometries will be extended to thinner sections to provide firm technical basis for test applications.

Experimental work has begun with the use of exceptionally wide frequency band equipment. A marked improvement in defect resolving power has been obtained with very short-time transmitted pulses with rise times shortened by special electronic techniques. Fed to transducers as small as one-eighth inch diameter, these signals have successfully resolved front and back surface echoes of the thinner-walled tubing in the case of longitudinal waves propagating normally to the tube surface. In this instance, the echoes were separated by only 0.2 microseconds in time. In practice, these small crystals are operated with water path lengths in the order of one-quarter inch.

Favorable results were obtained as this system was applied to the detection of bonafide and artificial defects. With the crystal oriented to generate

what are believed to be shear waves traveling in a circumferential direction, notch standards 1/20 and 1/10 of the 0.017 inch wall were clearly detected on either the inner or outer tube surface. Interesting, and as yet not completely understood, echo patterns were observed as the tubes were rotated in the ultrasonic beam. As a defect is rotated away from the beam impact point, for example, the signal is observed to go through regions of maximum and minimum amplitude. Echoes appearing later in time, however, appear to undergo a shift in frequency. In order to more fully study this effect, special electronic techniques of isolating and measuring the frequency of discrete echo signals are being developed.

Return signal patterns are also observed to be highly dependent upon the type of transducer employed. Short focal length crystals produce many simultaneous signals while flat or long focal length crystals produce only one. A study of the effects of focusing is being done in order to arrive at an optimum test arrangement.

Rough drafts of two topical reports, "Ultrasonic Amplitude Response as a Function of Notch Depth and Other Effects in Zircaloy Reactor Fuel Element Sheath Tubing", and "Masking Ultrasonic Transducers--A Method for Controlling Beam Configurations", were completed and are expected to be ready for publication in April. The terminal report for the first phase work, "Recommended Practice for Immersed Ultrasonic Testing of Zircaloy Fuel Element Sheath Tubing Using Lamb Waves", is expected to be ready for publication by mid-April.

A study of Lamb waves in a cylinder is proceeding. Checks of the frequency equation for a flat plate resulted in a proof that the symmetrical and asymmetrical waves, reported by both Lamb and Worlton, cannot exist in combination. This conclusion does not appear to have been reported earlier. Proceeding to the case of circumferential waves in a cylinder, a frequency equation has been derived for a solid cylinder, and another for a hollow cylinder. The first of these is comparable in complexity to the corresponding equation for a flat plate, and could be used without difficulty if a practical application existed. The second is extremely complicated. Attempts are being made to simplify it. If this cannot be done for the general case, an approximation will be made by assuming the wall thickness to be small compared with the diameter. Enough has been done on this approximation to show it can be done.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Continued analysis of the dispersion data obtained at Cape Canaveral, Florida, and Vandenberg Air Force Base, California, provided a basis for the comparison of the dispersion over a relatively flat area influenced by a maritime climatic regime (Cape Canaveral), a maritime region characterized by heterogeneous terrain (Vandenberg), and a semi-arid region of gently rolling hills (Hanford). This is the most thorough set of diffusion data (275 experiments) known and is especially adaptive to relative comparisons between sites since the same experimental technique was common to all.

The diffusive parameters that have been examined in detail are the width of the cloud as defined by the standard deviation of the cross-wind mass distribution and the peak exposure, which is the highest ratio of the mass obtained at any distance and the flow rate of the sampler collecting the mass. These parameters are sufficient for the evaluation of chronic pollution problems when their dependency on distance is also known.

The results are consistent with those that would be expected from theoretical considerations when the effects of terrain and seasonal climatological variations have been factored into the analysis. Theoretical studies, to date, however, yield only the direction in which the above factors affect dispersion. Reliable quantitative estimates could only have been obtained with experimentation.

The horizontal growth of the plume at all sites was remarkably similar when the seasonal effects and the variation of the wind over the area of the trajectory were considered. The most significant results which summarize horizontal growth are as follows:

1. The narrowest plumes at Canaveral and Vandenberg were observed in the winter, and are related to the circulation controlled by the pressure gradient which is characterized by winds of persistent direction. Plumes released in the day were slightly wider than those released at night during the winter, which again related to the wind direction fluctuations.

2. The plume widths observed in the summer at Cape Canaveral and Vandenberg (flat areas) were much alike and larger than those of the winter. This results from the sea breeze direction variability controlled by land-sea temperature differences.

3. The plume widths obtained during the summer over the rough terrain at Vandenberg were the greatest observed and the rate of growth was also greater than any other. This resulted from the enhanced mixing caused by the roughness of the terrain and by the increasing variability of the wind brought about by the diminishing influence of the sea breeze as the plume travels inland.

4. The Hanford plume growth, controlled primarily by mountain-valley temperature differences were comparable to those data obtained over rough terrain at Vandenberg. Because the data were primarily collected during the summer and the wind fields are similar, these results are not at variance.

The levels and decrease of the peak exposure with distance, like the horizontal plume growth, showed order when the effects of terrain, seasonal circulations, and diurnal variations in atmospheric stability were factored into the analysis. The following results summarize the collective data:

1. The Hanford day, Vandenberg day, and the Cape Canaveral summer day are all quite similar and show the lowest peaks of all the data.

2. The Cape Canaveral winter day is the exception to the results in (1) yielding higher exposures.

3. The highest exposures were observed during the night at all sites because of the diminished vertical diffusion brought about by the stable atmosphere. The data from all sites were very similar. The only deviation from agreement was the data from the rough terrain of Vandenberg. The exposures in this case were about 0.7 of those over the flat terrain at comparable distances.

Compilation and screening of all data obtained during ground source diffusion experiments at Hanford progressed with forty-six tests meeting the stringent quality criteria. The summary includes 1) normalized peak exposure, 2) wind-speed normalized peak exposure, 3) normalized crosswind integrated exposure, 4)standard deviation of the crosswind mass distribution, and 5) wind speed and stability data. Recalculations of "Green Glow" exposure and moment data following a new editing procedure showed only trivial differences from results reported earlier.

As part of the continuing quality control program in assaying filter samples for deposited zinc sulfide, filters assayed on August 23, 1962, were reassayed on February 14, 1963. No significant departure was found in counting rate or counting variance, indicating good system stabilization.

Vendor modification of the 3-level micrometeorological measuring system was essentially completed during the month, with only one piece damaged in shipment remaining to be repaired off-site. Tower mounts for the refurbished portable mast instrumentation were designed and fabricated.

Dosimetry

Information was received about the results of our whole body counting experiments at Swedish Hospital last November. Previously unknown tumors were found in two people who were scanned after administration of an I-131 tagged compound. Our estimates of the amount of potassium removed from people being treated with the artificial kidney were much higher than those from chemical measurements on products from the artificial kidney; we do not know why the estimates differ but are trying to find out.

Radiation Protection Operation was assisted further in the preparation of their mobile shadow shield counter. The shield is completed. Counting instruments are being installed. A new scintillator for use in the counter was found to have a high background and was returned to the manufacturer.

Detection of beta rays emitted from the surface of the body is being studied for possible use in the monitoring of pure beta emitters in the mobile shadow shield facility.

A program of measuring Cs-137 in urine samples of Alaskan Eskimos was begun. Parts of the state not visited with the whole body counter are being examined this way. Samples are being obtained through the U. S. Public Health Service.

The study of X-ray scintillation counters continued. It was found that use of quartz in place of glass in the scintillation crystals reduced the background. 9514-S photomultiplier tubes were found to be the best, i.e., lowest noise, of the EMI tubes tested. The pulse height spectrum of the noise was found to be a combination of two exponential distributions in the case of several different phototubes. This phenomenon is being studied in hopes it will help in the interpretation of noise phenomena.

Annual maintenance was performed on the Van de Graaff. Installation of a new stapled belt did not improve the operation. Installation of a new accelerator tube did seem to cure the problems we have had recently.

A study was made of the response of the polyethylene double moderator as a function of the angle of incidence of the neutrons in order to correct measurements of cosmic ray neutrons.

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A calorimetric measurement of the specific power of the radiation from promethium-147 is nearly done.

The study of the heat amplifier is continuing. A family of amplifier characteristic curves for operation at 41°C was prepared. Maximum amplification of about five, about the same as in the previous study, was obtained. Characteristic curves for dry ice temperatures are now being obtained.

Instrumentation

The solid-state detector sensor elements planned for use in the solid-state multiple detector neutron dose meter has been received. An invention report outlining the theoretical basis for the device is being prepared.

Testing of the experimental single-trip point (50 mr) signalling personnel dose meter was started in cooperation with Radiological Development, Radiation Protection Operation. Tests have not been completed as yet. Experiments continued with the second experimental dose meter which employs a conducting Teflon center rod and a platinum recharging fiber. Extended testing has shown this combination reduces the tendency for the recharging fiber to stick.

Experiments were conducted with several types of oscillator circuits for use in biology data telemetering transmitters. The best performing type was selected and designed for 46 kc/s operation. The best oscillator proved to be a phase-shift type tunable over a limited range. Modulation experiments were started.

Progress was made in experiments to accurately measure air filter collected Pu-239 using CsI crystals of 2-inch diameter by 0.003 inches thick for scintillation alpha energy spectrometry. The Pu-239 peak falls slightly below that for radon; therefore, by using two single channel pulse height analyzers, it is possible to select pulses from the higher energy alphas in the radon decay chain to "balance out" those which fall in the slightly lower energy Pu-239 channel. Introduction of known Pu-239 sample sources provided direct and distinct readout through the lower channel. This approach will be actively pursued. The advantage of the 2-inch diameter CsI detector over solid state surface barrier detectors is that of considerably larger effective detection area.

Three sensitive scintillation detector probes were designed and assembled for low-level, low-energy beta detection for Pm-147. Each probe employs a 5-inch diameter EMI 9530-S phototube and a 0.01 inch thick NE-102 (terphenyl-in-polyvinyltoluene) detector covered by two layers of 1 mg/cm^2 double aluminum coated Mylar. The probes of 80 cm^2 effective area are used

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with the HAPO model II Scintrans, and each probe has a detection sensitivity of 1.6×10^{-5} μc/cm^2 or 1.6×10^{-8} μg/cm^2 of uniformly distributed Pm-147 as compared to 3.2×10^{-4} μc/cm^2 detection sensitivity for HAPO mica-window GM tubes for similar source distribution conditions. The probes provided good service during the 325 Building Pm-147 contamination problem.

A modified detection head assembly was designed for use in the in-field, continuous luminescent particle detection system for use in atmospheric diffusion and transport studies. The modified head permitted faster ZnS particle transport from the light activation chamber to the photo-tube portion. This method provides a larger light availability from the decaying particles so a pulse-type electronic system can be used in place of the present direct current method to provide simpler stabilization for temperature variations. In addition, particle size analysis should be obtainable.

Experimental circuits were designed and are being fabricated for measurement of fission product contamination on air filters under relatively high and varying gamma background conditions. The scintillator being used is diphenylbenzene suspended in toluene and sprayed onto the photo-tube to about 0.015 inch thickness. The approach proved superior to anthracene and to terphenyl-in-polyvinyltoluene. Initial tests indicate that a mixed fission product concentration of 1×10^{-9} μc/cc could be detected in 10 minutes for a 10 CFM air flow under gamma background conditions of 40 to 50 mr/hr.

Development, design, and experimental fabrication proceeded on the Atmospheric Physics experimental portable mast data logging system. Discriminators were added to the parallel-to-serial converter to eliminate intermittent paper punching difficulties. An electronic escapement circuit with a quick recovery monostable multivibrator was designed for use with the paper tape punch. This method eliminates the mechanical contact function of the tape punch to provide a considerable increase in performance reliability. Internal cabinet component layout and connector mounting was completed for the cabinet which will be used at the base of the tower. Six wind speed solid state amplifiers were completed, and the reed relay circuits for thermocouple monitoring were completed. The developed fast recovery monostable multivibrator has a 98% duty cycle with pulse widths of 9 milliseconds with a recovery time of about 30 microseconds. Many other applications are foreseen in pulse generators and superior count-rate-meters.

A circuit design was completed for use by Biology personnel in thyroid uptake measurements of I-131.

Check-out of the 400-channel analyzer is continuing. Modifications were made to the address logic to allow easier switching to split memory sections. The logic is now ready for the development work of the analog-to-digital networks.

A magnetic tape record was received from the Atmospheric Physics Operation which contains data on wind speed, and horizontal and vertical components of wind direction. The tape was received for statistical analyses utilizing the transfer function analyzer and an analog computer.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

The mass spectrometer for this program provided isotopic analyses of program samples received during the month.

Development of a scintillation-type ion detector continued on the ion test bench. Pulse height distributions were obtained for monoenergetic electrons with energies from 30 to 60-kev incident on a plastic scintillator. Distributions were also measured for the 42-kev secondary electrons resulting from 48-kev cesium-133 ions incident on an aluminum target. These preliminary measurements indicate that on the average about six secondary electrons are produced for each cesium ion.

TEST REACTOR OPERATIONS

Operation of the PCTR continued routinely during the month. There were two unscheduled shutdowns, one caused by electronic failure and one by faulty by-passing technique. The experiment to determine k_∞ data for Pu-Al (20% 240), 19 one-half inch fuel element clusters in $10\frac{1}{2}$ inch graphite lattice was completed during the month. Null reactivity measurements and foil activation traverses were made.

Operation of the TTR was on an intermittent basis during the month. There was one unscheduled shutdown due to electronic failure. The TTR was made available to the University of Washington Graduate Center on a two-nights-a-week basis during the month.

CUSTOMER WORK

Weather Forecasting and Meteorological Service

Consultation service was rendered on meteorological and climatological aspects of 1) oxides of nitrogen release in 300 Area to IHO, 2) low altitude

sampling of fallout to CR&D, 3) NPR Hazards Appraisal, 4) sulfuric acid mist release from the 300 Area Power House, and 5) scheduling of boiler repairs in N-Area. A forecast of temperature and flow-rate of the Columbia River following the peak runoff season of 1963 was prepared.

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

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	84	83.1
24-Hour General	56	83.0
Special	147	93.2

February was warmer than usual with near-normal precipitation. There was less wind and much more fog than in any previous February since the station was established.

Instrumentation and Systems Studies

The automatic conveyor-type protective clothing contamination monitor operated another month without difficulty or malfunction.

The field-model coincidence-count alpha particulate air monitor for Pu-239 continued to perform correctly for another month in the 308 Building.

Discussions were held with Radiation Monitoring personnel of CPD regarding their interest in fabricating both a coincidence-count alpha particulate air monitor for Pu-239 and several air filter counters which utilize the same counting technique to reduce the interference effects of radon-thoron buildup.

The special light-level detector which provides an alarm in the event of plutonium oxalate buildup in Hood 9 was completed, tested, and delivered to Plutonium Process Engineering, CPD.

Circuit, layout, and other assistance was provided 328 Electronic Shop in support of their efforts to fabricate a Mark II Logarithmic and Linear Response Scintillation Area Monitoring Instrument. The unit will be used by Facilities Engineering, HL.

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A printed circuit master, suitable blueprints, and a sample prototype circuit for a special annunciator for use with HAPO portable rate-type survey meters were prepared for Radiological Development and Calibrations, HL. A number of these plug-in units, proved highly successful in field tests, will be fabricated off-site.

Two Pu-239 air filter counters employing coincidence-count techniques in order to reduce the interference effects of radon-thoron buildup are being assembled. One is for Radiation Monitoring Operation, HL, for 327 Building, and the second is for Chemical Development, HL, for FRPP use.

Advice and assistance was rendered to Chemical Research, HL, regarding commercial logic circuits, control flip-flop design, and circuit troubleshooting.

A transistor voltage amplifier of wideband type was designed for Chemical Effluent Technology, HL.

Two new precision temperature control systems were installed at 100-KW for use on the uranium swelling capsule program. Shakedown of the systems was accomplished on an ex-reactor mockup capsule. The systems are now on-line, controlling two in-reactor capsules. Two over-temperature systems are currently being used for equipment protection. This arrangement allows a much smaller temperature differential between operating temperature and experiment trip point since precision temperature monitors (which are not completely fail-safe) can be used. Reactor protection is then accomplished by the same fail-safe system as before. The 60 point Minneapolis-Honeywell data logger is now on line logging the temperatures of the two swelling capsules. Other than an occasional feedback alarm from the IBM typewriter, the data logger is performing satisfactorily. Construction of the creep capsule 96-point data logger is progressing. Ninety percent of the parts have arrived and an instrument assembly has begun.

Ultrasonic inspection and recording equipment has been developed for the testing of special nuclear materials within the 234-5 Bldg. Preliminary evaluation shows the equipment sensitivity capable of resolving 13 mil flat bottomed holes drilled 20 mils below the test specimen surface. Particularly good defect recording resolution has been achieved by restricting the effective ultrasonic beam diameter to approximately 20 mils. The system has been designed for routine operation, and is suitable for production line applications. The equipment has performed well during preliminary evaluation tests. It will be made operational as special test hoods become available.

Development of a prototype channel of instrumentation for monitoring PRTR fuel assembly vibrations is estimated to be 80% complete. Purchase specifications have been written and issued for most of the components that will be required for the fabrication of a six-channel motion analyzer for use in a mockup facility. The completed instrument will utilize eddy current sensing coils built into a special fuel assembly, and the readout will be presented both on a strip chart recorder and on an oscilloscope.

Optics

The electrical readout traverse mechanism (ERTM) being built for Irradiation Testing, IPD, for use in determining the distortion of old reactor process tubes was tested for range, sensitivity, and linearity. Feeler contact range is ± 0.96 inches, sufficient for any process tube which will accommodate an eight-inch fuel element, in comparison to the ± 0.6 inch range of the existing optical unit. The sensitivity is 0.0007 inch per digit with ± 1 digit repeatability. This is equivalent to the 12-12 optical unit. Linearity was measured as 2.6% of full range.

Tests were also made to determine the effect of vertical displacements upon horizontal readings. With proper adjustment the horizontal readings were observed to be constant to within 10 digits for extreme vertical bending.

Studies have begun on the properties of luminescent materials and laser materials for the purpose of developing a laser pumped with nuclear energy.

During the three-week period (January 27-February 17) included in this report, a total of 360 man-hours shop work was performed. The work included:

1. Fabrication of two quartz seal plates for Ceramic Fuels Research.
2. Evaporation of Cobalt on 20 targets for Nucleonic Instrumentation, HL.
3. Fabrication of parts for an ultrasonic probe manipulator for Physical Measurements, HL.
4. Fabrication of a small lens for a film densitometer for External Dosimetry, HL.
5. Modification of a roll film camera for use on a Model L B&L Macro camera, Photography Operation and Ceramic Fuels, HL.
6. Repair of two process tube borescopes for Irradiation Testing, IPD.
7. Fabrication of six quartz rings for Radiometallurgy.
8. Repair of one bench microscope for Metallography Labs, HL.
9. Repair of one crane periscope head for Redox, CPD.
10. Modifications to a solar cell pyrometer for Ceramic Fuels, HL.
11. Modifications to a cut film camera back for Photography Operation, HL.

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Physical Testing

A nominal increase in testing service work has occurred as increased emphasis was placed on the K-reactor process tube pre-installation examinations. A total of 5,825 tests, up from 2,737 for the previous month, were made on 3,727 items representing 78,035 feet of material. Use of Physical Testing services increased to include thirty-two HAPO operating departments and service organizations and two AEC contractors. Advice was given on eighty-one different occasions on general testing theory and applications.

An initial lot of fifty Zircaloy-2 process tubes for the K-reactor's re-tubing program was released for reactor use upon completion of test evaluation. To date, twenty-six tubes have been rejected.

Field activities consisted of magnetic particle inspection of a railroad cask car and pump shaft couplings; fluorescent penetrant examination of a downcomer expansion joint, and converted fuel element nozzles.

A technique utilizing a portable Iridium-192 source has been developed for the in-place radiograph inspection of plastic blocks within 231 Building. The new method eliminates the need for dismantling and transporting the specimens to 234-5 Z Building for inspection purposes.

Special equipment needed for the heat transfer testing of I & E fuel element and caps was designed, installed and test run against a standard.

Changes in the ultrasonic tube test equipment was made to test Zircaloy tubing in the range of .422 inch O.D. x .020 inch wall x 52 inch in length. Notch standards were fabricated on both the O.D. and I.D. surfaces of one, two and three mils in depth to a length of forty mils each. A 15 mc, 3/16 inch point focus crystal proved to have the beam size required for the longitudinal test. Instrument modifications were made to increase the repetition rate to 5 kc. For the transverse test a 10 mc, 3/8 inch point focus crystal was used.

A single transducer, pulse-echo technique was developed and applied to a bond test of six zirconium clad, thorium-uranium oxide fuel elements fabricated at the Idaho Falls reactor site. Gross unbonding was found.

Development work for bond testing of aluminum clad, Pu-Al fuel elements was started. These elements will be used in the C-1 loop to obtain corrosion data.

Work was started to determine the physical properties of uranium oxide-tungsten ceramic metals of various composition. Data on the shear modulus,

bulk modulus, rupture modulus and modulus of elasticity will be obtained from the tests.

The adaption of in-motion radiography of circular patterns marks an advancement in weld, eutectic braze and unbond evaluations. Excellent correlation of surface and internal discontinuities with a sensitivity of 1% was made on circumferential welds of dissimilar metals, such as Haynes 25, Hastelloy and stainless steel. Linear motion radiography is progressing satisfactorily with a view of NaK swelling capsules prior to irradiation. The parallax free radiograph will permit accurate sectioning of the capsule after irradiation.

ANALOG COMPUTER FACILITY OPERATION

The analog computer problems considered during the month included:

1. NPR Pressurizer Injection System.
2. NPR Graphite Coolant Gas Composition.
3. D-Reactor Automatic Control.

The purchase requisition for a new 156 amplifier analog computer was processed and requests for quotations were sent out. The bids are to be opened March 4.

The analog computer facility is now on a scheduled two-shift operating basis in order to handle the increased computing load. The facility is scheduled for three-shift maintenance starting in March.

Eighty-five percent of the GEDA and ninety percent of the EASE equipment was in good operating condition this month. Computer utilization was as follows:

GEDA	EASE	
133	133	Hours Up
19	19	Hours Scheduled Downtime
152	152	Hours Total

The Mark 200 recorder has been placed on a weekly routine to ensure proper operation and calibration.

INSTRUMENT EVALUATION

The evaluated production prototype combined alpha-beta-gamma scintillation solid-state hand and shoe counter, one of seven being fabricated off-site by Instrument Laboratory, Inc., of Seattle, was returned to the manufacturer. Detailed information was forwarded for error correction and modification of the instruments and a cost estimate was prepared regarding the added work necessary.

Final laboratory calibration, battery life, and temperature tests were nearly completed on the experimental solid-state circuit GM portable survey instrument which has both a count-rate meter and a gated oscillator annunicator. Field tests will follow.

RS Paul

Manager
PHYSICS AND INSTRUMENTS LABORATORY

RS Paul:mcs

CHEMICAL LABORATORY

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Effluent Water Radioisotope Studies

The two reactor tubes, one aluminum and one zirconium, being cooled in these studies by deionized water, are now charged with zirconium clad fuel elements. Based on Ga-72 and Na-24 concentrations in the effluent from these tubes, the relative corrosion rates of tubes and fuel elements may possibly be estimated. On January 29, changes were made in the bed regeneration procedure for the deionizers which have resulted in water of 2 to 2.5 megohm resistivity, an improvement over the maximum 1.5 megohms previously obtained. The increased purity, however, has had a negligible effect on the concentrations of radionuclides in the effluent water. These data add further confirmation that removal of parent material alone from the process water is not sufficient to effect large reductions in radionuclide output in the effluent water.

Filter breakthrough problems have continued to hamper tests at the Water Treatment Pilot Plant. A core sample taken recently from the filter bed indicates that the sand had shifted in a manner which may explain the poor performance. The recent cold weather period seemed to hinder floc formation and growth sufficiently to reduce the filter efficiency. The next test for this operation will be a test of aluminum nitrate as the flocculating agent and nitric acid for pH adjustment.

Various coating agents were applied to aluminum coupons and evaluated in laboratory tests for their ability to inhibit adsorption of arsenate ion tagged with As-76. Coatings of many silicones, epoxy resins, coumarone-indene binding resins, and other agents were compared to the marking ink previously found to be effective. The adsorption was measured after several days' exposure to boiling process water. Under these conditions some of the silicones, epoxy resins and coumarone-indene resins were as effective as the ink in reducing the adsorption (about one-third that of an untreated coupon).

Release Studies

Tests were conducted to find an available material suitable for containment of molten bismuth in "B" metal release studies. Of the

materials tested, only a quartz-lined stainless steel boat withstood the high temperature and oxidizing atmosphere. The tests showed that molten bismuth would destroy the integrity of a stainless steel cask within a few minutes if the temperature approached 1300 C.

SEPARATIONS PROCESSES

Disposal to Ground

Water samples from wells monitoring the Redox 216-S-7 and Purex 216-A-10 process condensate cribs were collected daily for one week and analyzed for long-lived radionuclides. Results show total strontium present in the ground water beneath the 216-S-7 crib in concentrations ranging from 1.0×10^{-7} to 2.4×10^{-6} μc/cc in the fifteen samples analyzed. Strontium results from wells monitoring the 216-A-10 were below the detection limit of 7×10^{-8} μc/cc. Cesium-137, Sr-90 and Co-60 analyses have not yet been completed. Recommendations for replacement of the 216-S-7 crib were forwarded to the Chemical Processing Department.

Shallow wells in the Purex 241-A tank farm were monitored for the first time with a gamma scintillation probe on December 28, 1962, and results indicated possible contamination in the areas of a few wells. (Previous monitoring results with the less sensitive GM probe were negative.) The wells which showed greater than background (nominally 1100 cpm) counting rates were reprobed with the gamma scintillation probe on February 7, 1963. The only significant change noted was an increase of 1600 cpm at the 74-foot level in well #20 which is located on the south side of the 106 tank. Periodic resurveys of these tank monitoring wells are scheduled in order to assess the significance of these data.

A series of tests was completed simulating leaks in a direct-buried pipeline which will be used to transport fission product recovery solution between C-Farm and B-Plant. The tests were conducted by pumping water into sections of one-inch pipe buried to a depth of seven feet in the immediate vicinity of the western terminal of the transport line, the 241-B-154 diversion box. Evaluation of the empirical results showed that leak rates of 10 to 20 gpm would likely result in ground surface contamination if sustained for a period of from 30 minutes to several hours. Lesser simulated leak rates, 0.5 to 3 gpm, did not result in moisture approaching closer than four feet to the surface, even after five hours pumping time. The probability that ground-water contamination will result from such leaks is unlikely, unless detection is delayed significantly and prolonged leakage at low rates occurs, due to the relatively large

volume of vadose zone soil available for specific retention storage. Evaluation of the heating of the contaminated soil mass around a pipe leak by radioactive decay of sorbed and precipitated radionuclides indicates that relatively high temperatures might be reached in a matter of weeks, unless provisions are made to improve the transfer of heat away from the contaminated zone.

Redox Process Water Treatment

The anion resin in Unit 2 of the Redox demineralized water system was replaced with a new batch of Nalcite SAR. Very good water was produced for about five regeneration cycles (ca. 100,000 gallons of water treated). Shortly after February 7, 1963, the water quality dropped and efforts to improve the quality were unsuccessful. This period of time coincides with the appearance of significantly higher turbidity at the 283-W water treatment plant. This fragmentary evidence indicates organic fouling of the anion resin occurred.

Use of Uranium(IV) in the Purex Partition Columns

Plutonium reflux could be a problem when uranium(IV) is used as reductant in the Purex 1B partition columns since the scrub section will operate with only a holding reductant (hydrazine) present. In the current flowsheet, a reducing agent (ferrous ion) and a holding reductant (sulfamic acid) are present. Batch contact studies designed to measure the rate of oxidation of plutonium in the scrub section as a function of aqueous nitric acid and hydrazine concentration were made. The rate of oxidation of plutonium was not markedly influenced by nitric acid concentration in the 2 to 4 M range. A hydrazine concentration of 0.02 M in the 1BP stream appears adequate to limit plutonium reflux to the same level maintained in the present flowsheet.

WASTE TREATMENT AND FISSION PRODUCT EXTRACTION

CSREX Process - Laboratory Studies

Laboratory work on the CSREX process (for simultaneous solvent extraction of cesium, strontium and rare earths) included: studies aimed at improving the kinetics of cerium and promethium extraction; hot cell experiments with full-level Purex 1WW; mixer-settler runs with tracer-level Purex FTW; and solvent-extraction tests on simulated Redox waste.

Extraction of cerium and promethium with either CSREX solvent (BAMBP-D2EHPA mixtures) or D2EHPA alone from citric acid complexed solutions is much slower than required for efficient recovery in counter-current

pulse columns. While the kinetics can be considerably improved by operation at elevated temperatures, this mode of operation is deleterious to cesium equilibria, and an alternate solution would be desirable. Cerium and promethium kinetics and extraction coefficients were found to be much improved by substituting tartrate or HEDTA for citrate; however, there is evidence that iron may also be extracted from the latter solutions (based on earlier experience with D2EHPA only). Tartrate and HEDTA have the further disadvantage of being more expensive than citrate. Experiments performed during the month showed that the advantages of both types of complexant can be achieved by using citrate-tartrate mixtures. Sufficient citrate is used to complex all of the iron, chromium, nickel and aluminum, and 0.05 to 0.1 M tartrate is used in addition. Cerium (and promethium) extraction is at least five-fold more favorable than with citrate only.

A hot-cell CSREX test was conducted with full level Purex 1WW. The feed was complexed with 0.5 M citrate and 0.1 M tartrate and batch extracted with 0.5 M BAMBP, 0.3 M D2EHPA, Soltrol-170 at a pH of about 4.4. Equilibrium for both cerium and cesium (followed with the in-cell gamma spectrometer) was achieved in less than two minutes. Three successive extractions removed essentially all of the cerium and 98.5 percent of the cesium, in good agreement with expectations based on prior tracer-level work. Decontamination from ruthenium and zirconium-niobium was excellent, both being below detection limits in the organic phase. Strontium results are not yet available. Portions of the loaded organic will be stripped after various periods of time to determine the extent of radiation damage.

Tracer-level mixer-settler runs are being conducted with feed stock prepared from synthetic FTW to demonstrate CSREX Process Flowsheet No. 2 conditions. The runs are designed to provide loss and decontamination data as well as to provide organic solutions for subsequent solvent wash studies. Under extraction column conditions, cesium, strontium and cerium losses were 0.17, 0.46 and 8.8 percent, respectively; decontamination factors (based on cesium) for sodium, ruthenium, and zirconium-niobium were 23, 470 and 1200, respectively. Cesium and strontium losses under partition column conditions were 2.1 and 0.72 percent, respectively; cerium decontamination factor was 130. The cesium loss under strip conditions was 3.4 percent.

Use of cesium-only (BAMBP-Soltrol) and CSREX solvents for extraction of Redox wastes was scouted briefly. Cesium, strontium and cerium were all readily extracted from acidic Redox waste (complexed with citrate and tartrate and adjusted to pH 4.5) with CSREX solvent. Cesium was likewise easily extracted from alkaline Redox tank supernate with 0.5 M BAMBP, Soltrol-170 (extraction coefficient, $E_a^o = 11.6$).

Unlike Purex supernate, which must be butted with ca. 0.2 moles of NaOH per liter to achieve favorable extraction coefficients with BAMBP (in the absence of synergism), the Redox waste is already sufficiently alkaline, and no addition is required. The 0.01 M dipicrylamine-nitrobenzene also yielded favorable extraction coefficients. Significance of these experiments is that the solvent-extraction equipment proposed for B-Plant operations may have the dual capability of processing Redox as well as Purex wastes.

In other solvent extraction work, dipicrylamine extraction measurements were made which allowed assignment of equilibrium constants for each of the alkali metals to the reaction:

$$M^{+}_{(a)} + DPA^{-}_{(a)} \rightleftharpoons M \cdot DPA_{(o)}$$

The values of K ranged from 150 for Li^{+} to 2×10^{6} for Cs^{+}, with the pK values exhibiting a linear dependence on ionic radius. The results are included in a comprehensive formal report (HW-76222) summarizing all of the dipicrylamine studies.

CSREX Process - Engineering Studies

Additional pilot plant runs have confirmed the satisfactory performance of the CSREX process reported last month. Most of the new runs were devoted to improving cerium extraction and sodium scrubbing in the 1A column. Highlights of the runs were as follows:

1. Cerium extraction was only slightly improved by using an HEDTA (hydroxyethylene diamine triacetic acid) complexed feed in place of the citrate-complexed feed. The cerium loss at room temperature was reduced from 34 percent to 26 percent, within the range of experimental error. This performance disagreed with laboratory studies which indicated cerium extraction to be greatly improved with HEDTA.

2. Use of a mixed tartrate-citrate feed lowered the cerium waste loss from 49 to 35 percent. The feed contained 0.015 M chromium, 0.05 M tartrate and 0.135 M citrate, with a total of 0.05 M excess complexing agent.

3. Sodium scrubbing was effectively carried out using a 1 M formic acid solution neutralized to a pH of 2.4. The scrub section sodium decontamination factor was 9 at an aqueous to organic flow ratio of 0.2 in the 9-foot-tall column. A decontamination factor of 20 was obtained by increasing the formic acid concentration to 1.25 M. Strontium and cesium reflux was less than eight percent in all runs.

4. Efficient sodium scrubbing in the 1A column permitted use of lower acidities and aqueous to organic flow ratios in the 1B column than were required in the earlier runs. The desired pH profile was achieved with a 1BX stream of 0.035 M HNO_3 at an aqueous to organic ratio of 0.6. The strontium and cesium losses, however, were in the four to six percent range, or about four times the values reported last month.

5. Cerium waste losses of three to four percent were obtained in the 1C column using 0.5 M HNO_3 at a volume velocity of 400 gph/ft^2 (sum of both phases); the loss increased to eight percent on raising the flow rate to 600 gph/ft^2. Generally, the column capacities and the behavior of strontium and cerium with the CSREX flowsheet parallel the corresponding performance with the D2EHPA flowsheet. Consequently, it should be possible to extrapolate many of the earlier observations with D2EHPA to the new CSREX process. Preparations are being made to test the CSREX process in a three-column cascade. Column diameters are all 3-inch-diameter, and the heights are approximately equal to those allowable in B-Plant.

Cesium Recovery by Ion Exchange

Radiochemical analyses of feed and eluate samples from laboratory column experiments with Linde AW-400 exchanger and actual aged Purex alkaline supernatant waste show the following decontamination factors (DF) from Cs-137:

Isotope	DF
Zr-95	59
Nb-95	19
Ru-106	590
Ce-144	22
Sr-89,90	49

These decontamination factors are believed adequate for cesium recovery from aged alkaline supernatant wastes.

The effect of feed flow rate on the AW-400 alkaline supernatant waste flowsheet has been determined with synthetic waste. At 0.5, 1, 2, 3 and 4 column volumes per hour the one percent cesium breakthrough was at 34, 25, 22, 20 and 18 column volumes, respectively. Flow rate reduction of eluant from 2 to 0.5 column volumes per hour only decreased the volume of 8 N $(NH_4)_2CO_3$ required for 90 percent cesium elution from 8.0 to 7.4 column volumes.

An experiment was run with AW-400 to determine the effect of the sodium to cesium ratio on the cesium elution efficiency. The volumes of 8 N $(NH_4)_2CO_3$ required for 90 percent cesium elution were 7.2, 9.3 and 11.0 where the columns had been loaded with solutions with sodium to cesium ratios of 1×10^1, 1×10^3 and 1×10^5, respectively. This indicates that elution would be easier after loading with Purex formaldehyde-treated waste (FTW) than with Redox supernatant waste, for example.

A sample of Norton's new "Zeolon B" was received and tested for cesium recovery from both Purex FTW and alkaline supernatant waste. The new Zeolon was very hard and had a high resistance to breakage. Cesium breakthrough (50 percent) was at about 100 column volumes for FTW which is somewhat better than with clinoptilolite. However, kinetics and elution were somewhat poorer. Twelve column volumes of 8 N $(NH_4)_2CO_3$ were required to remove 90 percent of the cesium from the exchanger. With alkaline supernatant waste, 50 percent cesium breakthrough was at 18 column volumes which is about the same as clinoptilolite and Duolite C-3, but much poorer than Linde AW-400 and AW-500. It appears that the new Zeolon (bonded differently than the original sample) is slightly more selective for cesium over sodium than the original Zeolon.

Sugar Denitration of Purex Waste

The sugar treatment process for the denitration of Purex waste was successfully tested in the Purex Plant during the month, culminating the laboratory and Cold Semiworks development. A 2500-gallon batch of 1WW, initially 4.5 M H^+, was heated, and 1600 pounds of sucrose were slowly added. Final hydrogen ion concentration was 0.68 M (i.e., approximately 85 percent of the initial acid was destroyed), and the efficiency of reaction was 18.2 moles of nitric acid destroyed per mole of sugar added, in good agreement with the laboratory predictions. Additional plant tests are planned.

Because of the high cost of refined sucrose, several less expensive (semi-refined) sugar concentrates are being tested in the laboratory. Very promising preliminary results have been obtained with a molasses (which costs only about a third as much as sugar); analytical results on these experiments are not yet complete.

Treatment of Purex Stored Waste Sludge

As part of the waste management program, studies aimed at development of flowsheets and procedures for treatment of Purex stored

waste sludge were continued. Experiments performed so far have been with samples from the 3-inch-long core and from the mid-section of the 30-inch-long core taken from the 103-A tank. The two materials show some differences in chemical composition, but their behaviors toward leaching with water are very similar. The experiments this month have emphasized leaching the sludge at relatively low pH to destroy the large amount of sodium carbonate present in it. A portion of water was added to a sample of sludge, then nitric acid was added slowly until a pH in the range 3.4 to 7 was reached. From 9 to 10 meq of nitric acid per gram of sludge were required to reach these terminal pH values. Strontium removal varied from 8.8 percent at pH 7 to 67 percent at pH 3.4, compared to less than two percent with exhaustive water leaches. While these results are not encouraging from the standpoint of strontium solubilization with the leach, the procedure may still be of interest as a means of disintegrating the hard top layer of sludge. The sludge remaining after these leaches was treated with 1 to 2 M HNO_3. An average of about 15 percent of the strontium remained undissolved.

Solvent Extraction of Neptunium and Plutonium from Purex Waste Solutions

Studies to define a flowsheet for batch trilaurylamine (TLA) extraction of neptunium and plutonium from Purex Plant 1WW solutions were continued. In the current study flowsheet, 1WW solution is made 0.02 M in hydrazine to reduce Np(V) to Np(IV). Neptunium(IV) and Pu(IV) are then co-extracted with an equal volume of 0.3 M TLA-Soltrol and are co-stripped with one-half volume of 0.1 M oxalic acid. In a demonstration of this flowsheet, one batch of solvent was processed through six successive extraction-strip cycles (aqueous feed was synthetic 1WW solution of approximately current composition). Extraction neptunium and plutonium losses remained constant at about 3 to 5 and 1 to 2 percent, respectively. In each strip contact, about 80 percent of the neptunium and plutonium present were removed.

In another demonstration, one batch of solvent was contacted with six equal-volume portions of aqueous feed (synthetic 1WW) without intervening strip contacts. Plutonium losses increased from two percent in the first seven percent of the last contact. Similarly, neptunium losses increased from five to 20 percent.

In other experiments, little or no Np(IV) was extracted when synthetic FTW solutions of estimated 1965 composition were contacted with the TLA-Soltrol extractant.

Plutonium not stripped from solvent with oxalic acid was readily removed by contacting the solvent with one-tenth volume of 2 M NaOH - 0.1 to 0.5 M Na_2CO_3 solution.

Promethium Purification and Shipment

Analytical results verified last month's conclusion that the "heart cut" of over 10,000 curies of promethium from the recent C-Cell ion-exchange purification run is indeed chemically and radiochemically pure, with all impurities below analytical detection limits. Most significant was the complete separation achieved from europium, americium and curium, which are not separated from promethium by other reported promethium purification processes, and whose removals are important for some applications.

Following measurement of the promethium neutron absorptium spectrum (by Nuclear Physics Research Operation), 12.90 grams of promethium oxide (10,000 curies) were loaded into a double-walled container (a screw-capped aluminum ampoule inside a stainless steel tube) and removed from the cell (free of all smearable contamination) and placed in a 200-pound ("belly band") pig for shipment. Radiation reading through the side of the pig was less than 2 mr/hr.

Measurements made on purified promethium during the month included a preliminary half-life determination (2.69 ± 0.1 year), measurement of the optical absorption spectra of promethium in perchloric, nitric and sulfuric acid media, and determination of the X-ray diffraction pattern of Pm_2O_3. Promethium oxide was found to have a fluorite type structure. A shift of the 545 mμ absorption peak of Pm^{+++} in 1 M HNO_3 and 1 M H_2SO_4, relative to perchloric acid, indicated at least some slight complexing which will be explored further at higher acid concentrations. The X-ray diffraction patterns of promethium oxides prepared by different routes will also be compared and the effect of decay of promethium to samarium determined.

Fission Product Packaging

Samples of eluate from three different inorganic ion exchangers which had been heated to 1000 C before elution were analyzed for Al^{+3} and Si^{+4}. When Linde AW-400 was eluted with 2 M NH_4NO_3, 0.025 percent of the silicon and 25 percent of the aluminum from the exchanger were found in the eluate. When Linde 13-X and Decalso were eluted with 1 M HNO_3, less than one percent of the silicon was removed from each; all of the aluminum was dissolved from the 13-X and about one-half from the Decalso.

A study on the elution and vaporization of radioactive isotopes loaded on zeolites and heated to 700 C is almost complete; long-time heating runs are now in progress. Results so far show that the amounts vaporized at 700 C are comparable to those vaporized at 1000 C (reported last month, HW-76315 C). The completeness of the elution of Sr-85 and Cs-137 after heating to 700 C was markedly improved over that after heating to 1000 C. For example, 20 column volumes of 1 M HNO_3 removed only 86 percent of the Cs-137 and 62 percent of the Sr-85 from 13-X that had been heated to 1000 C, whereas greater than 98 percent of both isotopes were removed from the same material heated to 700 C. The amount of Sr-85 removed from 4A also increased from 56 to 98 percent under comparable conditions.

Scouting studies indicate that cerous ion is readily adsorbed on 13-X zeolite. Results from equilibrium experiments show 1.9 meq of Ce^{+3} adsorbed per gram of 13-X from a solution of 0.006 N Ce^{+3} and 0.3 N Na^{+}. The rate of Ce^{+3} adsorption on 13-X appears to be satisfactory for practical use. These preliminary findings show that zeolites may be suitable for storage and/or shipping of recovered rare earth isotopes.

EQUIPMENT AND MATERIALS

Air Pulser Studies

The Recuplex air pulser system was installed on the D2EHPA 1B-1C column and compared with the standard mechanical bellows pulser. There was no discernible difference in either the extraction efficiency or the capacity attainable with either unit. The poor performance in shakedown runs of this unit in Recuplex is tentatively attributed to a high back pressure in the air exhaust, leading to a lower-than-anticipated pulse amplitude-frequency product. As installed and operated in the pilot plant, the air pulser performance was very satisfactory.

Stress Corrosion Cracking of Mild Steel in Nitrate Solutions

Two additional 3 ft. x 3 ft. x 3/8 in. weldments of ASTM A 201-57T mild steel were fabricated and are being exposed to synthetic Purex alkaline waste. A weld repair was made on one of the weldments prior to exposure.

Notched and stressed 1 x 3 x 1/4 in. specimens of mild steel ASTM A 283 and A 201 are being exposed to boiling 50 percent $NaNO_3$ and 50 percent $NaNO_3$ - 20 percent NH_4NO_3. To date, two specimens have

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cracked. An A 201 specimen stressed to 100 percent yield cracked after seven days' exposure in the sodium nitrate-ammonium nitrate solution. An A 283 specimen plastically deformed in tension (ca. 3 percent) and stressed to 100 percent yield cracked after five days' exposure to the sodium nitrate solution.

Test for Susceptibility to Intergranular Corrosion

A test based on the number of coulombs per unit surface area required to attain passivity in the sulfuric acid - nitric acid system was evaluated. Optimum solution composition for the test appears to be about 1 M H_2SO_4 - 0.01 M HNO_3. Susceptible 18-8 stainless steels are easily detected by the method. However, the sensitivity of the test is less than that of the Huey test by a factor of about five.

Non-Metallic Materials

Cerac 4 (20 AlN:1 SiC), a ceramic product of Allis-Chalmers, proved to be quite porous to molten salt at 800 C.

One nitrile and eight silicone rubbers were tested in boiling 10 percent nitric acid and in several typical separations plants liquids at room temperature. None of the materials will be useful in the boiling acid, but five would serve well in 60 percent nitric acid at room temperature. None of the nine appear to have any usefulness in Purex HAX, Recuplex CAX, carbon tetrachloride, hexone or 25 percent DBBP - 75 percent CCl_4. All of the silicones would do well in nitrobenzene service. These elastomers were chosen from those which have been tested in radiation fields by Reactor and Fuels Laboratory personnel.

PROCESS CONTROL

Instrumentation for New Plutonium Reclamation Facility

In the development of the Gradient Control System for the Plutonium Reclamation Facility, information is needed concerning the relationship between measured and actual values of the column static head. A bonded strain gauge flow transducer was tested in the laboratory to determine its suitability for dynamic measurements of pulser hydraulic velocity, acceleration and pressure. The transducer responds to the square of the liquid velocity; its output signal varies from 0.8 mv at the low end of the scale to about 8 mv at full scale. A log-log plot of signal vs. measured flow rate yielded a straight line which matched the vendor-supplied calibration curve within the

test measurement accuracy of ± 0.5 percent of full scale. A study is now underway to determine the effects of pulse characteristics on static head measurement.

Calorimeter Studies

Laboratory investigations were continued in the development of methods to measure heat generation rates in fission product streams. A technique was devised using a bridge circuit containing one thermistor in direct contact with the solution and another thermistor surrounded by a heat sink. The temperature difference between the two thermistors is directly related to the heat generation rate of the solution. Heat rates as low as 0.4 watts per liter were measured by this method. A disadvantage of this technique is the fact that four to ten minutes were required to reach a condition of constant temperature difference between the thermistors. Preliminary investigation was made of the potential for using a barium titanate pyroelectric thermometer, which has capability of detecting temperature changes of 2×10^{-7} C.

A-Column Optimum Temperature Profile

The rate constant for the extraction of UNH from an aqueous phase to an organic phase increases with temperature according to an exponential relationship. However, the driving force decreases with increasing temperature; thus, for any triplet of aqueous uranium, organic uranium, and aqueous nitric acid concentrations there will be some temperature for which the rate of extraction will be maximum. In some cases, it may not be possible to obtain this temperature due to other limitations.

A preliminary study was undertaken to determine, in the range of concentrations normally encountered in an A type column (HA, 2D, etc.), whether there exists an optimal temperature between the normal operating limits of 25 to 70 C. It was found that for acid conditions above 1 M (HA column) the optimum temperature is above 70 C; for acidity less than 0.2 M and uranium concentrations below 0.15 M the optimum temperature is below 25 C. In the transition region of 0.15 to 1.0 M acid (which is normal for the 2D column) the optimum temperature lies between 25 and 70 C for uranium concentrations below 0.15 M. This study is continuing toward determination of the optimum temperature profile for the 2D column for operation under various feed stream conditions. Practically, it should be possible to obtain this temperature profile by suitable adjustment of inlet stream temperatures.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Flowsheet Development - Fine-grained deposits of plutonium dioxide, which break up into small particles upon washing with dilute acids, have been prepared by electrolysis of Cl_2-O_2 sparged molten chloride salt solutions of plutonium chloride. Successful depositions have been made from a KCl-LiCl melt at 550 C, using a 60 percent O_2 - 40 percent Cl_2 sparge and from a CsCl melt at 700 C using an 80 percent O_2 - 20 percent Cl_2 sparge. No chemically precipitated plutonium dioxide was noted in either system. The electrodeposition reaction is believed to involve reduction of a plutonium(V) or (VI) intermediate (see section on Absorption Spectra of Plutonium for supporting information).

Plutonium dioxide which exhibits well-defined crystal faces and sharp edges under 100 power magnification has been prepared by precipitation from molten CsCl-3 w/o Pu with an oxygen sparge. A sieve analysis of this material showed the following particle size distribution of the washed and dried precipitate: +65 mesh, 12 percent; -65 +100 mesh, 27.5 percent; -100 +200 mesh, 26.2 percent; -200 mesh, 34.3 percent.

Precipitation Studies with Cerium as a Stand-In for Plutonium - Studies of the precipitation of cerium dioxide as a stand-in for plutonium dioxide from a 2.5 LiCl-KCl fused salt mixture were continued in an attempt to find a workable means of recovering the precipitate from the fused salt. The cascading filter apparatus used in conjunction with a gas-lift (HW-76315 C) failed after three runs because of cracks apparently caused by attack on the quartz by the Cl_2-O_2 gas-fused salt mixture. Bonding of the CeO_2 precipitate to the quartz may also have promoted weakening of the quartz.

Direct precipitation of CeO_2 on the bottom of the vessel by sparging followed by gas lifting of the salt through a filter to recover the precipitate was attempted. Since recovery of the precipitated CeO_2 was only seven percent in the cascading filter assembly using the sparge-recovery technique, the approach was abandoned.

The tenacious adherence of CeO_2 and PuO_2 to the quartz suggested the use of a high porosity filter consisting of a quartz Gooch crucible filled with large quartz chips to pick up the CeO_2. With this apparatus a 60 percent O_2 - 40 percent Cl_2 sparge precipitated

90 percent of the cerium, and about 70 percent of that precipitate was recovered on the filter and gas lift. The CeO_2 was removed from the quartz by washing in 8 M HNO_3 - 9.2 M HF solution. A 6 M NaOH solution was also found to be effective for removal. The success of these methods indicates that the CeO_2 is removed by attack on the quartz.

A sieve analysis of the CeO_2 product yielded 10 percent +100 mesh and 56 percent -100 +200 mesh particles. X-ray analysis of the CeO_2 product has shown that the precipitate from a 2.5 LiCl-KCl mixture containing 10 w/o U and 0.63 w/o Ce is comprised of 91 percent CeO_2 and 9 percent UO_2 as a solid solution with diffraction lines of CeO_2.

Crushing of Electrolytic UO_2 - To prepare a feed suitable for vibratory compaction, electrolytic UO_2 which has been removed from the electrode in large chunks must be washed free of salt and crushed in remotely operable equipment. The present goal in crushing experiments is to produce UO_2 which has the approximate composition 55 percent -4 +10 mesh, 20 percent -35 +65 mesh, and 25 percent -220 mesh. This size distribution is optimized for vibratory compaction fuel fabrication. The major difficulty is in obtaining a high yield of the largest particles in a narrow size range. Experiments with a 6-inch Iler disc pulverizer have produced a maximum of 40 percent of the product in the -4 +10 mesh range although excellent control of the location of the peak in the particle size distribution was obtained. Attempts at crushing the UO_2 in several stages with gradual reduction of the particle diameter at which the size distribution peak occurred and with recycle of only the +4 mesh material resulted in a maximum of 50 percent -4 +10 mesh particles.

A zero-impact device, consisting of several parallel wedges mounted on the same hydraulic press which was used previously (HW-76315 C) to remove UO_2 from the electrode, was operated in several crushing experiments. Since this apparatus would probably be employed without dust containment, the fines produced are of major concern. Operation with 5/8-inch-spaced, 5/8-inch-deep wedges reduced the UO_2 to pieces passing through a one-inch-square screen, but with only 6.7 percent -4 mesh particles. Successive treatment of the +4 screen fraction from this operation with 5/16-inch wedges and 5/32-inch wedges produced 54 percent -4 +10 mesh particles, and 10 percent -35 +65 mesh particles. More of the latter fraction may be produced by further treatment of the -10 +35 mesh fraction.

Experiments with a high speed paint mixer containing steel balls have shown that even the largest particle size feed is easily reduced to -200 mesh fines in 15 minutes in a closed container.

Density of Electrolytic UO_2 - Mercury porosimeter measurements on samples from four 25-pound electrolytic UO_2 deposits produced in equimolar LiCl-KCl molten salt gave an average density of 10.90 g/cm^3 (excluding open pores greater than 9 microns diameter and a closed porosity of 0.3 percent including open pores less than 0.06 microns diameter). Similar measurements on five UO_2 deposits produced in 2.5 LiCl-KCl gave 10.81 g/cm^3 and 0.6 percent closed porosity. The products from both salt baths meet the tentative ASA nuclear grade UO_2 specifications of 10.80 g/cm^3 at 9 micron pore diameter. If the closed pores were completely filled with salt, the chloride contents would be about 500 and 1000 ppm for the UO_2 from LiCl-KCl and 2.5 LiCl-KCl salt systems, respectively. Washing experiments have resulted in much lower chloride contents, indicating that a large fraction of the "closed" pores are connected to the surface by narrow passages.

Absorption Spectra of Plutonium in Molten Chloride Salt Solutions - Spectrophotometric investigations of the chemistry of plutonium in molten chloride solutions have begun. The solvent employed in experiments to date has been the LiCl-KCl eutectic (1.4 LiCl-KCl). In the temperature range of 400 to 800 C, spectra were obtained of solutions containing only plutonium(III) and of mixed plutonium(III)-plutonium(IV) solutions prepared by saturating the solutions with chlorine (at one atmosphere). Percentages of total plutonium existing as plutonium(IV) in the chlorine saturated solutions were estimated to be approximately 80 percent at 400 C, 45 percent at 550 C, 25 percent at 700 C, and 15 percent at 800 C. The uncertainty in these values increases with temperature, being perhaps a factor of two at 800 C. Evidence for the formation of plutonium(V) and plutonium(VI) on sparging the solutions with oxygen and oxygen-chlorine mixtures was also obtained.

Determination of Chloride in UO_2 - Equipment has been set up, and calibration curves prepared, for the determination of the chloride content of UO_2 in the very low concentration range. The method, which works best at chloride concentrations less than 4 ppm in the final dilution, involves measurement of the light scattered at low angles by precipitated silver chloride. Several samples of electrolytic UO_2 were analyzed by this technique. A sample of finely divided material which had been given a prolonged washing with tap water was found to contain only 35 ppm chloride, a figure which agrees with the chloride content expected from the potassium and lithium impurity levels.

RADIOACTIVE RESIDUE FIXATION

Zeolite Properties

Synthetic faujasite, or 13-X, is a zeolite with very accessible anionic, or exchange, sites. The smallest dimension through which cations must diffuse to accomplish exchange is about 9 Å. The dimension inside the roughly spherical cavity containing the anionic sites is about 20 Å. There is, therefore, ample room for all alkali metal cations, including Na^+, K^+ and Cs^+ to reach all the anionic sites in 13-X.

However, the sodium-cesium and potassium-sodium isotherms at 25 C were found to be irregular or sigmoid in form, representing a selectivity reversal, while the potassium cesium isotherm at 25 C was found to be regular and normal in form. Apparently 13-X exhibits yet another type of cation selectivity influence. The various-sized cations are not all bonded in energetically equivalent positions on 13-X. The form of the isotherm is a function of the difference in size between the two cations. For example, the difference in the diameters of Cs^+ and Na^+ is 1.46 Å according to Ahrens, and the resulting isotherm is markedly sigmoid. The difference in the diameters of K^+ and Na^+ is 0.78 Å, and the isotherm is somewhat sigmoid in form. The diameter difference between Cs^+ and K^+ is 0.68 Å, and the isotherm is regular in form. Apparently the 0.68 Å difference in cation diameters is not sufficiently great to cause the Cs^+ and K^+ bonding positions in 13-X to be greatly energetically dissimilar. The selectivity reversal occurs as it becomes increasingly difficult for the incoming cations to find energetically favorable positions.

Condensate Treatment

The assembly and installation of the engineering scale equipment to demonstrate the decontamination of condensate streams was continued.

A small-scale study on the decontamination of steam-stripped Purex tank farm condensate was performed. The condensate with the pH adjusted to 3 by the addition of nitric acid was passed through a bed of 200 x 400 mesh Dowex 50W resin at a flow rate of about 5 gpm/ft^2 (10 cv/min). The bed, which was 5.1 cm in diameter and only 1.7 cm high, maintained a strontium decontamination factor greater than 1000 for over 5000 column volumes. The cesium decontamination factor varied from 700 at the beginning of the run to 70 at 2250 column volumes when MPC_w was reached. Pressure drop across the bed increased from 2.1 psi at the start of the run to 3.3 psi when the test was completed.

Full-Level Calcination of Radioactive Wastes

Following the High-Level Radiochemistry Facility run reported last month, the spray calciner and off-gas system were thoroughly decontaminated, and piping changes were made (remotely) in the sampling headers to prevent recurrence of the cross contamination of samples which rendered the results of Run 11 largely meaningless. Principal conclusion from Run 11 was that use of the special Phillips silica gel in the ruthenium absorber did not significantly improve the performance of this unit, which still yielded a ruthenium DF of only about five.

Following wash-down of the spray calciner, it was found impossible to obtain a vacuum in the system, and the cause was finally traced (by means of a smoke bomb) to a leak in the calciner itself. Although the leak is hidden from view (by the thermal insulating cans), its area was deduced by pressure drop measurements to be about 0.025 in^2. Origin of the leak is unknown and might reasonably have resulted from (1) the large concentration of excess sulfate present in the first runs, (2) sulfide attack on the Inconel during the run with sugar addition (a phenomenon recently reported by Gauvin and co-workers in Canada), or (3) faulty construction, possibly a cracked weld which was plugged with calcine prior to wash-down. To minimize the pressure drop across the off-gas system, and the resulting pressurization of the column, the off-gas sampling system was modified -- from passage of all of the off gas through a filter sampler (which caused a pressure restriction) to passage for a longer period of time of only a metered fraction.

Following the off-gas sampler changes, a spray calcination run (Run 12) was made with alkaline (vice acidic) feed and with reduced feed rate and elimination of blowback (to avoid column pressurization). Sodium hydroxide was added to the Purex 1WW to a terminal pH > 10, sugar equivalent to 200 grams per liter of initial 1WW was also added to convert sodium nitrate to sodium carbonate, and the resultant slurry (diluted about two-fold with respect to acidic 1WW) was pumped to the column at a rate of about two liters of (dilute) feed per hour over a period of six hours. No difficulty was experienced with pumping the slurry or with plugging of the nozzle. The column was maintained at near zero pressure (relative to the cell) by withdrawing off-gas at maximum rate, and very little activity was lost to the cell ventilation air during the run. Ruthenium volatilization was markedly reduced by alkaline operation -- to less than 0.1 percent (analytical detection limit of the in-cell gamma analyzer),

compared with about 20 percent volatilization with acidic 1WW and 3 percent for acidic 1WW plus sugar. The melt pot contained one liter of solidified melt with a density of 3.07 g/cc, equivalent to a volume reduction factor of five from the original waste. Other analytical results are not yet available.

Significance of the low ruthenium loss with alkaline waste (which verifies laboratory predictions) is that it may prove profitable to neutralize Purex-type waste prior to calcination to virtually eliminate ruthenium volatilization and the need for condensate recycle. Such neutralization would, of course, be preceded by destruction of most of the nitric acid (by formaldehyde or sugar treatment) to minimize required caustic addition.

Eighteen-Inch Radiant-Heat Spray Calciner

During the month eight runs were made in the 18-inch-diameter by 10-foot-long radiant-heat spray calciner containing a 13-3/4-inch diameter by 7-foot-long channel recycle tube. Major findings include:

1. With a nitric acid-ferric nitrate solution as the calciner feed, less than one-half of the calcine was deposited in the pot below the reactor or removed by the cyclone. The remaining calcine was effectively removed from the off-gas, however, by the sintered stainless steel filters. Attempts to blow back the filters during calciner operation were unsuccessful with the present system. With earlier simulated Purex feed, the carry-over to the filters was only about two percent.

2. With preheating of the feed stream and steam for atomization, the calciner capacity was increased by 20 percent over that previously measured for cold feed and air atomization. No detrimental differences in other aspects of the calciner operation or in the calcine appearance were noted when steam was used for atomization instead of air. The condensate volume was 114 to 120 percent of the feed volume.

3. The calciner was successfully operated at a 15-gallon-per-hour rate with simulated Purex waste and a furance temperature as low as 650 C. A tendency for calcine build-up on the inside of the channel recycle tube was noted. The amount of build-up increased with decreasing furnace temperature.

Electrostatic Bubble Scrubber

A scrubber was designed and ordered fabricated for studying the parameters controlling particle removal. The scrubber will permit various sieve-plate configurations and electrode geometries to be selected and facilitate investigation of their effects on particle removal efficiency under various flow conditions. A particle generator was assembled and tested which produces fluorescent dye particles in the less than one micron range. The generator will be used in efficiency tests of the bubble scrubber. The Turner fluorometer was calibrated for the fluorescent dye being used.

Induction Heater Control System

In a two-hour test of 10 KC, 10 KW induction heater auxiliary work station control system, a graphite work piece was held within ± 2 C of the set point. This accuracy, ± 0.2 percent of full scale, should be adequate for fused salt and multiple zone pot calciner applications of induction heating in pilot plant and future production installations.

The auxiliary work station controller is being installed permanently on the existing 3 KC, 15 KW induction heating generator in 321A Building to allow simultaneous operation of two work stations on this generator. Although the silicon-controlled rectifier work station controller was designed for use at 10 KC, it will control at any frequency up to 10 KC. Because of its conservative power rating the control system is operable up to 20 KW on 0.7 power factor loads or up to 30 KW on unity power factor loads.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

Work on the determination of stream and path functions for three-dimensional flow through porous media continued with the obtaining of general equations having the desired properties for transient path functions. A more concise description is being sought for the flux distribution by showing the conditions for existence of necessary integrating factors. If such is possible, then direct flow quantity determinations are possible by simple subtraction as contrasted to evaluation of a double integral to get the flux. These solution methods are needed to help determine the in-place permeability distribution for ground-water flow analog simulation, and later to determine flow paths.

A solution was obtained to prove the existence of unsaturated flow beneath a surface disposal site at steady-state flow conditions. The solution accounts for the negative pressures, created under certain conditions of seepage rate and depth to the water table, necessary to have unsaturated flow. Several computer solutions of surface pond seepage problems have shown this phenomenon to exist, but mathematical description of why it occurs was not previously treated.

Water level data from wells penetrating the Prosser bed confined aquifer in the vicinity of 200-West Area were examined. Although that aquifer exhibits piezometric heads appreciably less than the elevation of the 200-West Area ground-water mound, they are 10 to 30 feet higher than the elevation of the 200-East Area ground-water mound. This indicates that the likely source of recharge of the confined aquifer is in the hills to the west, or the 200-West Area mound, although rapid recharge from the latter source is contraindicated by the significant difference in heads.

Ground-water temperature profiles from wells 699-40-1 and 699-10-E12 indicate upward movement of water from the lower aquifers penetrated by these wells. Both wells are located southeast of the 200-East Area and within one mile of the Columbia River. During the past month these wells showed low but detectable concentrations of tritium and gross beta emitters. Piezometers are to be installed in these two wells to assist in determining at what depth contamination enters the casings. This knowledge will greatly assist in delineating the hydrologic patterns and geologic structure between 200-East Area and the Columbia River.

RADIOLOGICAL AND HEALTH PHYSICS

Environmental Studies

Ten membrane filters containing particulate material from air sampled for several days in different months during 1962 were analyzed for Pu-239 and for Cs-137 as a long-lived comparator. The Pu-239 values ranged from 0.085 to 3.7 $\mu\mu c/10^6$ ft^3 with an average value of 1.16 $\mu\mu c/10^6$ ft^3 while the Cs-137 varied from 14 to 490 with an average value of 183 $\mu\mu c/10^6$ ft^3. The ratios of Pu-239 to Cs-137 differed by factors of as much as 50 which should be indicative of the type of explosion or explosive material. These Pu-239 values compare fairly well with the maximum value of 7 $\mu\mu c/10^6$ ft^3 reported by the British for the spring of 1962.

Uranium Ore Inhalation Study

Three samples of uranium ore containing from 1 to 22 percent U_3O_8 were analyzed using the Frisch grid alpha energy analyzer. The U-238 daughter chains in each case were found to be very nearly (if not completely) in equilibrium and the samples contained less than 1 percent of Th-232 or its daughters. On the basis of these measurements a pitchblende ore containing 22 percent U_3O_8 was selected for the ore inhalation study.

Radiation Chemistry

The relative rate constants for reaction of hydroxyl radicals with galactose and agarose (a linear polymer of alternate galactose and 3,6-anhydrogalactose units) were measured over a wide range of concentrations by the dye-competition technique. In solution the rate of reaction between hydroxyl radicals and agarose per unit galactose residue is only 16 percent for monomeric galactose. In gel-forming concentrations (about 0.1 percent and greater), the agarose reaction rate is further reduced. These data suggest that portions of the agarose chain are increasingly inaccessible to the free radicals as the concentration is increased or a gel structure is formed. Thus, agarose appears to have a tightly packed structure whose interior is impervious to hydroxyl radicals.

Measurements of the electron spin resonance from dried irradiated radish seeds held at 50 C since irradiation has now been made for one month. Even at this relatively high temperature the resonance is diminishing only slowly with loss of about one-half in 17 days. This indication of long lives for the free radicals enhances the prospect of significant protection by chemical protective agents.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition in Conduits

Data obtained in the deposition studies to date were studied to determine the sources of error and source contribution to the overall errors. Fluorometric determinations of particle concentrations in the range used were found to be reproducible within a maximum standard error of 1.2 percent. Most determinations showed a standard error of less than one percent. Non-reproducibility in deposition measurements much larger than 1 to 2 percent must be ascribed to other sources of error which too are being evaluated.

Since predicted deposition depends upon the friction factor in the conduit, the actual friction factors are being computed from pressure drop measurements taken during the experiments. This provides an opportunity to cross-check literature values for friction in smooth conduits passing air at various flows. Initial results show that significant differences may exist. Possible explanations for the friction factor variances are being pursued.

ISOTOPES DEVELOPMENT - 08 PROGRAM

Strontium and Cerium Extraction with D2EHPA

Design of a fission product recovery, purification, and heat source fabrication plant is the subject of a current engineering study. This plant is to be coupled to the waste extraction process; due to recovery needs certain minor adjustments to the waste extraction processes are required. For example, recovery of significant quantities of cerium and promethium, not necessary for waste management reasons, are required. Thus, improvements are being sought to improve cerium and promethium extraction kinetics.

The use of the mixed complexing agent was suggested by work which indicated that chromium ion (the primary contributor to poor cerium extraction kinetics) is complexed better by tartrate ion than by citrate ion. The reverse is true of iron; in fact, appreciable iron could be extracted out of a tartrate-complexed feed. Crouse, et al., at ORNL (ORNL-TM-265) and Bray at Hanford have shown that rare earth extraction is appreciably faster out of tartrate-complexed feeds than out of citrate feeds. Thus, it seemed reasonable to assume that addition of sufficient tartrate ion to complex only the chromium ion might result in improved cerium extraction kinetics (citrate ion being supplied as the remainder of the necessary complexing agent to present iron extraction). This effect seems to be borne out by the runs described below.

The addition of 0.03 M tartrate to a citrate-complexed feed resulted in decreasing the cerium waste loss about two-fold in the pilot plant pulse column. With the D2EHPA extraction the cerium waste loss prior to tartrate addition was 23 to 27 percent at 35 to 40 C. After this base point was established, tartrate was added to the same feed and digested at 60 to 70 C for about an hour. The cerium waste loss with this feed was 8 to 12 percent, also at 35 C. The loss increased to 28 percent at ambient temperature, compared with 50 percent with the citrate feed.

Packaging Plant Study

Considerable effort was devoted during the month to completing and documenting the Fission Product Production Plant chemical flowsheet. In addition, an experimental investigation has been undertaken (with Ceramic Fuels) to adapt the Dynapak process to fission product packaging and to determine its degree of range and versatility. Two compactions were made during the month to evaluate a change in can design (heavy weld bead around the top lid to eliminate edge cracking) and modified pre-compaction heating schedules. Encouraging results were obtained. Future tests will be aimed at resolving or demonstrating the following:

1. Feasibility of a "heavy wall" can with a high integrity full penetration weld at each end and reasonably uniform wall thickness after compaction.

2. Feasibility of using two concentric "thin wall" cans for the primary container to insure high integrity containment during and after the compaction operation.

3. Solubility of $SrTiO_3$ prepared by the Dynapak process.

4. Cesium glass preparation by Dynapak methods.

5. Feasibility of Dynapaking other materials, such as CsCl, SrO, Nd_2O_3 (stand-in for Pm_2O_3), CeO_2 and Ce_2O_3.

6. Preparation of Ce_2O_3 for compaction.

E. R. Irish

for Manager
Chemical Laboratory

WH Reas:cf

[redacted]

BIOLOGY LABORATORY

A. ORGANIZATION AND PERSONNEL

No changes of major significance occurred.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

No difference in mortality or growth at 40 days of test is apparent in young chinook salmon currently being monitored for comparative effect of the corrosion inhibitor, Quachrome Glucosate, and dichromate, in 3 and 6% reactor effluent.

The toxicity of the reactor tube cleansers, SULFAM-3 and BISULF-16, reported last month, is most likely due to reduction in pH of the raw water. The greater toxicity of SULFAM-3 over BISULF-16 can also be explained on this basis.

Columnaris

No columnaris was recovered from 40 salmon fry hatched this year and raised in river water. This indicates that the columnaris recovered from older trout, sampled at the same time, must have been retained from last summer's exposure, and not introduced from the river water.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY AND TRANSFER OF RADIOACTIVE MATERIALS

Copper, Molybdenum, and Radiation

Thirty days after 850 r X ray to rats, mortality on the stock diet was 2/5; on the 750 ppm molybdenum diet, 3/5; and on the 1000 ppm copper diet, 3/5. Leukocyte values had essentially recovered to control levels at the end of the 30-day period following X-irradiation, but erythrocyte levels were still severely depressed. Urinary copper excretion increased about 9-fold within 24 hours after animals were placed on the 1000 ppm copper diet.

Zinc

The gill filaments of trout continue to show the highest concentration of Zn^{65} at 64 and 94 days following a single oral dose of 46 μc. The average concentrations at day 94 in gill filaments, gastrointestinal tract, bone, eyes, liver, plasma, blood, testes, and muscle were: 0.11, 0.098, 0.039, 0.024, 0.009, 0.005, 0.004, 0.003 and 0.002 $\mu c/g$, respectively. The body

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burden was 25 and 19% of dose for day 64 and day 94. The relatively high concentration of Zn^{65} in the gastrointestinal tract (8% of the dose at day 94) is of particular interest. During the first week following oral dosing, Zn^{65} was rapidly lost from the gastrointestinal tract, reaching a level of about 11% of dose at day 8; only small changes have been subsequently observed.

Strontium

Sr^{85} titanate was fed to a young miniature swine to determine the gastrointestinal absorption of this form of strontium. Approximately 0.3% of the administered dose was deposited in skeleton and retained at 7 days post-administration. The uptake and retention of Sr administered orally as a chloride in animals of similar age is ~5-10%.

Severe bone changes were observed radiographically in a one-year-old miniature pig exposed to high levels of Sr^{90} early in life. Exposure in utero and during the suckling period was via its dam which was ingesting 625 μc Sr^{90}/day. From six weeks to three months of age, it received 156 μc Sr^{90} daily. Several of the long bones were deformed due to alteration of the growth of the bone at the epiphysis. Severe hematological changes were evident early in the animal's life, but in recent months values were normal.

Strontium and Calcium

The ratio of Sr^{85} to Ca^{45} absorbed by rat liver slices incubated at 37 C in solutions containing these isotopes was essentially one, in the absence of added calcium. This ratio was decreased to 0.64 in the presence of 5 mM calcium. When the incubation was carried out at 4 C, the ratio of Sr^{85}/Ca^{45} absorbed was 0.67 with or without added calcium. Measurements of this type may provide a useful measure of liver damage.

Iodine

In cows fed 5 μc I^{131} daily the I^{131} concentration in the milk is fairly constant at approximately 6.4% of the daily administered dose per liter. Thyroid content of I^{131} in the three cows is 1.5 to 3 times the daily dose. One cow showing signs of estrum exhibited a temporary increase in thyroidal I^{131} concentration. Blood samples taken three hours before, three hours after, and eight hours after I^{131} feeding showed the lowest concentration of I^{131} in the pre-feeding sample and the highest concentration in the eight-hour sample. In two cows, the blood concentration was slightly higher than that in the milk, while in the third cow it was slightly lower. (In sheep, the milk concentration may be five times the I^{131} concentration in the blood.)

In three miniature goats fed 0.15 μc of I^{131} daily for one month, the thyroid burdens are 3 to 4 times the daily feeding level. The thyroids of their newborn offspring had I^{131} burdens about 1/3 those of the dams. The I^{131} concentration in the blood of the fetuses was two to three times that in the dam's blood.

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The thyroid region of 11 young sheep were exposed to X-ray doses of from 500 to 1500 r. Five of the animals will receive two additional exposures at weekly intervals. The pathological response of the thyroid will be compared with that of six animals which received from 1.25 to 5 mc of I^{131}.

Plutonium

In further studies of the comparative toxicity of Pu^{238} and Pu^{239}, rats were injected with Pu^{239} in the presence of a large molar excess of citrate (75x) or a small molar excess of citrate (4x). Animals receiving the larger quantity of citrate survived for longer periods than those receiving the smaller amount of citrate. These findings support the thesis that the acute toxicity of Pu^{239} is markedly affected by the physical form of the administered plutonium.

An additional quantity of TTHA was obtained, and studies initiated to determine its toxicity and therapeutic effectiveness when administered orally to rats. Very preliminary results indicate a lower toxicity than previously feared, but also a lower effectiveness in plutonium removal. The TTHA still appears to be somewhat more effective, orally, than DTPA.

Five white miniature swine were injected subcutaneously with 5 μc of Pu^{239}(IV) nitrate in each foreleg. Four hours later, one pig received an intravenous injection of Na_3Ca DTPA at a level of 0.0014 g/kg body weight (approximately one-tenth the recommended human therapeutic dose), followed by three daily injections at the same level. A second pig received 100 times as much (0.14 g/kg body weight) Na_3Ca DTPA, following the same regimen. Animals were autopsied 16 days after plutonium injection. There was no effect of the DTPA treatment on the amount of plutonium found in the regional lymph nodes. Liver plutonium content of the pig which received the smaller amount of DTPA was 87% of the control; the skeleton showed no reduction in plutonium concentration. Liver plutonium content of the animal which received the larger amount of DTPA was 36% of the control, and skeletal plutonium content, 72% of the control. Reductions in liver and skeletal plutonium deposition to 9% and 32%, respectively, of control values were achieved by surgical excision performed within two hours after injection. Excision was therefore the most effective of the methods tested.

Inhalation Studies

Five dogs on the long-term plutonium study continue to show elevated respiratory rates of 80 to 120/min. One dog has lost weight, and within the past month has shown symptoms indicative of impending congestive heart failure due to chronic pulmonary congestion. The mean lymphocyte counts of the circulating blood in all of these five dogs are less than 1,000 cells/mm^3, and there is evidence of a changed blood albumen-globulin ratio.

Tests were continued to determine the optimal conditions for fatty acid synthesis in lung homogenates. Maximum synthesis occurred at concentrations of 14 μmoles of acetate and at a pH of 7.45.

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The count median diameter of uranium ore dust (pitchblende) was reduced to 0.58 μ by ball milling for three days. Radiochemical analyses by the Radiological Chemistry Operation indicate that the ores are in near equilibrium with respect to the uranium decay series. There is virtually no Th^{232}. Analyses for U^{238}, U^{234}, Th^{230}, and Ra^{226} can be carried out with 20 per cent accuracy on tissue samples containing 1 to 2 μg of ore. Samples of ore dust are being analyzed for gross alpha and gamma activity to determine the feasibility of rapid aerosol filter sample analyses and whole-body monitoring of exposed rats. The first exposure of rats to uranium ore dust will begin this month.

Tissue digestion procedures for samples containing Ru^{106} are being evaluated, preparatory to exposure of dogs to $Ru^{106}O_2$ aerosols.

Dogs exposed last summer to millicure quantities of $Ce^{144}O_2$ show no major changes except for possible increases in respiratory rates, decreased leukocyte counts, and elevated blood CO_2 levels.

The use of a skeleton dog phantom for calibration of the whole-body counter was evaluated by comparison with an intact dog containing a known quantity of Ce^{144}-Pr^{144} in the lungs. Estimates of lung burden obtained using the phantom were high by about 10 per cent. AEC laboratories working with beagles are considering the purchase of beagle phantoms similar to those which have been employed for human studies.

The following were tested in rats for removal of inhaled $Pu^{239}O_2$ without apparent success: DTPA + Pluronics (by aerosol), $NaHSO_3$ (aerosol) + KI (oral and intraperitoneally), and TR (sodium bis (tridecyl) sulfosuccinate) (by aerosol). In cooperation with Dr. Kelly of Industrial Medicine Operation, bronchial lymph nodes (which accumulate plutonium) were surgically removed from a dog. This rather drastic surgical technique is being examined to determine its feasibility, in the event that present long-term studies show harmful effects of plutonium accumulated in the bronchial lymph nodes. It is also planned to determine the effectiveness of flushing Pu^{239} from the lungs of recently exposed dogs by intratracheal administration of various fluids.

Gastrointestinal Radiation Injury

A group of rats were surgically prepared with biliary fistulae draining into plastic vessels implanted within the abdominal cavity. These animals were exposed to 1000 r abdominal X ray and injected intraduodenally with C^{14}-labeled taurocholate. This labeled bile salt was rapidly absorbed from the intestine and excreted with the bile into the plastic vessel. Daily injections and collections are being made to obtain the complete post-irradiation response of the intestine with respect to bile salt metabolism.

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Plant Studies

Treatment of barley plants with chloramphenicol caused reduced uptake of Rb^{86} and I^{131} as compared with untreated plants. By relating the amount of water and the amount of ions taken into the plants, it was shown that more I^{131} was taken into the plant than could be accounted for simply by water movement. Such active transport was not observed for Rb^{86}.

Plant uptake of three heavy metals was compared in Ephrata and Cinebar soils using Pb^{210}, W^{185}, and Pu^{239}. Ratios of concentration in dry leaves to concentrations in dry soil were:

0.0014	-	Pb^{210} in Ephrata soil
0.0038	-	Pb^{210} in Cinebar soil
0.0001	-	W^{185} in Ephrata soil
0.00001	-	W^{185} in Cinebar soil
0.00009	-	Pu^{239} in both soils

As one might predict from the insolubility of tungstic acid, W^{185} shows a lower uptake from the acid Cinebar soil.

Microbiology

Differences were noted in sensitivity of yeast strains to D_2O, and related to their manner of oxidizing substrates. Loss of capacity to carry on aerobic metabolism was correlated with an increased sensitivity to D_2O.

Phosphorus-32 was incorporated into the phospholipids of yeast cells during the time of glucose and K uptake. When glucose uptake ceased, further incorporation of P^{32} did not occur, and much of the P^{32} previously incorporated was lost.

Columbia River Limnology

Regular sampling at the Vantage and Priest Rapids stations was suspended after collection of one year's data from these locations. Analysis of plankton samples from below 100-F shows that Asterionella continues to be the dominant diatom in the river phytoplankton.

A site was selected for the collection of data pertaining to P^{32} and Zn^{65} accumulation by Columbia River plankton. Equipment is being readied for this study.

Rattlesnake Springs Limnology

Preparations were made for introduction of Rb^{86} into Rattlesnake Springs. After analysis of the data from this initial introduction, a chronic study will be initiated with the same isotope. Distribution and uptake of this nuclide will be studied in both aquatic and terrestrial biota.

Plant Ecology

On February 20, maximum soil moisture readings were obtained for the current plant growing season. Water stored in a greasewood soil profile amounted to 630 g/dm^2 as compared to 660 g/dm^2 in a sagebrush soil profile.

Fallout

Radioiodine concentrations in deer, elk, and reindeer thyroids are decreasing with an apparent half-time of 25 to 30 days. This is longer than the half-time observed a year ago because of continuing addition of radioiodine to the environment from nuclear weapons tests and/or greater precipitation in the sample areas. Maximum concentrations occurred in Montana elk, as shown below:

Location	No. animals	Nannocuries I^{131} in wet tissue	
		Per Gram	Total Thyroid
Montana elk	3	1.7	15
Maryland deer	6	0.48	2.1
California deer	2	0.28	0.44
Alaska reindeer	6	0.10	0.73

R. Thompson

BIOLOGY LABORATORY

RC Thompson:es

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TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORY

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposiums

None

b. Seminars (Off-Site and Local)

None

c. Seminars (Biology)

Van Bruggen, J. T., Professor of Biochemistry, University of Oregon Medical School, Portland, Oregon. Studies in active transport. February 5, 1963.

Cline, J. F. Uptake and transfer of radioactive strontium in the field. February 12, 1963.

Rickard, W. H. Fallout in relation to natural landscape. February 12, 1963.

Becker, W. A., Department of Poultry Science, Washington State University, Pullman, Washington. Application of multiple decision statistical procedures to biological experiments. February 15, 1963.

Brown, R. E. Wild animals I have known. February 19, 1963.

Bustad, L. K. Effects of prolonged low level irradiation of mice. February 26, 1963.

Eberhardt, L. L. Ecological sampling and analysis. February 26, 1963.

II. Articles Published

a. HW Documents

None

b. Open Literature

McClellan, R. O., J. R. McKenney, and L. K. Bustad. 1962. Changes in calcium-Sr^{90} discrimination with age in young miniature swine. Life Sciences 12: 669-675.

Erdman, H. E. 1963. Arrested development in X-rayed larvae of *Ephestia kuehniella Zeller* (Lepidoptera: Phycitinae). American Midland Naturalist 69: 34-37.

II. Articles Published

b. Open Literature (Continued)

Barber, D. A., and H. V. Koontz. 1963. Uptake of dinitrophenol and its effect on transpiration and calcium accumulation in barley seedlings. Plant Physiology 38: 60-65. 1963.

O'Brien, R. T./ Relationship between deuterium inhibition of growth and glucose catabolism in Saccharomyces cerevisiae. Proc. Soc. Exptl. Biol. and Med. 112: 186-188.

Rickard, W. H., A. D. Wiggins, and J. K. Frenstad. 1963. Caesium-137 in dried milk products in relation to phytoclimatic zones. Nature 197: 197-8.

III. Visits and Visitors

a. Visits to Hanford

2-5-63 Dr. W. Dickson, Dept. of Pharmacology, Washington State University, Pullman, Wash. Consultant - discuss research with B.J. McClanahan.

2-5-63 Dr. Murray Weiner, Geigy Labs, Yonkers, NY. Discuss use of chelating agents with Dr. Kornberg and staff.

2-6-63 Dr. F. L. Earl, Food and Drug Administration, Washington, D.C. Discuss Hanford miniature swine with Dr. Bustad and associates.

2-6-63 Dr. Van Bruggen, Vernon Walters, Neil Moir, and Paul Russell, University of Oregon Medical School, Portland, Oregon. Dr. Van Bruggen presented a seminar. They discussed research in progress and toured facilities.

2-7 & 8-63 Dr. H. D. Bruner, USAEC, Washington, D.C. Visit facilities and discuss program with Dr. Kornberg and staff.

2-11-63 Science Youth Day - facilities were toured by 37 students from various high schools in the vicinity.

2-15-63 Drs. Galgan and Dyer, Washington State University, Pullman, Washington. Visit facilities and discuss research and graduate programs with Hackett and McClanahan.

2-15-63 Jack Greene, Civil Defense, Washington, D.C. Discuss committee work in Civil Defense and research with Dr. Bustad.

2-15-63 Dr. Carl Miller, Stanford Research Institute, Palo Alto, Calif., tour facilities and discuss research with Dr. Bustad.

III. Visits and Visitors

a. Visits to Hanford (Continued)

2-15-63 Dr. W. A. Becker, Thomas Bogyo, and Surendra Sinka, Washington State University, Pullman, Washington. Dr. Becker presented a seminar. Toured facilities and discussed research with members of staff.

2-27-63 Ten Columbia Basin College students and T. W. Galbraith toured the facilities.

2-28-63 Lt. Col. R. L. Brown - USAF Staff Vet., and 1st Lt. J. G. Mercer, USAF, Larson Air Force Base. Tour facilities and discuss research with Capts. Casey and Wood.

b. Visits Off-Site

2/10-12 H. A. Kornberg attended the AEC Bio-Medical Program Directors Meeting sponsored by the University of Utah and held at Brighton, Utah.

2/13 B. O. Stuart attended a meeting at the Cancer Research Lab in Salt Lake City, Utah, to discuss requirements for construction of a "phantom dog".

2/21 W. H. Rickard and L. L. Eberhardt collected ecological samples and data at the Wooten Game Range, Blue Mountains, Washington. Members of the State Game Department were contacted.

2/27 V. G. Horstman and H. A. Ragan attended "Swine Day" at the Prosser Experiment Station, toured facilities and discussed research with staff.

IV. Achievements

No degrees were earned, nor did any professional licensing or certification occur.

V. Honors and Recognitions

None

VI. Professional Group or Organization Assignments

None

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APPLIED MATHEMATICS OPERATION

MONTHLY REPORT - FEBRUARY, 1963

ORGANIZATION AND PERSONNEL

There were no changes in organization or personnel in Applied Mathematics Operation for the month of February, 1963.

OPERATIONS RESEARCH ACTIVITIES

Work began on creating a model of the structure of HAPO as a prerequisite to further work on the Flow-of-Funds model.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Irradiation Processing Department

Ingot impurity data submitted by the feed sites are summarized using censored distribution techniques to account for data falling below minimum detection limits for the various impurities. The computational method used was audited this month because of obvious errors in the computation of some of the adjusted means and variances. It was found that the errors were due to changes in the detection limits for two of the impurities.

Two gauges were used in an experiment in which the diameters of 50 fuel elements were measured. From a plot of the data, it was apparent that a nonconstant bias may well exist between the two gauges, but this could not be determined with existing data unless the gauges were presumed to have comparable precisions. Therefore, it was recommended that the pieces be remeasured on each of the gauges, thus permitting estimation of all the parameters of interest.

Considerable work was done on various problems associated with rail heights on self-support pieces to be charged in the K reactors. Several MERCY analyses were performed involving different types of gauges and various observers to determine how well rail heights can be measured. An analysis was performed on measured rail heights of pieces which had been charged in the process tubes and then pulled out and measured to estimate the components of variance due to different sources. Analyses on "R" data, descriptive of temperature imbalance in a process tube, are being made for tubes into which fuel elements having rails of different heights were charged. Consulting assistance was provided on a number of other problems associated with rail heights.

Additional reports were written on the frequencies of ledge and groove corrosion attack on fuel elements. These were primarily aimed at relating frequencies of attack to water treatment.

A breakdown was made of Quality Certification data to differentiate between frequencies of hot spot and hot area attack. This was motivated by the need for such a breakdown in evaluating bumper performance for columns irradiated under this program.

In connection with the control of measurement biases in C-Basin, an audit program was proposed in which each of the six operators involved would run a minimum of two audit runs weekly on some piece of equipment. By treating weeks as "blocks", after each 15-week period, the data at a given station will form a balanced incomplete block arrangement. MERCY output can be so analyzed as to pick out observer and/or time differences in the equipment. A report will be issued specifying how this can be done, and what can be learned from such an arrangement.

Data from an experiment to determine the best canning conditions for KVE fuel elements with respect to UE-1 and UT-4 results are being analyzed. The experiment involves two factors, time and temperature, each at five levels. A quadratic model was fitted to the data and an optimum region of canning was determined from the resulting equation.

Data are being analyzed from production test 490 in order to determine whether there are dimensional distortion differences between fuel elements canned under the AlSi process, the Hanford Hot Press process, and the Sylcor Hot Press process. The reactor environmental effects on the fuel elements were "blocked out" by the charging patterns in the tubes.

Further assistance is being given in evaluating results from an experiment to establish optimum canning conditions with the hot die sizing process. At present the effects of time, temperature, ram speed and percentage wall reduction ratio on the clad thickness at the cap, base and middle of the fuel element are being studied.

An experiment was designed to determine the optimum conditions for putting end caps on fuel elements canned under the hot die sizing canning process. The experimental design used is an incomplete three-level factorial model, the three factors being time, temperature, and pressure.

Analysis of reactor activity data for IPD for the period April, 1961, through December, 1962, has been completed. Unfortunately, the required knowledge of how staffing is arranged for these periods does not exist in the central information system, of which outage time accounting is a portion. It will now be necessary to determine by interview techniques how each facility

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staffs craftsmen to perform the various recovery functions. IPD personnel have been authorized to implement a demonstration recording and processing system designed to associate craft expenditures with incremental reactor systems work to permit accurate delineation of man-system relationships. This work is to be restricted to the area of preventive and repair maintenance.

In the analysis of badge-pencil dosimeter data, small readings had been eliminated, primarily because of the impossibility of obtaining negative pencil readings and the resulting bias which is introduced in the estimation of the parameters. A program, called the CELERY routine, was written to permit utilizing these smaller readings. This treats pencil readings of zero for nonzero badge readings, as censored values and obtains maximum likelihood estimates of the parameters. The data are currently being processed using this routine.

A preliminary study on the operation of and the processing of warped tube data from optical boroscopes has prompted a much more thorough investigation of the sources of possible errors. In addition, work was begun on a statistical error analysis of another proposed method of measuring process tube distortion.

N-Reactor Department

Data was submitted giving estimates of compound layer continuity in N-fuels as determined by visual examination after sectioning. These were compared with pulse readings obtained prior to sectioning using ultrasonic equipment to see if the pulse readings characterized the compound layer integrity. It was suggested that additional data be collected, and that the sample pieces be measured more than once prior to sectioning.

Work continued on the N-Reactor reliability study. Success of the present approach to the study is contingent upon the satisfactory development of a computer program which will reduce Boolean expressions to the adjunctive normal form. Through consultation with N-Operations, a more articulate description of the physical relationships between the reactor sybsystem has been achieved, and steps are now being taken to translate this description into Boolean expressions suitable for computer use.

Work continued on estimating parameters for crack frequency distribution of NPR primary piping. NELLY was used on data grouped by welding method. Several cases failed to converge. Other techniques are being worked on to obtain acceptable parameters for these cases.

Work continued on the problem of statistical evaluation of longitudinal weld quality of NPR primary loop piping. One part of a summary report has been issued, and a second part of the summary report is awaiting results of digital computer calculations.

Chemical Processing Department

The investigation of the effects of gauging measurement errors on producer's and consumer's losses resulting from the acceptance inspection of parts is continuing. Data from three measurement points were analyzed on a shipment-by-shipment basis. Also, the relationships between measurements made on the parts at various steps (machining, inspection, and certification) were studied.

Due to nonhomogeneity of button impurities as determined by a comparison of top and bottom samples, the method of sampling was changed to better insure obtaining a representative sample.

Discussions were held with representatives from several groups relative to action to be taken with respect to recording the U^{235} content in shipments of UO_3 in view of temporary inaccessability to the mass spectrometer used to determine the official HAPO record.

Work continued on the refinement and programming of a mathematical model of spare parts and general inventory control.

Construction Engineering and Utilities Operation

Correlations between category averages from the recent attitude survey were computed as requested for employee groups within CE and UO, and a cover letter was prepared explaining the significance of the correlation coefficients.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HL

2000 Program

Pulse Column Facility

The analysis continued of data from five equilibrium runs on the pulse column using the BIMD power spectrum estimation program. Current efforts are directed at separating the power spectrum into two components - first, the lower frequencies which correlate positively with the frequencies observed in independent driving variables; and second, the higher frequencies which seem to monitor the pulse of the mass transfer process in the column and may prove useful in quantifying the description of flooding conditions in the column.

3000 Program

General

Coding and debugging of an extremely versatile EDPM program for obtaining numerical solutions to a simultaneous set of first order ordinary differential

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equations is virtually complete. The program has been written so as to permit the user's selection of any one of several well known and universally accepted numerical routines each of which exhibits particular virtues depending upon specific properties exhibited by the differential equations being considered. The first application of this program will be to study the NPR stack gas chemical reactions in order to learn more about the Zirconium-Graphite compatibility problem.

4000 Program

Swelling Studies

The quantitative metallography FORTRAN program for estimating volume fraction and density of a spherical phase in a three-dimensional matrix from Zeiss particle size distributions is being modified to handle large diameters, above the limits of the particle sizer, which are being visually sized by the observer. The program is also being made more flexible so that it can handle data collected from other than the shadowed negative type of replica.

Fuels Development

Theoretical particle size-proportionate mix designs were computed for annular-shaped vibrationally compactified fuel elements.

5000 Program

Actinide Element Research

Work continued on the second phase of the problem of indexing hexagonal crystals. Test data with known standard deviation were developed using a Gausian error function. From these data, studies were made of the program's estimates of the standard deviations of theta, sine squared theta, and lattice constants. Some work was done on the question of developing a meaningful lower bound on the error.

Several more test cases were run on the orthorhombic program. The debugging of this program is approaching its final stages. Some techniques developed in the hexagonal-tetragonal program are being incorporated into the logic of the orthorhombic solution.

Computation and Statistical Analysis

A meeting was held at Hanford on February 19, 20, and 21 with offsite personnel to consider the further analysis of program data. As a result of the meeting, several new data evaluations are getting underway.

Radiochemical Analysis

The use of the IRA II data analysis system was initiated during the month. With the beginning of routine use of the system, the need for certain improvements and modifications became apparent and work on defining these has already started. These improvements fall into four general areas:

1. extending the system beyond the originally planned scope;
2. reducing operating costs;
3. improving the usefulness of reports; and
4. adding refinements.

Initially, the IRA I file was used to generate the new IRA II active file. The translation was completely successful and IRA II active file and report generation mechanisms are working correctly. Card input data are now being collected which will define the recalculation using the GEM program of all existing environmental estimates. This recalculation and regeneration of the file will be done sometime during March.

6000 Program

Biology

Work was begun on the statistical analysis of data from an experiment involving neutron irradiation of Tribolium castaneum.

General

Personnel Monitoring

A study of the efficacy of compositing bioassay specimens in order to reduce the analytical work necessary to identify individuals with Pu d/m emission rates above a certain permissible level was completed. The results indicate that a 50 percent reduction in analytical work may be realized under random selection of bioassay samples which may in turn be improved upon by selective grouping of samples. A further recommendation is that the whole sampling and error structure be investigated more thoroughly.

Other

In analyzing attitude survey results, percent "favorable" is the statistic customarily used. A study is in progress to determine if other statistics are more descriptive of employee attitude, and hence, are more efficient at locating potential trouble areas.

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Studies were made to determine an optimal geometrical design for the manufacturing of shielding bricks which were to meet several requirements on their method of fabrication, their stacking stability, their versatility in use, and their ability to prevent radiation streaming.

Carl A. Bennett

Manager
APPLIED MATHEMATICS

CA Bennett:dgl

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REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

DECAY HEAT SOURCE PRODUCTION STUDY

In accordance with an AEC request, a "crash" effort is under way to calculate the possible production of Cm-244, Cm-242, and other decay heat source isotopes in current power reactor types. Specifically, the reactor types to be considered are the simulations described in HW-72217, "Fuel Cycle Analysis for Successive Plutonium Recycle - I. Results for Five Reactor Concepts."

SPWR - Simulated Advanced Pressurized Water Reactor (Based upon report TID-8502.)

SBWR - Simulated Boiling Water Reactor (A generalized description.)

SGCR - Simulated Gas-Cooled Graphite Reactor (General type patterned after "Bradwell".)

SHWR - Simulated Heavy Water Reactor (With zirconium tubes - a general type.) (In HW-72217 this reactor was referred to as "D_2O - Simulated Deuterium Oxide Moderated Reactor.")

It was decided to eliminate the OMR - Simulated Organic Moderated Reactor which was based upon report TID-8501.

To complete the study, it has been necessary to revise cycle physics and associated fuel economics codes extensively and to develop effective cross sections for the many intermediate and product isotopes involved. The Applied Physics group is developing and reporting the necessary cross sections.

The MELEAGER physics code has been revised so as to consider all isotopes of interest from Th-232 to Cm-244, inclusive. It is required to be able to assign economic values to each isotope and, also, to the total fission products. Accordingly, the MELEAGER code has been reorganized to consider isotopes which include Am-241, Cm-242, Cm-244, Np-237, Am-243, and Pu-238. Other special isotopes can be designated. An additional complication which has been taken care of is that special tallies are required if the isotope is fed into the reactor as feed as well as being concurrently generated in the reactor. Burn out of the initially charged portion of the isotope must be costed as a function of time in accordance with the currently stipulated accounting practices to be employed. On the other hand, the formation of the isotopes within the reactor is not treated as an economic asset until

the fuel is separated. This requirement necessitates defining a pseudo fueling system (as has been done in the past for plutonium fueling) that maintains the quantity of the charged isotope destroyed distinct from the quantity of this isotope that grows in. The reactor physics aspects, of course, ignore this arbitrary accounting feature. As revised, the code can handle any of the isotopes as pseudo fuels to provide complete versatility for a wide variety of fueling systems. For example, it is now possible to study plutonium enrichment of thorium or U-238 with the sale of Cm-244, as well as U-233 enrichment of thorium with sale of U-236, Np-237, and Pu-238. The MELEAGER code also required revisions in the method used to solve the differential equations representing the burn up and growth of isotopes as a function of time. These revisions were primarily concerned with setting of the error limits on the Runga-Kutta routines which are used to solve the differential equations. Generally, it was necessary to redefine the various coefficients upon which the error limits are set and to adjust these limits. If the error limits are set too close, extraordinarily long computation times are involved, which add no precision to the calculations. Error limits are set upon the amount of the relative changes in each isotope and upon the relative changes in neutron flux level associated with each Runga-Kutta step. These now appear satisfactory for the anticipated calculations, but further changes have been recognized as being desirable and will be incorporated at some future date.

The economics codes (QUICK and MINIMIZER) have been rewritten similarly to allow assignment of a price for each isotope. Thus, Pu-239, Pu-240, Pu-241, and Pu-242 can each have different prices. Similarly, U-233, U-234, U-235, U-236, Np-237, Pu-238, Am-243, Cm-244, fission products, and other special isotopes can be assigned prices. The possible combination of assigned prices can be inordinately large; and to facilitate handling the data, the Applied Mathematics group is supplying an analysis technique so that comparable values can be developed. Thus, if Cm-244 is to sell for $500 a gram, an equitable price for its precursors Am-243 and Pu-242 can be set. Furthermore, it appears that the value of Pu-242 as a source of Am-243 will influence the value of Pu-241 and finally Pu-239. The QUICK code is being arranged to produce several production coefficients such as the grams of Pu-242, Am-243, Cm-244 produced per year, per ton, and per electrical kilowatt-day. Such coefficients are valuable to ascertain the potential production of Cm-244 that can be made in the various reactors considered.

Fuel Costs of U-233-U-238 Mixtures

Plutonium is chemically separable from U-238 and thorium. U-233 is chemically separable only from thorium. The fact that plutonium is chemically separable from uranium is a special asset because it can exploit the low priced U-235 existing in uranium diffusion cascade tails. In fact,

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as previously reported, plutonium can achieve this by mixing it with any uranium containing more U-235 than that in tails. There are various ways in which U-233 (although not chemically separable from U-238) can exploit, to a degree, the fact that U-235 is available at a low price in uranium cascade tails. On one hand, U-233 can be added to tails or natural uranium; the fuel mixture can be burned; and the resulting spent fuel can be enriched in U-233 with a cascade because the separative duty costs for enriching U-233 in U-238 are sufficiently less than U-235 in U-238 -- ignoring inventories, turn around, and production level. On the other hand, U-233 can be added to tails or natural uranium; the fuel mixture can be burned; and the resulting spent fuel can be re-enriched by adding additional U-233. The least fuel value for U-233-U-238 is developed by mixing U-233 with pure U-238. The relative merits of these systems have been approximately analyzed during the month and compared with the current cycle studied using slightly enriched uranium from the cascade selling plutonium.

The merits of adding U-233 to natural or depleted uranium can be judged by computing the cost of burning such a fuel in a power reactor and comparing with the cost of burning U-235 enriched uranium from the diffusion cascade. As in plutonium recycle, the spent fuel from the U-233-U-238 cycle can either be sold as discharged from the reactor or re-enriched with additional U-233 for another burn-up cycle.

The logical price for the U-233-U-238 portions of the spent fuel should depend on the cost of re-enriching it in the diffusion cascade. That is, the price that must be assigned to the spent fuel so that, if it were used as the feed to a diffusion cascade, the cost of producing a composition equal to the initial enrichment of the reactor would equal the price actually paid. The spent U-233-U-238 also could be enriched with additional U-233 and used to fuel another cycle without cascade enrichment. Repeating this latter operation will lead, eventually, to the situation wherein purchased U-233 is used to enrich pure U-238. It is evident, therefore, that the fuel cost of this mode of operation will continually increase as the U-235 is burned out (provided U-233 is priced sufficiently higher than U-235). (Some U-235 will be bred in the U-234 generated by the U-233 captures and by U-234 from decay of U-238 but this will ordinarily be a small amount.) The fuel costs of the recycle series will be bounded by the single cycle cost on the low side (U-233 in natural uranium or tails) and the cost of U-233 in pure U-238 on the high side.

Minimized fuel costs for the following fuel loadings in the SPWR are shown in Table I: (1) slightly enriched uranium from the cascade, (2) U-233-U-238, (3) U-233-depleted uranium, and (4) U-233-natural uranium. In each of these cases, the purchase price of the U-233 was assumed to be $10 per gram and $15 per gram; and plutonium sales price was assumed to be $10.24 per gram fissile. (Note that all of the U-233 enriched fuel costs are optimistic

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because the spent fuel credit is based on an idealized price schedule that ignores inventory and turn around costs. The schedule is based on optimized tails composition and assumes that the unit separative duty cost has the same proportion to the AEC cost as the ratio of the relative difficulties in separating U-233 and U-235 from U-238 [i.e., $10.71 per kilogram].)

The data shown in Table I suggest that, for each fueling mode, there is a definite U-233 price such that the fuel cost will be identical to that of the slightly enriched uranium mode (Fuel Loading No. 1). For these data, the value of U-233 when used to enrich pure U-238 is roughly $11 per gram, and when used to enrich uranium containing U-235 is about $12 per gram.

TABLE I

FUEL COSTS FOR U-233-U-238 MIXTURES IN THE SPWR(1)

Plutonium Credit = $10.24/gram fissile
FEFJ = $40/pound
Depreciating Interest = 12.5 percent
Nondepreciating Interest = 4.75 percent

Fuel Loading	Fuel Costs, mills/kwh$_e$			
	U-233 Priced at $10/gram		U-233 Priced at $15/gram	
	Batch	Graded	Batch	Graded
1. Cascade Enriched Uranium	2.21	1.63	2.21	1.63
2. U-233 Enriched U-238	2.08	1.59	2.83	2.16
3. U-233 Enriched Depleted Uranium(2)	1.95	1.46	2.61	1.96
4. U-233 Enriched Natural Uranium(3)	1.97	1.45	2.56	1.89

(1) SPWR as described in TID-8502.

(2) 0.4 wt percent U-235 in U-238 priced at $3 per kilogram uranium.

(3) Priced at $10 per kilogram uranium.

Recycled plutonium charged in this reactor simulation is worth about $12 per gram fissile to produce discharged plutonium worth $10.24 per gram fissile.

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Flux Weighted k_{oo} for Graded Irradiation

In ideal graded irradiation computations, the value of k_{oo} at some in-reactor time is the ratio of the total number of neutrons produced since start-up to the total number of neutrons absorbed in that time. The version of the MELEAGER burn up code used in the past omits the flux when it forms this ratio. This version of the MELEAGER graded k_{oo} formula was devised when graded calculations were made by maintaining a constant flux and, consequently, a variable reaction rate. This is a satisfactory method, but requires iterating which increases computation costs. Presently, it is desired to develop data for the graded irradiation from a constant fission rate -- variable flux irradiation by adjusting the time scale to conserve the product of flux and time. That is, a "graded in-reactor time" is to be calculated by multiplying the true time by the ratio of the average flux at that time to the average flux at the terminal point. This approach is advantageous, in that batch and graded data can be obtained from one burn-up calculation and that these have the same average specific power. A disadvantage is that the flux spectrum varies during the burn up, which can be eliminated for the graded cycle by iterating the calculations until the flux spectrum is appropriately constant, as was previously planned for MELEAGER. If the present formulation is used to develop the graded case directly, the contribution of the high reactivity portion of the fuel is overemphasized and, consequently, predicts exposures that are too great. A formula for the flux weighted value of k_{oo} was derived in terms of the instantaneous and average values of k_{oo} calculated by MELEAGER. This formulation will be incorporated into the PROTEUS code.

The importance of flux weighting k_{oo} for a particular reactor type is shown by Table II. The reactivity limited exposures for a range of initial enrichments in the SPWR have been calculated by flux weighted technique and compared to the exposures calculated by MELEAGER. It can be seen that the difference between these is negligible in low enrichment cases but that it increases with the degree of burn up. The same kind of result would be obtained in other reactor types, although the magnitude of the differences may vary substantially. For example, a reactor having a lower conversion ratio will, in order to maintain a constant reaction rate, have a large change in the flux during the burn up. Therefore, the error in assuming a constant flux to compute the reactivity will be large in such types.

A practical problem involved in the flux weighted calculation was that only the time average of the flux at various times was available. The difficulty arises because, unlike numerical integration wherein an initial error tends to damp out as the number of points is increased, an initial error in numerical differentiation contributes to every subsequent point. This problem was resolved in a not completely satisfactory manner by assuming that the flux average had been generated by Simpson's Integration formula. As planned, more rigorous calculations of the graded cycle will be done with ALTHAEA and other dimensional physics codes.

TABLE II

COMPARISON OF EXPOSURES CALCULATED FOR GRADED IRRADIATION IN THE SPWR(1)

Initial Enrichment, Wt. Percent	Reactivity Limited Exposure, MWD/T		Percent Difference(2)
	Flux Weighted Calculation	Calculated by MELEAGER	
1.6	4,890	4,900	0.2
2.0	14,790	15,410(3)	4.2
2.5	24,690	27,150	10.0
2.9	32,180	36,880	14.6
3.5	41,330	51,770	25.3
4.2	52,130	67,130	28.8

(1) SPWR as described in TID-8502.

(2) Based on flux weighted exposure.

(3) This exposure is representative of the cases studied and reported with MELEAGER to date for which this approximation appears satisfactory.

Computer Code Development

PROTEUS Code

Modifications of the PROTEUS code occasioned by the revised MELEAGER and QUICK codes were completed. Much of the internal logic used in the Combined Cycles work was outmoded because of these revisions and was removed to streamline the code. In addition, a heat transfer limit check was added to the code. This feature will allow a specified maximum-to-minimum heat transfer ratio in graded operation (or a maximum-to-minimum flux ratio in batch operation) to terminate the burn up before the reactivity limited lifetime is attained. The flux weighted k_{oo} calculation described above will be available as an option in the PROTEUS code when it is completely debugged.

UCOST Code

A routine to calculate the burn-up costs associated with various uranium enrichments was incorporated into the UCOST code. This calculation was coded as the BURNUP subroutine of the UCOST code. If the BURNUP option is used, the program will calculate the cost per gram of uranium destroyed of six degrees of burn up for each

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enrichment specified in the preceding price schedule calculation. Note that this calculation is made with the generalized price schedule equation so that burn-up costs can be circulated for toll separation situations and other nonoptimized cases.

A routine to calculate the enrichment corresponding to a specified price in a given price schedule was developed. This routine will be used to find the minimum priced enrichment for the burn-up calculation when a minimum price is specified.

REACTOR TECHNOLOGY DEVELOPMENT - 02 PROGRAM

In support of the analysis of the production of "clean" U-233 in Hanford reactors, the Th-230 content of thorium ores has been investigated. The expected wide variations in the Th-230 content of normal thorium obtainable from a variety of commercial sources were confirmed by the following analytical results:

Sample Source	Th-230 Content, ppm Th-232
Thorium hydroxide, Lemhi Pass, Idaho - Montana thorite deposit	0.17
Thorium nitrate, South African monazite	0.37
Thorium nitrate, Australian monazite	1.24
Thorium oxide, Canadian uranium mill by-product	83.0

These ores show a difference in Th-230:Th-232 ratio by a factor of nearly 500 between the highest and lowest values. Significant differences in the U-232 content of U-233 produced by the irradiation of these thorium materials should be observable. Test irradiations are being planned to establish the magnitude of this effect.

W. K. Woods

Manager,
Programming

WK Woods:jm

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RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF FEBRUARY 1963

A. ORGANIZATION AND PERSONNEL

Ronald A. Walters transferred from Finance and Administration to Internal Dosimetry. Ronald L. Olson transferred from Internal Dosimetry to the Chemical Processing Department. Myrtle S. Ghent was granted a medical leave of absence from External Dosimetry. Goldie S. Groom retired from the Company.

B. ACTIVITIES

Occupational Exposure Experience

There were no new plutonium deposition cases confirmed during the month. The total number of plutonium deposition cases that have occurred at Hanford is 316 of which 229 are currently employed.

Three CPD employees received detailed hand examinations with the plutonium wound counter. There was no evidence of previously undetected contaminated injury.

During the month there were five incidents involving six CPD employees and two incidents involving HL employees where possible inhalation of plutonium contaminated air required special urine sampling for plutonium analysis.

Widespread contamination involving promethium-147 occurred in the 325 Building. The spread of contamination apparently originated from analytical samples which were stored or used in rooms 303 and 309. Approximately one-third of the main floor of the 325 Building became contaminated before control of the contaminants could be established. The upper floor and basement areas were unaffected. Personnel surveys of approximately 200 employees in the building revealed twelve cases of detectable skin contamination, but generally the activity was found to be low and the areas contaminated small. The maximum skin dose did not exceed 4.5 rads. Evaluation of bioassay samples submitted by 29 employees indicated that there were 14 cases of measurable deposition of Pm^{147}. The preliminary estimate of the maximum internal deposition was < 1% of the maximum permissible body burden for Pm^{147}. Minor levels of contamination were found and removed from seven employees' homes. As a precautionary measure, the homes of five high school students, who had visited the 325 Building on Monday, February 11, as part of a science student tour of Hanford Laboratories' facilities, were also surveyed. No contamination was detected. The event was categorized as a Class A incident by the AEC according to AEC Manual Chapter 0502.

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Vehicle traffic was temporarily diverted to one side of Highway 4S between the 300 Area and the 300-N burial ground exit to permit survey of the road when fission product contamination was detected on a truck and the driver after completing a burial trip from the 327 Building. Surveys indicated that the contamination was confined to the burial ground area except for one spot of beta-gamma contamination of 40,000 c/m near the 300 Area power house. The contamination spread was not detected at the burial ground because of a failure to follow established release survey procedures. Beta-gamma contamination to 30,000 c/m was detected on the protective clothing of the truck driver.

A criticality false alarm occurred at the 308 Building as the result of power surges following a planned power outage by week-end maintenance personnel. Employees evacuated to the back dock of the building, with the exception of one power operator, until surveys established that a false alarm had occurred.

Radioactive particles with beta-gamma activities to 80,000 c/m per particle were accidentally released in the 327 Building canyon. The contaminant apparently originated from a shipping cask received from the 100-H Area which, upon receipt, was found smearably contaminated to 70,000 c/m. Surveys of all 327 Building personnel revealed no significant personal contamination.

PRTR operation resulted in minimal personnel exposure and contamination control problems. A total of 63 air samples were taken throughout the containment vessel with a maximum concentration of 4.8×10^{-6} μc T/cc (~ 1 MPC_a) found in A cell during a reactor shutdown. Dose rates ranging from 3 to 5 r/hour were observed during the charge-discharge operation of fuel elements. The PRTR radiation exposure from tritium was summarized for CY 1962. There were 422 instances when the individual dose in a single personnel dosimeter processing period exceeded 25 mrem. The total radiation exposure from tritium for PRTR operations was nearly 38 rems in CY 1962.

The removal of material from burial ground number 6, which will eventually be the location of the Fuels Recycle Pilot Plant (FRPP), was completed. The bulk of the removed material was placed in adjoining leaching trenches and the rest in 300-N burial ground. The maximum contamination encountered was beta-gamma activity of 40,000 c/m on a piece of uncovered glassware.

The annual report to the AEC showing the distribution of radiation exposures received at Hanford in CY 1962 was submitted to the AEC. The report included the required information on the three employees who received excessive doses in the Recuplex criticality accident. In preparing the report it was noted that, excluding exposures received in the Recuplex accident, the maximum penetrating dose received for the calendar year was 4.6 rems, including 1.8 r gamma radiation, by a CPD process operator. There were eight employees who received gamma doses exceeding 3 r, the operational control for calendar year gamma exposure.

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A practice criticality evacuation was conducted at the 327 Radiometallurgy Building in which all building personnel participated. Total evacuation and re-assembly were completed within 55 seconds of the alarm. All responsible personnel observed performed their prescribed tasks very satisfactorily.

Environmental Experience

Average particulate fallout concentrations in the Pacific Northwest ranged from 4.6 to 12 μμc gross beta/m^3 of air during February. The January range was 2.0 to 10 μμc gross beta/m^3 of air.

One routine aerial monitoring flight was made which followed two standard patterns in the near vicinity of the project, and one pass over the river from 100-B Area to Pasco. No abnormal conditions were observed.

The following 150 biological, produce, and food samples were obtained for radiochemical analysis:

Milk	61 samples	143 gallons
Oysters	2 samples	4 pounds
Ground Round	2 samples	4 pounds
Beef Thyroids	26 sets	
Fish	59 samples	

Sampling of 325 and 325-A stacks was begun. Analytical results were very low. An appropriation request was approved and a purchase order written to obtain a fixed filter continuous air sampler for the 325 Building.

Studies and Improvements

Successful float travel measurements were made between Vernita Ferry and the BPA Powerline Crossing, and between the Richland Ferry Crossing and the Pasco Water Plant. The latter test was made in conjunction with special river sampling at 300 Area and Pasco. Results indicate that previous estimates of flow time between the 300 Area and Pasco at river flows of about 90,000 cfs may be too low.

Additional analyses of radiological consequences of accidents in the Plutonium Recycle Critical Facility were performed. Further changes in the facility design are contemplated which will need to be included in a complete analysis.

A study was begun of the feasibility of constructing a portable thyroid counter suitable for installation in a pickup truck or similar vehicle. Such a device would have particular application to environmental surveillance programs.

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A new film rack was designed and fabricated for holding finger ring film during processing with the mechanized film developing machine. A new film blanking die was fabricated and set up ready for routine use. Identification of the new finger rings by providing a five-digit number on the plastic film covering disc proved satisfactory. A semi-automatic stamping machine was ordered for preparing appropriate identifications on the new plastic discs.

Experimental data indicates that rhodium is more sensitive for detecting slow neutrons with DuPont 508 emulsion film than cadmium, samarium or gladolinium. Table I presents the relative response of the filters when exposed to a PuBe neutron source in the Sigma pile.

TABLE I

Relative Film Response for Filters

Filter Material	Response
Rhodium	1
Samarium	0.44
Gadolinium	0.42
Cadmium	0.40

Experimental data indicates that a slow neutron dose of about 10 mrem can be detected. With the PuBe source, an optical density difference between rhodium and tin for ten mrem of thermal neutrons was found to be about 0.065 units.

Preliminary results from the low-level burst at the Sandia Pulsed Reactor Facility showed the same trend as those reported last month for the high-level burst. Results of the low-level burst are summarized in Table II.

TABLE II

Diode Dose Measurement - Reading Current 25 ma

Dose Expected (rads)	Diode Dose Measured (rads)	Ratio $\frac{\text{(Dose Expected)}}{\text{(Dose Measured)}}$
315	182	1.73
315	161	1.96
156	153	1.02
156	160	0.98
65	62	1.05
65	43	1.51
34	35	0.97
34	29	1.17

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The doses measured by the diodes are below the expected dose; however, a correction for this may be possible when the sensitivity of each individual diode is measured. The diodes from the high-level burst were read a second time about two weeks after the burst. Some damage annealing, less than five percent, had taken place.

Preparations were made to initiate a study of plutonium particle sizes in breathing air when radiation incidents occur involving potential inhalation of plutonium particles. A purchase requisition was completed to obtain special sizes of membrane filter paper for use at selected building air sample locations where it is most probable that radiation incidents will occur. It is planned to perfect techniques enabling particle size analysis for air filters that are related to specific serious radiation incidents. Only a few select filters can be analyzed because of the tedious and time-consuming efforts necessary for radioactive particle size analysis.

Improvements were made in the placement of components on the printed circuit board for the Audio Annunciator, and in the physical assembly of the instrument. Drawings were completed and checked, a purchase requisition prepared and a contract package was delivered to GE Purchasing for their processing. This procurement is covered by AR 63-HL8-180 ($4000). Quotations were requested for 35 instruments.

Two Victoreen r-meter sets (No. 1514 and No. 1515) were sent to the Victoreen Instrument Company for repair and calibration. After any necessary repair, r-meter set number 1515 is to be sent to the U. S. Bureau of Standards for certification.

Portable instrument rejection rate improved 1.4% over January and 6.8% over the 1962 year average. Several meetings were held with portable repair shop personnel and standard rejection definitions were adopted for use by both groups.

C. RELATIONS

Six suggestions were submitted by personnel of the Radiation Protection Operation during the month. Four suggestions were adopted. Four suggestions are pending evaluation.

Safety meetings were held throughout the Section during the month. Films entitled "Fire Fighting in the Nuclear Age" and "The Science of Fire" were shown. A safety bulletin entitled "Plastics May be Dangerous" was also discussed.

Radiation protection orientation lectures were presented to personnel in Metallurgy Development, office employees of the 2101-M Building, CPD maintenance group, personnel of the Non-Metal Materials Development Operation and to all of the HAPO Technical Graduates. A lecture on plant

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emergency plans and procedures was presented to members of the IPD Fuels and NRD Fuels groups. G. E. Backman presented a talk entitled "Radiation Control Philosophy" to Radiation Monitoring personnel at the Purex facility.

D. SIGNIFICANT REPORTS

HW-76525-1 - "Radiological Status of the Hanford Environs for January 1963" by R. F. Foster.

HW-76210 - - "Effects of Hanford Operations on Columbia River Temperatures - Interim Report" by J. P. Corley (Secret).

HW-76772 - - "Radiological Aspects of Hanford Land Utilization" by R. F. Foster.

HW-76851 - - "Monthly Report for February 1963 - Radiation Monitoring Operation" by A. J. Stevens.

HW-SA-2809 - "Environmental Contamination as a Factor in Nuclear Plant Siting Criteria" by E. C. Watson and C. C. Gamertsfelder.

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PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

External Exposure Above Permissible Limits	February	1963
Whole Body Penetrating	0	0
Whole Body Skin	0	0
Extremity	0	0
Hanford Pocket Dosimeters		
Dosimeters Processed	4,767	11,102
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	8,510	18,020
Results - 100-300 mrads	226	331
- 300-500 mrads	32	47
- Over 500 mrads	10	14
Lost Results	37	61
Average Dose Per Film Packet - mrad (ow)	9.13	6.26
- mr (s)	35.79	33.32
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	1,714	3,208
Results - 50-100 mrem	2	3
- 100-300 mrem	0	0
- Over 300 mrem	0	0
Lost Results	23	29
Fast Neutron		
Film Read	412	774
Results - 50-100 mrem	48	62
- 100-300 mrem	133	306
- Over 300 mrem	2	4
Lost Results	11	11
Hand Checks		
Checks Taken - Alpha	36,508	74,798
- Beta-Gamma	59,121	119,872
Skin Contamination		
Plutonium	23	58
Fission Products	43	85
Uranium	0	0
Tritium	0	0
Thorium	0	0

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Whole Body Counter

Subject	Number of Examinations 747-A WBC	1963	Mobile WBC	1963
GE Employees				
Regular	58	123		
Incident Cases	10	19		
Terminations	9	15		
New Hires	0	0		
Special Studies	16	27		
Non-Employees				
Children	0	1		
Visitors	1	2		
Environmental Studies	0	4		
	94	191		

Bioassay

Analysis	Current Reporting Limit	Results Above Reporting Limit February	1963	Samples Assayed February	1963
Plutonium	2.2×10^{-8} µc/sample	121	216	1,018	1,613
Fission Product	3.1×10^{-5} µc/sample	15	24	961	1,599
Strontium	3.1×10^{-5} µc/sample	0	0	0	0
Tritium	5.0 µc/l	164	336	236	575
Uranium	0.14 µgm/l	0	0	165	296
Special Studies		0	0	132	179

Calibrations

	Number of Units Calibrated February	1963
Portable Instruments		
CP Meter	1,031	2,114
Juno	270	551
GM	551	1,133
Other	187	406
Audits	105	212
	2,144	4,416
Personnel Meters		
Badge Film	492	1,520
Pencils	135	240
Other	176	530
	803	2,290

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	Number of Units Calibrated	
	February	1963
Miscellaneous Special Services	912	1,538
Total Number of Calibrations	3,859	8,244

AR Keene
Manager
RADIATION PROTECTION

AR Keene:CMU:ljw

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FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

An analysis of the President's Budget for FY 1964 revealed a significant reduction in Research and Development from the RLOO-AEC request for Hanford Laboratories' programs. However, the President's Budget includes $1,500,000 more for operating costs and $300,000 less for equipment than the current planning for FY 1963. Detail of the effect on AEC programs under the direction of Hanford Laboratories is shown below.

(Dollars in thousands)	RLOO-AEC Request	President's Budget	Increase (Decrease)
04 Program			
Operating Cost	$15 805	$12 885	$(2 920)
Equipment	2 100	825	(1 275)
05 Program			
Operating Cost	1 660	1 545	(115)
Equipment	143	145	2
06 Program			
Operating Cost	3 901	3 559	(342)
Equipment	293	283	(10)
08 Program			
Operating Cost	425	400	(25)
Equipment	85	45	(40)
Total			
Operating Cost	21 791	18 389	(3 402)
Equipment	2 621	1 298	(1 323)

An increase of $150,000 in 02 Program Equipment funds was authorized during the month as a result of our request for additional funds. The total Hanford Laboratories' authorization, which is now $1,800,000, is allocated to Sections as follows:

	(In thousands)
Finance and Administration:	
Facilities Engineering	
Projects (H-L)	$ 325
Equipment	74
Technical Administration	1
Physics and Instruments	148
Reactor and Fuels	496
Chemical	380
Radiation Protection	225
Test Reactor and Auxiliaries	151
Total	$1 800

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An authorization in the amount of $100,000 was received during the month to extend the period of performance of Project Whitney fabrication work to July 1, 1963. The total FY 1963 authorization is now $272,000, including $72,000 provided for plutonium scrap recovery.

Activities for which special accounting codes were established during the month are described below:

Accounting Code	Activity
.3J	Shipment of air sampling supplies to Edwards Air Force Base. Filters and related items were originally purchased for 06 Program and will be billed at our cost of approximately $7,300. Purchase orders have been issued to replace these items.
.6R	Technical assistance to Phillips Petroleum in the use of one of the Physics codes on the 7090 computer. Seventy hours have been authorized to cover this problem and other assistance, if needed, during the remainder of the fiscal year.
.6T	R. T. O'Brien - Teaching assistance to Washington State University. The sum of $1,750 was negotiated to cover salary and continuity of service for the time involved plus travel and subsistence. A billing will be made near the end of the semester.
.6V	Provide assistance to the General Electric Jet Propulsion Laboratory, Cincinnati. F. E. Bowman was requested to participate in a review of work with which he was connected prior to his transfer to HAPO. Billing will be for four days of salary, 100 percent ILE, plus travel and subsistence.
.6W	Provide analytical services to APED. Bioassay will perform analysis to determine the plutonium content in urine by Alpha Tract Counting Method. Billing will be made at a full cost rate of $7.53 per one liter specimen. Estimated total costs will not exceed $500.

Arrangements were made to perform the cost accounting and budgeting for the Visitors Center located in Richland and for HAPO tours and visits concurrent with the transfer of these responsibilities to Hanford Laboratories from Relations and Occupational Health Operation on February 11, 1963. Costs prior to March 1, 1963 will be retained in the Relations and Occupational Health Operation.

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The personnel forecast required for the Budget for FY 1963 and Revision of Budget for FY 1964 submission was prepared and transmitted to Contract Accounting Operation. Other phases of the budget preparation are proceeding on schedule. The composition of Data Processing requirements, which were determined and submitted to Contract and Accounting Operation, was:

	FY 1963	FY 1964	FY 1965
Programming	$ 98 600	$160 000	$160 000
Test Reactor & Auxiliaries	5 300	6 000	5 000
Physics and Instruments	146 600	174 000	199 000
Reactor and Fuels	47 600	39 100	39 600
Chemical	119 700	114 000	119 000
Biology	10 200	15 000	17 000
Radiation Protection	30 300	45 500	28 100
Applied Mathematics	57 000	70 000	80 000
Finance and Administration	14 600	13 000	13 000
Total H-L	$529 900	$636 600	$660 700

General Accounting

Following is the status of letters seeking AEC concurrence to certain actions not directly covered by the Prime Contract:

AT-288	Participation in Standardizing Activities ASTM Committee C-6 - Pyrolytic Materials (Nightingale)	To AEC 2-25-63
AT-	AEC Monograph on "Radiation Effects on Structural Materials" (Bush)	In Process

Work authorized by letter under Agreement AT-6 was as follows:

Subject	Date Accepted
Use of Van de Graaff Accelerator - University of Washington	2-6-63
Proposed Visit to Japan by Dr. Paul Gast	2-9-63
Assignment of Dr. Wolfgang Stoll to Hanford	2-18-63
UO_2 for Savannah River Plant	2-19-63

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Travel activity continues at a pace consistently higher than in FY 1962 as shown below:

	February	FY to Date
FY 1962	93	719
FY 1963	111	835

During the month a total of $947,908 was transferred to Plant and Equipment accounts from Work in Progress Accounts. Included in the transfer were the following projects:

CGH-857	Physical & Mechanical Properties Testing Cell - 327 Building	$344 452
CGH-858	High Level Utility Cell - 327 Building	382 035
CGH-951	A-C Column Facility - 321 Building	38 027

Hanford Laboratories' material investment at February 1, 1963 totaled $27.5 million as detailed below:

	(In thousands)
SS Material	$25 751
Reactor and Other Special Materials	1 482
Spare Parts	253 -1)
	$27 486

(1- Includes a reserve of $79,500 established at February 1, 1963.

The value of nuclear material consumed in research this fiscal year to February 1, 1963, applicable to Hanford Laboratories is $3.3 million. The following is a detail by program:

2000 Program	$ 995
3000 Program	884
4000 Program	1 423
	$3 302

Reports of results issued during the month for the physical inventories of movable cataloged equipment in the custody of Radiation Protection Operation and Reactor and Fuels Laboratory show:

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	Radiation Protection		Reactor and Fuels	
	Quantity	Value	Quantity	Value
Balance Prior to Inventory	2 085	$1 150 770	5 280	$12 034 420
Reconciled Adjustment	(19)	(8 590)	(207)	(202 543)
Adjusted Book Balance	2 066	1 142 180	5 073	11 831 877
Physical Inventory	2 051	1 135 556	5 062	11 829 511
Missing Equipment	15	$ 6 624	11	$ 2 366
% of Missing Equipment to Adjusted Book Balance	0.73%	0.58%	0.22%	0.02%

Missing equipment in both groups is attributed primarily to transfer or disposal of equipment without proper documentation. Section Managers in both cases indicated on their Missing Plant and Equipment Report that personnel would be reacquainted with the need for proper documentation.

Savannah River Operation provided a new price list for heavy water as of January 1, 1963. The price is now $19.94 per pound, including $13.71 fund and $6.23 nonfund, representing a reduction of $.05 per pound in the fund portion. Price allowance for scrap returned has also been reduced which will result in an additional charge of $10,657 to Cost based on the January 31, 1963 scrap inventory.

On March 4, 1963, 22,268 pounds of heavy water scrap material were shipped to SROO. Preparations are being started on a second shipment which will be made in mid-May. The heavy water inventory at the end of February shows a loss of 2,781 pounds amounting to $38,500. Heavy water scrap valued at approximately $600 was generated during the month.

At our request, C&AO Property Accounting made a review of H-L Building Depreciation Accounts. This review disclosed a deficit of $674,229 in the building reserve account at December 31, 1962 and confirmed that the variance in our building account resulted primarily from establishing individual units of property within building service systems where depreciation rates are generally higher. Approval was obtained from RLOO to record the adjustment in Costs - Depreciation Reserve Adjustment, which will be transferred to the AEC at close of the fiscal year.

Laboratory Storage Pool activity for the month of February 1963 is summarized below:

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Equipment	Current Month Quantity	Current Month Value	FY to Date Quantity	FY to Date Value
Beginning Balance	1 288	$757 199	1 081	$562 200
Items Received	70	70 296	906	540 394
Items Reclaimed by Custodians	(15)	(25 340)	(95)	(124 174)
Equipment Transfers	(22)	(4 168)	(153)	(51 134)
Items Disposed of by PDR			(116)	(15 809)
Items Disposed of by Excess	(246)	(77 178)	(548)	(149 403)
Adjustment				(41 265)
	1 075	$720 809-1)	1 075	$720 809

(1- Includes 114 items valued at $81,020 on loan at February 28, 1963.

During the month, 25 items valued at $5,188 were loaned and/or transferred in lieu of purchases. A total of 228 items valued at $113,548 have been redirected to useful purposes this fiscal year.

Total investment of equipment and materials in custody of the Laboratory Storage Pool at February 28, 1963 is $1.2 million, including Reactor and Special Materials valued at $156,000 and other materials valued at $237,000.

Action as indicated occurred on the following projects during the month:

New Money Authorized H-L

CAH-822	Pressurized Gas Cooled Loop Facility	$ 15 000
CAH-916	Fuels Recycle Pilot Plant	182 000
CAH-982	Addition to Radionuclide Facility	4 500
CGH-992	Additional Fuel Loading Equipment - 308 Bldg.	75 000
Miscellaneous Capital Work Orders		16 000

Construction Completion and Cost Closing Statements Issued

*CAH-901 Structural Materials Irradiation Test Equipment - ETR
CAH-927 Additions to 271-CR Bldg. - Waste Demonstration Facility
CGH-957 Small Particle Technology Laboratory

*AEM Services Only

The following contracts were processed during the month:

CA-376 P. E. Church
SA-266 Washington State University
SA-267 International Nickel Company, Inc.
SA-268 International Nickel Company, Inc.

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Contract CA-359, Elmer A. Foskett, consultation services at Hanford Visitors Center and Lease Agreement L-35, City of Richland, for the lease of space in Community House for Visitors Center, were transferred to Hanford Laboratories from Relations and Occupational Health Operation.

Personnel Accounting

The following employees will retire on March 1, 1963:

G. W. Bailey, Normal Retirement
G. S. Groom, Optional Retirement

These employees were removed from the payroll on February 21, 1963, with the remainder of the month to be considered as vacation.

Mr. G. B. Barton received two patent awards of $125 each during the month: HWIR No. 1499 - Glasses Containing Fission Product Cesium and HWIR No. 1500 - Glasses Containing Fission Product Strontium.

Number of Hanford Laboratories Employees Changes During Month	Total	Exempt	Nonexempt
Employees on payroll at beginning of month	1 637	695	942
Additions and transfers in	24	14	10
Removals and transfers out	22	11	11
Employees on payroll at end of month	1 639	698	941

Overtime Payments During Month	February	January
Exempt	$ 7 746	$ 3 678
Nonexempt	29 701	21 607
Total	$ 37 447	$ 25 285

Gross Payroll Paid During Month		
Exempt	$ 667 463	$ 658 236
Nonexempt	522 803	512 525
Total	$1 190 266	$1 170 761

Participation in Employee Benefit Plans at Month End	February		January	
	Number	Percent	Number	Percent
Pension	1 461	99.4	1 471	99.4
Insurance Plan - Personal	397		399	
- Dependent	1 235	99.8	1 236	99.8
U. S. Savings Bonds				
Stock Bonus Plan	157	43.4	158	43.4
Savings Plan	69	4.2	69	4.2
Savings and Security Plan	1 136	88.7	1 131	88.4
Good Neighbor Fund	1 178	71.7	1 184	72.0

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Insurance Claims	February		January	
Employee Benefits	Number	Amount	Number	Amount
Life Insurance	1	$16 800	0	$ 0
Weekly Sickness and Accident	10	843	8	633
Comprehensive Medical	81	4 422	93	4 983
Dependent Benefits				
Comprehensive Medical	137	11 938	159	14 621
Total	229	$34 003	260	$20 237

TECHNICAL ADMINISTRATION

Employee Relations

Twenty-two nonexempt employment requisitions were filled during February; twenty-six remain to be filled.

Tuition Refund payment applications processed totaled 41, including 16 from other HAPO departments.

Information and Presentations

During the month Technical Administration accepted, from the Relations and Occupational Health Operation, responsibility for HAPO plant tours and operation of the Visitors Center in the Richland Community House.

The "Weekly List of Additions to Technical Information Files" has been converted to an "Official Use Only" document. Classified titles previously included are being listed in a supplement issued biweekly.

Documented information flow during the month was comprised of 955 titles (9,652 copies) received at Hanford, and 64 titles (5,824 copies) sent off-site.

The annual inventory of secret research and development reports and atomic weapon data reports was completed and revealed no additional "unaccounted for" copies.

A report on "Speech and Article Clearance Activity" which occurred during CY 1962 was published and distributed.

The annual listing, "Hanford Formal Research and Development Reports--1962," was published. Redesigned, this issue contains complete bibliographic information on the "Subject Categories Section." Cross-referenced to this section are three supplementary indexes: Author, Progress Report, and Report Number.

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Professional Placement

Advanced Degree - Seven Ph.D. applicants visited HAPO for employment interviews. Two offers were extended; no acceptances occurred, and two rejections were received. Three offers are currently open.

BS/MS (Direct Placement) - Six offers were extended; four acceptances and three rejections were received. Seven offers are currently open.

BS/MS (Program) - Ninety-three offers were extended. Five acceptances and 20 rejections were received. Current open offers total 139.

Technical Graduate Program - Eight technical graduates were placed on permanent assignment. Four new members were added to the roll. Current program members total 46.

FACILITIES ENGINEERING

Projects

At month's end Facilities Engineering Operation was responsible for eight active projects having total authorized funds in the amount of $6,244,500. The total estimated cost of these projects is $7,689,000. Expenditures through January 31, 1963 were $622,000.

The following summarizes the project activity in February:

Number of authorized projects at month's end ---------- 8

Number of new projects authorized ---------- 1
CGH-992 - Additional Fuel Loading Equipment - 308 Bldg.

Projects completed ---------- 0

New projects submitted to the AEC in February ---------- 1
CAH-995 - Air Conditioning Modifications, 309 Bldg.

Projects awaiting AEC authorization ---------- 4
CGH-974 - Analog Simulation Facility
CAH-985 - Addition to the 222-U Building
CAH-986 - 300 Area Retention Waste Expansion System
CAH-995 - Air Conditioning Modifications - 309 Building

Project proposals complete or nearing completion ---------- 6
Neutron Calibration Facility - 3745-A Building
Irradiated Structural Materials Testing Facility
Atmospheric Physics Building
PRTR Storage Basin and Experimental Facilities Modifications
Heat Transfer Apparatus for Model Studies
High Temperature Lattice Testing Reactor

Other project proposals being prepared:

PRTR Decontamination and D_2O Cleanup
Plutonium Recycle Critical Facility Conversion to Light Water
300 Area Crib Waste Transport System

The current status of projects authorized or awaiting authorization is as follows:

CAH-916 - Fuels Recycle Pilot Plant - Design is complete. Total project funds amounting to $5,360,000 were authorized. The construction bid package is expected to be issued by the Commission early in March. Excavation and removal of an old burial ground fill at the construction site was performed by the forces of the CPFF Construction Services Contractor.

CAH-922 - Burst Test Facility for Irradiated Zirconium Tubes - Design is complete. Revisions to drawings and specifications prior to preparation of the bid package have been completed. The CPFF Construction Services Contractor will begin excavating and other site preparation work early in March.

CAH-936 - Coolant Systems Development Laboratory - Construction is 91% complete. The contractor received the laboratory furniture on February 15, and is currently installing it. The Commission plans to extend the directive completion date to April 1, 1963.

CAH-958 - Plutonium Fuels Testing and Evaluation Laboratories - 308 Building - Design is complete. Specifications for procurement of the metallograph were given to the Commission for review.

CAH-962 - Low Level Radiochemistry Building - Commission comments on the design criteria were received January 31, 1963. The comments were incorporated and the criteria multilith masters returned to the Commission on February 6, 1963. Although the Commission has not formally approved the criteria, they have verbally indicated intent to approve it.

CGH-974 - Analog Simulation Facility - Project is not authorized and no work is being done.

CAH-977 - Facilities for Radioactive Inhalation Studies - A meeting was held with the Commission to discuss their comments on the design criteria. The revised criteria and the scope tracings were approved by the Commission on February 25, 1963.

CAH-982 - Addition to Radionuclide Facilities - 141-C Building - A work authority was issued February 4, 1963 authorizing the Company to prepare a design criteria. The criteria is approximately 5 percent complete and is temporarily delayed because of work load. The criteria completion date was extended to April 15, 1963.

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CAH-985 - Addition to the 222-U Building - Project is not authorized and no work is being done.

CAH-986 - 300 Area Retention Waste System Expansion - Project is not authorized and no work is being done.

CGH-992 - Additional Fuel Loading Equipment - 308 Building - Directive HW-549, dated February 20, 1963, assigns project management to the General Electric Company and authorizes interim funds in the amount of $75,000 for performance of design, removal of existing equipment, procurement of long lead time items not requiring design effort at this time, and performance of other work to be charged to operating expense.

CAH-995 - Air Conditioning Modifications - 309 Building - The project proposal requesting authorization of this project was transmitted to the Commission on February 19, 1963.

Engineering Services

Engineering service was provided to laboratory customers as requested. Principal accomplishments were: (1) completion of design for refurbishing laboratory 11-B, 329 Building, (2) design was completed and an air dryer order placed for Critical Mass Laboratory, (3) study of noxious fumes at 622 Building and recommended modifications, and (4) electrical power source for equipment in 3717-B and 306 Buildings.

Pressure Systems

Engineering review and inspection of pressure systems continued, and 38 pressure vessels were inspected by the Third Party Inspector. Work receiving attention during the month included: (1) design of the Irradiation Studies Loop, (2) review of bids for the pressure bonding autoclave, (3) review of waste calcination vessels, and (4) consultation on 189-D pressure piping revisions.

Plant Engineering

Consulting service was provided to maintenance and operating forces on electrical and mechanical work. Other work performed included: (1) completion of installation of the 325 Building vacuum pumps and design of water flow pressure switches, (2) filter box testing confirming that proper installation of filter obtains a tight seal, (3) completion of design for a double filter box for use at required locations, and (4) design of electrical power modifications to suit requirements.

1250434

Facilities Operation

Landlord costs for January were $206,565, which represents 105% of the forecast for the month. The total cost to date is $1,049,092, which is 88% of the predicted. During this month there was a $55,000 reduction in the budget to adjust for steam billings. On this basis, the year-to-date expenditure becomes over 95% of predicted. Improvement maintenance was $27,042. Costs in all categories exceeded the monthly forecast except for janitors, operators, other utilities and administration.

Waste disposal operations during January, compared to the previous month, are summarized below:

	December	January
Concrete barrels disposed	12	8
Loadluggers of dry waste	27	29*
Crib waste, gallons	325 000	290 000

*Two of which went to the "Wye" burial ground (PM^{147} contaminated waste from the 325 Building).

Assistance was provided to the 325 Building personnel in cleanup and control of a contaminated spill.

The "Wye" burial ground was reactivated upon approval of RLOO-AEC.

Drafting

The equivalent of 86 drawings were completed during the month for an average of 44 man-hours per drawing.

Major jobs in progress are: PRTR "As-Builts", PRTR Shim Rod Control, PRTR Cladding Cutter Assembly, Inhalation Studies Hood, Fuel Element Identification System, Salt Cycle for "C" Cell, Capillary Loading Glove Box, and Hi-Pressure Furnace.

Work performed by CE&UO and Bovay Engineers during the month was 107 and 180 man-hours, respectively, and work assigned was 24 and 60 man-hours.

Construction Supervision

Activity during the month on construction work (J. A. Jones Company) being performed for Hanford Laboratories' components is given below:

1250435

	Unexpended Balance	Waste Calcination (Job 7005)
Orders outstanding beginning of month	$454 324	$354 250
Issued during the month (inc. suppl. and adj.)	166 727	5 250
J. A. Jones expenditures during month (incl. C.O. costs)	157 361	46 132
Balance at month's end	463 690	314 368
Orders closed during month	85 169	

In addition, work on seven maintenance work orders having a total face value of $11,609 issued to plant forces was supervised.

W Sale
Manager
Finance and Administration

W Sale:JVM:whm

04 PROGRAM - REACTOR DEVELOPMENT

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output for February was 752 MWD for an experimental time efficiency of 70% and a plant efficiency of 50.5%. There were three operating periods during the month, all of which were terminated manually as a result of excessive helium and D_2O leakage.

Fuel exposure history at month-end was:

Maximum UO_2 exposure/element	2,756 MWD/TU
Average UO_2 exposure/element	1,864 MWD/TU
Maximum Pu-Al exposure/element	77.3 MWD
Average Pu-Al exposure/element	58.6 MWD
Maximum Moxtyl exposure/element	39.4 MWD (788 MWD/TU)
Average Moxtyl exposure/element	20.4 MWD (409 MWD/TU)

The core loading on February 1 consisted of 24 UO_2 elements, 24 Pu-Al elements and 37 mixed-oxide elements. The core loading at the end of the month consisted of 22 UO_2 elements, 22 Pu-Al elements and 41 mixed oxide elements. The status of the various test elements on February 28, 1963, is shown in the Table below. Those test elements which had reached their assigned goal exposure or had been permanently discharged for other reasons prior to February 1, 1963, have been deleted from this table.

PRTR Test #	Channel Location	Fuel Element Number	Description	Date Initial Charge	Date Discharged	Approximate Accumulated MWD
10	1447	1082	UO_2-Hot Swage	11/3/61	--	75.1
10	1647	1067	UO_2-Vipac	11/3/61	--	78.9
13	1853	5094	Pu-Al Physics	12/3/61	--	77.3
14	Basin	5097	Moxtyl-Swaged	4/2/62	6/4/62	19.6
14	1356	5098	Moxtyl-Vipac	5/8/62	--Repad	31.5
14	1544	5099	Moxtyl-Vipac	5/8/62	--Repad	29.9
37	1449	1096	UO_2-Physics	5/12/62	--	40.5
37	1649	1097	UO_2-Physics	5/12/62	--	40.0
37	1552	1098	UO_2-Physics	5/12/62	--	36.1
37	1548	1099	UO_2-Physics	5/12/62	--	38.8
37	1651	1100	UO_2-Physics	5/12/62	--	33.4
37	1451	1101	UO_2-Physics	5/12/62	2/24/63	39.6

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PRTR Test #	Channel Location	Fuel Element Number	Description	Date Initial Charge	Date Discharged	Approximate Accumulated MWD
48	1243	5150	Moxtyl ($\frac{1}{2}$" x $\frac{1}{2}$" pads)	8/1/62	--	23.4
54	1948	5116	Moxtyl (clip-on pads)	5/8/62	--	33.6 (18.1 w/clip)
54	1443	5117	Moxtyl (clip-on pads)	5/8/62	--	39.4 (24.4 w/clip)
54	1459	5118	Moxtyl (clip-on pads)	5/8/62	--	35.1 (20.1 w/clip)

D_2O and helium losses for the month were 2,781 pounds and 127,600 scf, respectively. Of the indicated losses, approximately 1500 pounds resulted from transfer of D_2O from a temporary uncalibrated storage tank to shipping drums. Net indicated losses of 1200 pounds correlate reasonably well with indicated stack loss measurements.

Thirty-two Pu-Al fuel elements, with an average Pu-240 content of 23 percent for the 5.9 Kg of plutonium, were shipped to Chemical Processing.

Equipment Experience

A total of 129.5 outage hours were charged to repair work. The major contributors were 102 hours for process tube gasket replacement and 15 hours for internal valve leakage on several helium system valves. Twenty-seven tube-to-nozzle and 16 jumper-to-tube gaskets were replaced. A programmed gasket replacement activity with improved gaskets has steadily improved D_2O leakage rates at process tube joints.

Several areas of severe corrosion developed on the water lines for the chemical feed system.

The containment bellows for the export steam line developed two cracks which were repaired by welding. A leak monitoring program was initiated and a replacement, seamless bellows was ordered.

Preventive maintenance required 192 manhours or 4.2% of the total maintenance effort.

Improvement Work Status (Significant Items)

Work Completed:

Keithley Power Plug Modification
Primary Pump Control Circuits

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Work Partially Completed:

Enlarge Chemical Feed System
Reactor Core Level Indicator
Primary Loop Drain and Flush Valve Modifications
Interlock Between C-D Machine Shroud Seat and Discharge Hoist
Shim Rod Connector Modifications
Recording Ammeter for Primary Pumps

Design Work Completed:

Pressurizer Vent Valve Relocation to Minimize D_2O Loss
Compressed Air Supply Modifications
384 Emergency Power to Primary Pumps
Control Room Criticality Alarm
Low Pressure Light Water Injection Control Valve
Actuator with Hydraulic Snubber for Bottom Blowdon Line
Shim Well Rotameter Modification

Design Work Partially Completed:

Additional Fuel Storage and Examination
Install Vibration Snubbers for Earthquake Protection
Effluent Activity Monitor Replacement

Process Engineering and Reactor Physics

Four flux measuring wires were charged in a monitoring tube and discharged following one operating period. Evaluation was not completed at month-end.

A combined reactor power and reactor poisoning code, POWPOI, was compiled and checked out.

Eight process tubes were examined. No significant damage was observed although there were many shallow marks of approximately 0.001-inch observed in areas where wire wraps could hit the process tube.

Studies of the effects of a DC power failure were completed during the month. The results of the DC power failure study showed that necessary manual actions can be taken to compensate for the inoperability of the electrical circuits.

Procedures

Revised Operating Procedures Issued	5
Revised Operating Standards Issued	7
Temporary Deviations to Operating Standards Issued	5
Revised Process Specifications accepted for use	0
Maintenance Manuals and Procedures Issued	2

Drawing As-Built Status:	February	Total
Approved for as-built	26	910
Ready for approval		6
In drafting		25
Voided		79
		1 020
Scheduled for review		401
		1 421

Personnel Training	
Qualification Subjects	223 manhours
Specifications, Standards, Procedures	123
Fueling Vehicle	0
Maintenance Procedures	61
FEEF	4
	411 manhours

Status of Qualified Personnel at Month-end	
Qualified Reactor Engineers	9
Qualified Technicians	6
Qualified Technologists	16
Provisionally Qualified Technologists	1

Plutonium Recycle Critical Facility

The planned loading of 25 UO_2 elements was completed, and the desired data was generated. These initial loading steps proceeded very smoothly and yielded expected results with no problems of any significance.

A BF_3 chamber failed, requiring replacement and the in-limit switch of one safety rod failed.

Twenty-seven manhours were devoted to training and all PRCFO personnel were qualified to operate the facility.

1250440

Fuel Element Rupture Test Facility

Project Status (CAH-867)

The mechanical design test was 50% complete. The total containment portion of the design tests was performed. The pressure transient and movement of piping were measured and found to be satisfactory. Operation of the system under automatic pH control with both single pass and recirculation was conducted. The system work satisfactorily except that minor pH measuring system revisions will be required.

Work orders were issued to maintenance for installation of a mockup test section in B Cell, a sampling system, and a leakage collection system in B Cell. Work on the sampling system was started and work on the other two items awaits material.

Work status by construction forces is as follows:

1. Shielding - 15% complete
2. Revised high pressure and low pressure relief systems - 25% complete
3. Emergency depressurizing valve 75% complete
4. Install shutoff valve in inlet piping - 25% complete

Operation

Operating personnel participated in design tests to gain system familiarity. Hours devoted to training were 148.

CAS COOLED POWER REACTOR PROGRAM

Pressurized Gas-Cooled Loop Facility (Project CAH-822)

Total funds authorized the General Electric Company were increased to a total project authorization of $1,170,000 by AEC Work Authority CAH-822 (7), dated February 14, 1963. The project is 94 percent complete versus 94 percent scheduled as of the first of March.

Insulation is now fully complete for the performance of design tests.

Following a thrust bearing failure on the number 2 unit being tested by Bristol Siddeley, both blowers have been rebuilt and are now undergoing final performance tests. According to the latest information the number 3 unit has run at conditions of 10,000 rpm 465 psia inlet pressure and temperatures up to 350°C with carbon dioxide. These conditions were not all achieved concurrently but do represent a marked improvement from past test results.

1250641

Design test status is as follows:

No. 1, completed and issued.
No. 2, completed except for adjustment of helium loop pilot valves.
No. 3, completed except for correcting loss of prime on diaphragm compressor.
No. 4, preparations for this test are in progress.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 21,565 hours. This includes 15,353 hours performed in the Technical Shops, 5,109 hours assigned to Minor Construction, 780 hours assigned to off-site vendors, and 323 hours to other project shops. Overtime hours worked during the month was 8.1 percent (1,447.9) of the total available hours.

Distribution of time was as follows:

	Man-Hours	% of Total
N-Reactor Department	3,378	15.20
Irradiation Processing Department	3,853	17.87
Chemical Processing Department	827	3.83
Hanford Laboratories	13,607	63.10
Construction Engineering and Utilities	0	0

Requests for emergency service increased requiring an overtime ratio of 8.1% versus 5.6% for the previous month.

LABORATORY MAINTENANCE OPERATION

Total productive time realized was 20,673 hours of a possible 22,392 hours potentially available. Of the total productive time realized, 87.6% was expended in support of Hanford Laboratories components with the remaining 12.4% directed toward providing service for other HAPO organizations. Overtime worked during the month was 2.7% of the total available hours.

Manpower utilization for February is summarized as follows:

A.	Shop Work		2,574 hours
B.	Maintenance		8,954 hours
	1. Preventive Maintenance	1,800 hours	
	2. Emergency or Unscheduled Maintenance	2,300 hours	
	3. Normal Scheduled Maintenance	4,700 hours	
	4. Overtime	600 hours	
C.	R&D Assistance		9,145 hours

WD Richmond:bk
March 8, 1963

1250442

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
J. O. McPartland, L. A. Pember, and J. A. Christensen	HWIR-1597, Direct Energy Conversion
C. L. Kinney	HWIR-1603, Description of a Tube Movement Follower Eddy Current Test Coil Mount
W. L. Bunch	HWIR-1604, A Novel Concept for a Nuclear Reactor Startup Instrument
G. Jansen, Jr. and J. H. Kleinpeter	A method for securing the separation of electrolytically deposited uranium dioxide from graphite electrodes by utilizing a non-porous pyrolytic graphite coating.
K. L. Chubb	A patent disclosure was filed during February on a parity error detector designed by K. L. Chubb

Manager, Hanford Laboratories

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