DECLASSIFIED
[redacted]

HW--78758

DE93 007074

HANFORD LABORATORIES MONTHLY ACTIVITIES REPORT

AUGUST, 1963

SEPTEMBER 16, 1963

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

HANFORD ATOMIC PRODUCTS OPERATION

RICHLAND, WASHINGTON

GENERAL ELECTRIC

DISTRIBUTION RESTRICTED TO U.S. ONLY

MASTER

DECLASSIFIED
[redacted]

A.E.C. - G.E. - RICHLAND, WASH

DECLASSIFIED

[redacted]

HW-78758

This document consists of 198 pages. [redacted]

HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT
AUGUST 1963

DECLASSIFIED

By Authority of CC-PR-2

RDI ITEN 8-25-92

By TL Phillips 16 23-92

Compiled by

Section Managers

September 16, 1963

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

[redacted]

[redacted]

PRELIMINARY REPORT

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

Route To:	P. R. No.	Location	Route Date	Signature and Date

GROUP 1

Excluded from automatic downgrading and declassification

[redacted]

DECLASSIFIED

DISTRIBUTION

Copy Number	
1	W. E. Johnson
2	H. M. Parker
3	F. W. Albaugh
4	C. A. Bennett
5	F. E. Crever, GEAPD
6	L. H. Dearnley
7	W. E. Foust
8	P. F. Gast
9	A. R. Keene
10	H. A. Kornberg
11	M. Lewis
12	R. S. Paul
13	W. H. Reas
14	W. D. Richmond
15	W. Sale
16	H. P. Shaw - W. S. Frank
17 - 20	Atomic Energy Commission, Richland Operations Office Attn: J. E. Travis
21 - 23	F. P. Baranowski, Director, Division of Production Washington 25, D. C.
24	Savannah River Operations Office, Aiken, South Carolina
25	300 Files
26	Record Center

TABLE OF CONTENTS

	Page
Force Report and Personnel Status Changes	iv
General Summary Manager, H. M. Parker	v through xxvii
Reactor and Fuels Laboratory Manager, F. W. Albaugh	A-1 through A-57
Physics and Instruments Laboratory Manager, R. S. Paul	B-1 through B-34
Chemical Laboratory Manager, W. H. Reas	C-1 through C-22
Biology Laboratory Manager, H. A. Kornberg	D-1 through D-9
Applied Mathematics Operation Manager, C. A. Bennett	E-1 through E-5
Programming Operation Acting Manager, M. Lewis	F-1 through F-13
Radiation Protection Operation Manager, A. R. Keene	G-1 through G-11
Finance and Administration Operation Manager, W. Sale	H-1 through H-12
Test Reactor and Auxiliaries Operation Manager, W. D. Richmond	I-1 through I-7
Invention Report	J-1

Table I - Hanford Laboratories Force Report

Date: August 31, 1963

	At Beginning of Month		At Close of Month		
	Exempt	Salaried	Exempt	Salaried	Total
Chemical Laboratory	149	133	147	131	278
Reactor & Fuels Laboratory	200	193	197	191	388
Physics & Instruments Laboratory	110	82	108	81	189
Biology Laboratory	45	64	44	65	109
Applied Mathematics Operation	20	5	21	5	26
Radiation Protection Operation	47	94	45	93	138
Finance & Administration Operation	150	120	154	120	274
Programming Operation	15	5	15	5	20
Test Reactor & Auxiliaries Oper.	58	308	56	307	363
General	3	4	3	4	7
TOTAL	797	1008	790	1002	1792

DECLASSIFIED

BUDGET AND COST SUMMARY

August operating costs totaled $2,800,000, an increase of $382,000 over the previous month; fiscal year-to-date costs are $5,218,000, or 16% of the $32,355,000 control budget. Hanford Laboratories' research and development costs for August compared with last month and the control budget are shown below.

(Dollars in thousands)	COST Current Month	Previous Month	To Date	Budget	% Spent
HL Programs					
02	$ 89	$ 70	$ 159	$ 1 180	13
03	42	25	67	250	27
04	1 208	1 010	2 218	13 485	16
05	133	109	242	1 456	17
06	269	254	523	3 604	15
08	15	15	30	100	30
	1 756	1 483	3 239	20 075	16
Sponsored by					
NRD	167	121	288	1 761	16
IPD	63	56	119	696	17
CPD	113	118	231	1 668	14
Total	$2 099	$1 778	$3 877	$24 200	16%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

Crush tests conducted on "N" inner and "N" outer fuel supports from fuel elements irradiated to approximately 86% "N" goal indicate that the deformation characteristics of these supports are not adversely affected by irradiation.

Postirradiation examination of an "N"-inner-size fluted fuel element that was irradiated to 2000 Mwd/ton in cold water has shown that grain boundary "tearing" is primarily responsible for the observed swelling of

the fuel. The irradiation temperature of these test elements was considerably less than the temperatures at which grain boundary tearing has been observed previously.

In the study to determine the effect of clad uniformity on strain limits for "N" fuel cladding, a series of test rods irradiated to 1800 Mwd/ton have been visually examined. Behavior of the striated cladding on these rods continues to indicate that plastic instability of the cladding can occur at low total cladding strains if the depth of the striations at the fuel cladding interface is 15% or more of the nominal cladding thickness.

Eight experimental lithium-aluminum target elements successfully completed 2 months of irradiation in KER Loop 1. Except for surface filming, after discharge these elements appeared unchanged. Irradiation of the remaining eight experimental target elements will continue.

High temperature irradiation testing of Zircaloy-2 clad metallic uranium fuel elements indicates that fuel swelling approaches a theoretical maximum at volume mean fuel temperatures of 525 C or above.

An electrically-heated test section representing a half-length column of N-Reactor fuel elements was completed and installed in the heat transfer laboratory.

Boiling burnout experiments were performed with a test section, simulating segregation of PuO_2 in mixed oxide fuel elements, which had high heat fluxes in narrow bands (0.1 in.) every inch along the test section. The heat flux in the bands was 3.25 times (up to 2.7 million Btu/hr-ft^2) the average heat flux for the test section. The test section experienced boiling burnout at approximately one-half the total power of a uniform tube, but the heat flux in the bands was approximately 60% greater than the burnout flux for a uniform tube. In every case of ten runs, the test section with the bands of high heat flux burned out at locations upstream of the outlet end.

DECLASSIFIED

DECLASSIFIED

Two tests were completed in KER-1 to evaluate NH_4OH as replacement for LiOH for pH control in N-Reactor. The first test ran for 12 days at 285 C with no evidence of off-standard conditions or unusual crudding. The second test ran for 15 days during which the purification rate was decreased to 50% of normal. The turbidity encountered during the startup period did not decrease as rapidly as in the first test.

Graphite burnout monitors from B, KW, and KE reactors showed no atypical effects.

A study of the kinetics of the reaction of CO_2 with H_2 at 840 C resulted in a rate constant of 4×10^{-2} $liters^{\frac{1}{2}} moles^{-\frac{1}{2}} sec^{-1}$.

The long term N-Reactor graphite irradiations in the GETR are continuing without incident.

A significant modification to the PRTR fuel fabrication process was made with the adoption of homogeneous UO_2-PuO_2 fuel made by high energy rate pneumatic impaction (Nupak). Nupak material for five PRTR clusters is now in process.

Improved analytical analyses revealed high concentrations of water, fluorine, and chlorine in PuO_2 powders prepared for PRTR and EBWR loadings. It was found that these impurities could be reduced to acceptable levels by a simple pyrolytic treatment; the method has been reduced to routine practice.

Three new swage-compacted UO_2-PuO_2, Mark-I fuel elements failed in PRTR shortly after startup because of catastrophic attack on the Zircaloy cladding by halide impurities in the fuel. Postirradiation examination of the Zircaloy cladding revealed small, radially oriented cladding penetrations characteristic of stress corrosion.

Compatibility tests of Zircaloy with the high impurity UO_2-PuO_2 suggest that the halogens are only released from the high temperature, central portion of the fuel.

Three nonautoclaved PRTR elements developed a smooth, oxide coating in the reactor.

A UO_2-PuO_2 mixed oxide fuel element was repaired by splicing and reconnecting a broken rod wire wrap in the PRTR water basin.

The rejuvenation fuel element, GEH-4-84, successfully completed its second cycle of irradiation in the MTR.

Fission gas release from UO_2-PuO_2 was affected significantly more by rod power than by PuO_2 concentration.

Flow tests on the fuel element mockup as well as a stress-rupture test on an end cap indicate that the fast reactor-thermal reactor exchange element is satisfactory for all operating conditions in both reactors.

Plutonium-zirconium carbides exhibit melting points ranging from 2260 to 2900 C for 10 mol% and 63 mol% PuC, respectively.

Split dies for pneumatically impacting PuO_2 cermet pellets are now flame sprayed with Al_2O_3, a very effective parting compound.

PuN-50 vol% tungsten cermets exhibited some decomposition at 1500 C. Studies were started on PuS-W, PuC-Ta, PuC-Va, and PuC-W cermets.

An irradiated specimen of PuC was discharged from the ETR at an exposure of approximately 45,000 Mwd/ton.

A UO_2 fuel capsule designed to generate 855 watts/cm^2 (2,700,000 Btu/(hr)(ft^2)) was fabricated for irradiation in the ETR. Calculations indicate that 93% of the fuel radius will be molten during irradiation; and the maximum core temperature will be 4100 C.

High purity, pneumatically impacted UO_2 was sent to Dounreay for vibrational compaction studies.

DECLASSIFIED

DECLASSIFIED

The infrared absorption spectrum of polycrystalline ThO_2 was found to be similar to that of UO_2.

Thin plates, approximately 20 mils in thickness, were pneumatically impacted using tungsten coated UO_2 spheres and tungsten powder.

Extensive cracking observed in pneumatically impacted UO_2-W cermet grids is thought to be related to the different thermal expansion coefficients of the steel mandrel and the cermet.

Two vacuum insulated, W-50 wt% UO_2 cermet specimens were irradiated for 2 and 4 hours, respectively, at surface temperatures exceeding 2100 C. Although both tungsten claddings developed cracks, preliminary analyses indicate small loss of fuel and fission gases.

Modulus of rupture and diametral compression tests were carried out on a 20 vol% UO_2-tungsten cermet produced by pneumatic impaction.

UO_2 vaporized from below the surface of a 50 W-UO_2 cermet heated to about 2500 C in an H_2-A atmosphere.

A cap-to-tube weld was made between two pieces of tantalum using magnetic force welding techniques.

Construction of a prototype irradiation capsule for irradiation at elevated temperatures is under way. The capsule will permit simultaneous irradiation of 288 single-crystal tensile X-ray specimens and a large number of polycrystalline foil specimens at four different temperatures.

Foils suitable for transmission electron microscopy have been prepared from massive single crystals of molybdenum containing 400-500 ppm carbon.

The variation of the yield stress of polycrystalline molybdenum with strain rate has been established for different levels of cold work in molybdenum tested in the unirradiated condition at room temperature. The results show

that room temperature deformation processes in molybdenum are substantially changed by cold work.

Crack arrest studies on irradiated Zircaloy pressure tubes may begin sooner than expected because of a newly developed method of introducing a starting crack.

A metallography specimen has been examined from the maximum irradiation section of a Zircaloy process tube removed after 4 years of exposure in KER Loop 2.

Corrosion testing of Zircaloy-2 (containing 410 ppm hydrogen and low hydrogen content Zircaloy-2 controls in 400 C, 1500 psi steam and 360 C water) indicate hydrogen content has little effect on corrosion or the rate of corrosion-product hydrogen absorption.

Oxygen analyses of the water from the ETR G-7 loop range between 0.07 and 0.7 ppm. These are the first data reported on dissolved oxygen content in the G-7 loop and are an important consideration in the evaluation of in-reactor Zircaloy-2 corrosion rates in 280 C, pH 9 water.

Out-of-reactor corrosion rates of abraded and etched Zircaloy-2 samples previously irradiated in the ETR G-7 loop are higher (50%) for the first 2 weeks of exposure than rates for comparable unirradiated coupons exposed simultaneously to 400 C, 1500 psig steam. There were no significant differences in corrosion between irradiated and unirradiated samples during a second 2-week exposure.

Two experimental alloys of iron with small amounts of titanium were prepared from Ferrovac-E iron and MRC zone-refined iron and evaluated by tensile testing. Neither specimen showed any trace of a yield point or the heterogeneous yielding that was observed in the unalloyed materials, thus the titanium is effective in combining with and stabilizing the interstitial impurities.

DECLASSIFIED

DECLASSIFIED

Two additional in-reactor creep tests on 20% Zircaloy-2 were started. A creep rate of 1.7×10^{-5} was found during irradiation at 320 C and 35,000 psi stress, and a rate of 2.6×10^{-5} at 365 C and 25,000 psi stress.

Notch tensile tests conducted on AISI 348 stainless steel irradiated to 2×10^{20} nvt reveals that 40% cold-worked material becomes brittle in the presence of a notch.

Irradiated tensile specimens of AISI 304 and 348 stainless in the annealed condition were tested and consistently showed a 4-fold increase in yield strength after fast neutron exposures of 1.1×10^{21} nvt.

The continuous colorimetric procedure to analyze zirconium in reactor coolant (as a measure of fretting corrosion) appears satisfactory for the 10 ppb range; work is continuing to develop a procedure suitable for the 1-5 ppb range.

Successful correlation of the production rate of displacements in a number of materials with a limiting neutron energy (E_L) has been demonstrated for several fluxes having widely different neutron energy distributions.

Long term irradiations of EGCR graphite are proceeding satisfactorily.

Samples of five nuclear graphites used in the construction of Hanford reactors have been irradiated to 16,000 Mwd/adjacent ton. Total and incremental dimensional changes are compared.

One-half percent CF_2Cl_2 in air decreased the rate of oxidation of graphite at 615 C by a factor of 2. The activation energy was approximately the same in the presence and absence of the additive.

Preliminary design has been started on the safety rod assemblies for the High Temperature Lattice Test Reactor prototype, with major engineering effort directed toward development of rod scram devices.

Physics calculations on the 300 Mw(e) Fast Supercritical Pressure Power Reactor are essentially complete. The core life is calculated to be 6 years of full power operation to reach a fuel exposure of 100,000 Mwd/ton of U + Pu. The initial critical mass is 1559 kg of plutonium, including 1059 kg of fissile plutonium (Pu^{239} + Pu^{241}).

A revised test program for the Containment Test Facility was prepared, giving greater detail on the sequence of testing, proposed equipment, and test procedures.

Irradiation of tubular Zircaloy-2 clad thorium-uranium fuel elements is continuing and will have achieved a maximum exposure of 2900 Mwd/ton at the end of the current ETR irradiation cycle.

Defect testing of Zircaloy-2 clad thorium-uranium fuels in 300 C water indicates a corrosion rate less than for unalloyed uranium tested under the same conditions, based on two initial tests.

A submicron dispersion of uranium carbide in uranium has been achieved by forming uranium shot from a carbon containing uranium melt. The uranium shot has been converted to a high density, Zircaloy-2 clad, uranium fuel rod by direct coextrusion of the uranium shot with Zircaloy-2 clad components. The dispersion of fine particles of uranium carbide will be tested for inhibiting the high temperature swelling of uranium fuels.

Single-phase and two-phase unalloyed plutonium has been fractured at a variety of temperatures in the range -196 to 165 C.

The increased precision afforded by the recently installed XRD-5 X-ray diffraction unit has permitted a more detailed evaluation of the effects of rolling on the crystalline characteristics of alpha plutonium. The previously reported development of a definite texture has been confirmed.

One hundred pounds of UO_2-1 wt% PuO_2 were pneumatically impacted in a single day.

DECLASSIFIED [redacted]

DECLASSIFIED

The new vibration compaction facilities for fabrication of EBWR fuels are essentially complete.

2. Physics and Instruments

Poison splines for N-Reactor have been evaluated in a PCTR experiment. Analysis of the data was completed this month, and an informal report on the work was issued.

The effect of enrichment on the reactor physics properties of an over-bored Hanford reactor is being measured in an exponential pile experiment.

A series of calculations of temperature coefficients for the N-Reactor lattice has been performed; results were reported to the customer. Work continued on N-Reactor utilization studies.

Review of specifications for the multichannel analyzer portion of the N-Reactor system for fuel rupture monitoring has been completed; calibration tests on the system's gamma energy spectrometer are in progress at KE test loop. The fast-scan rupture monitor has been reviewed and modifications devised to improve system performance.

Instrumentation installation for the PRTR fuel test loop is proceeding with delivery of flow sensing, recording, and controlling equipment. Rack cabinets are being readied for installation; the GM count-rate meter is completed. Acceptable bids have not been received for the gamma energy spectrometer; specifications were revised.

An analog computer technique has been developed which automatically determines optimum values of three dependent parameters. In trial tests of a simple quadratic function involving two variables, values within 5% of optimum were determined within 5 seconds.

A significant milestone was passed when personnel at the Critical Mass Laboratory brought a mixture of plutonium oxide and plastic at

criticality on August 21. The resulting data are the first which are applicable to chemical systems which are neither metals nor solutions, e.g., precipitates, polymers, wet powders, slurries, etc.

Subcritical approach measurements were made on the mixture of plutonium oxide and plastic before the criticality experiment. From the subcritical neutron multiplication experiments useful criticality data were obtained on a reflected cube, a parallelepiped (which approximates a 6-in.-diameter pipe), and a slab.

The development of a pulsed-neutron technique for application to criticality problems continued. The technique was used to estimate prompt criticality for the parallelepiped mentioned above. There was good agreement between critical mass values obtained from neutron multiplication measurements and from the pulsed-neutron measurements. Data obtained previously were analyzed to obtain the control strengths of a control rod inserted in plutonium nitrate solutions. Here again, the pulsed-neutron results compared favorably with those obtained using another technique.

Using theoretical calculations, criticality of uranium metal (1 wt% U^{235}) in tubular form was compared with the criticality of the same metal in rods. The results indicated that for diameters less than 1.2 in. OD the rod configuration always has the lower critical mass.

Nuclear consulting and advisory services continued at a high level. New specifications were issued; there was participation in the activities of hazards review committees. The safety of off-site shipments was reviewed.

Measurements of the scattering of slow neutron scattering by 95 C H_2O were completed using improved neutron energy resolution. Additional fast neutron cross section data were obtained using the positive ion accelerator.

The measurement of the reactor physics properties of PuAl-H_2O systems continued using 1/2 in. diameter rods. This month a lattice spacing

DECLASSIFIED

DECLASSIFIED

of 0.80 in. was used. The number of rods estimated for criticality was 796. This can be compared with 1378 rods needed for criticality with a lattice spacing of 0.66 in.

Startup measurements continued at the Plutonium Recycle Critical Facility. Neutron spectrum measurements were taken in the central thimble of the PRCF. The sensitivity of the PRCF to small Pu-Al samples with differing concentrations of plutonium was determined. Foil irradiations were made as part of an experiment to determine an absolute power calibration of the PRCF. Worth analysis was completed for a Pu-Al rod at the center of the PRCF.

The activity of cobalt-aluminum flux monitors (those irradiated with the fuel in the PRTR) was counted.

The modified gas equation is being evaluated for use in calculating the flux distribution in uranium, plutonium, and uranium-plutonium-fueled systems. In other neutron thermalization work, a method (producing results considerably closer to experiment than previous methods) has been developed for calculating the scattering law for water.

Analysis of the burnup data from three Pu-Al physics elements continued. A paper on the Pu-Al burnup experiment has been accepted for presentation at the American Nuclear Society Winter Meeting, November 1963, New York City.

In reactor calculation, the simpler multienergy diffusion theory representation of a thermal reactor seems to give more consistent agreement with experiment than the more detailed multienergy transport theory analysis. A recent study of round-off errors during the solution of the reactor equations on the computer has given some insight into this perplexing anomaly. An improved rounding logic has been developed which should yield a higher level of computational reliability in the transport analysis method.

Work continued on the development and checkout of the RBU system. Plans are being made for the ANS Master Cross Section Library Subcommittee meeting to be held at Hanford, September 18, 19, and 20. Hanford has proposed the use of the RBU Library for the Master Library. Extensive work continued on other reactor physics codes.

In Phoenix fuel reactor studies, analysis of the Xe^{135} and Sm^{149} poisoning effects indicates that the steady state and peak xenon poisoning is considerably less in the Phoenix cores than in comparable U^{235} cores. This would reduce the required excess reactivity of the Phoenix reactor and would improve its quick restart after scram capabilities. Analysis of the MTR as a facility for performing Phoenix fuel burnup experiments is continuing.

Work continues on the HTLTR. Calculations were made to determine the amount of gadolium required to give the HTLTR a zero temperature coefficient. For the case of UO_2 driver rods containing 10% U^{235}, it was found that 0.0045 wt% gadolium would be required. Planning is under way for a PCTR experiment to measure the doppler coefficient of fuels poisoned with gadolium.

Preliminary tests of five B^{11} neutron flux monitors have been completed with satisfactory results. A production test authorizing the final reactor exposure tests of these units in KE facilities has been written. Continuing favorable results are being obtained with laboratory evaluation of microwave methods of measuring plasma ionization. Attenuation measurements have revealed about 2 db attenuation per foot in standard 3/8-in. copper tubing waveguides. Two hundred and forty milliwatt Klystron oscillator tubes are being ordered to provide increased signal generating capacity.

Stability of recently developed comparator circuitry has proven more than adequate to make extensive eddy current examinations of N-Reactor process tubes for the presence of occluded hydrogen. Maps showing the relative electrical resistivities of 85 "normal" process tube samples have

DECLASSIFIED

DECLASSIFIED

been made and differences found on one sample have been metallurgically examined. Results of destructive examination confirm a metallurgical abnormality, although further vacuum diffusion tests will be necessary to establish the presence of hydrogen. Maps of the other samples will be compared with those that will be obtained after the samples are hydrided.

Fabrication of the prototype tester for 17-mil-wall tubing has been completed with the assembly and check-out of the electronic system, mechanical tube scanning equipment, and analog recorder readout. The system employs two ultrasonic beams to detect radial and longitudinal defects simultaneously at a rate of 18 in. per minute. One-to-three-mil notches on the tube inside and outside surfaces are readily detected. The use of accurately machined punches has been favorably evaluated as an economical way of preparing standard, artificial defects.

Using the Van de Graaff accelerator as a source of neutrons with energies ranging from 45 kev to 5 Mev, the integrated response of a lithium ion detector (with Rh^{103} and Cd foils, and a surface barrier detector with a U^{238} foil) closely approximated theoretical predictions. These results moved a step nearer the development of a composite neutron sensor capable of detecting neutron energies from thermal to about 10 Mev with an output response weighted in terms of biological effectiveness.

Further reduction of data from the I^{131} release of July 25, 1963, continues to confirm that the experimental objectives were met. Peak exposures at the centerline of the plume compared favorably with average values obtained over the same grid using zinc sulfide as the tracer material. Grass samples from plots located 200 meters' distance downwind of the source received activity depositions of 10,000 dis/min/sq ft.

The summer's whole body counting work in Alaska is almost finished. Counts are generally higher than last year, but there appears to be a variety of reasons. Urinalysis methods for Cs^{137} developed last year were confirmed and were applied to determining the average body burden in Richland.

To provide information needed to design the irradiation facility, work in support of the University of Washington spermatogenesis study was started.

Attempts were made to induce lasing action in uranium and neodymium laser rods by surrounding the rods with a P^{20} phosphor and inserting the assembly in a 3×10^6 r/hr Co^{60} radiation field. No lasing action was observed.

Nondestructive examination of N-Reactor facilities is continuing under a full, straight time effort. To date, the 4000 welded tube ends in each of ten steam generators have been statistically examined, and complete examinations of each tube within certain generators have been made to identify for repair of defect areas. Continuing tests are being made on other reactor components.

3. Chemistry

In a test completed August 2, substitution of aluminum nitrate for aluminum sulfate in the experimental water treatment plant yielded lower levels for all radioisotopes (except Cu^{64}) in the effluent water. P^{32} and As^{76} levels were both lower by a factor of about 2.

Application of a silica gel treatment to the present Purex kerosene diluent, Soltrol-170, improved its fission product decontamination performance to a level competitive with highly-purified (and considerably more expensive) n-dodecane in laboratory comparisons.

As confirmed by recently completed hot cell and Strontium Semi-works tests, clean separation of cerium from the other fission product rare earths was achieved by a simple batch solvent extraction process. Availability of this process will simplify meeting interim requirements, which may arise, for production of either Ce^{144} or Pm^{147}, and will allow stockpiling of larger quantities of promethium crude in existing B Plant tankage.

DECLASSIFIED

DECLASSIFIED

Electro-codeposition of PuO_2 and UO_2 has yielded enrichment of plutonium (relative to uranium) by factors as high as 44 and deposits with PuO_2 content as high as 36 wt%.

A spray calcination run was made with phosphate added to the feed to achieve a pseudo-glass structure in the calcine. Data thus far available indicate no apparent effect of phosphate on fission product behavior during the actual run.

Preliminary analytical results from the recent controlled iodine release experiment reveal that about 10% of the iodine (released as molecular iodine vapor) had become associated with particulate material after only a few hundred meters travel. At least 15% was found to be in a nonreactive form that was not collected by silver-coated copper turnings.

Applicability of the high-energy-rate compaction technique to both strontium oxide and cermets of strontium oxide and iron was demonstrated with the successful preparation of high density compacts of both systems.

An improved alkaline permanganate wash procedure has been developed for removal of ruthenium from extractants of trilauryl amine and Soltrol.

Zirflex decladding and nitric acid core dissolution tests were performed on sections of PRTR irradiated, Zircaloy clad UO_2 fuels. The tests of the proposed plant procedure were completely successful.

A mathematical model and computer program has been developed for heat transfer in the radiant heat spray calciner. The model predicts the measured capacity of the calciner with an error of less than 5%.

Equipment for continuously converting radiant heat spray calcined wastes to phosphate "glasses" or melts was successfully operated in the cold semiworks.

Initial samplings of Redox exhaust air upstream and downstream of a charcoal filter bed gave strong evidence that a significant fraction of the I^{131} in the main ventilation stream was present as a less reactive species than molecular iodine and would not be effectively retained on charcoal.

Linde AW-400 zeolite showed a highly favorable Cs^{137} removal from a synthetic Purex "1965 FTW." One percent breakthrough at 130 column volumes was observed.

4. Biology

Background counts in 100-F have risen to a point where it is impossible to count low-level samples. Presumably this is due to Ar^{41} from reactor effluent vapors permeating from breaks in the effluent line.

A new material, Dowell F-33 (an alkyl phenol derivative), is being used for cleaning N-Reactor pipes and is being released to the river. Our tests on fish show short-term toxicity at 10 ppm, with no toxicity observed within 24 hours at 5 ppm.

Water at 4 F above river temperatures causes a greater mortality by columnaris among fish than water at the river temperature. Further, death was found to occur earlier for younger fish than for older.

With the help of RPO, new dosimetry standards for Sr^{90} were prepared. These were made from Sr^{90} and plaster-of-paris. An extrapolation chamber was used to measure dose rates from the cubes which will be used for determining doses to bone by autoradiography.

Six and four-tenths μc Ra^{226} intravenously administered per kg of body weight to a pig caused nephritis 7 to 9 months later. A sex difference may be showing up here since we have not yet seen any damage among male miniature swine so injected.

By using C^{14}-labeled palmitate, it appears that the mechanism of fatty liver production by Np^{237} injected into rats is caused by a decreased rate of oxidation in the liver.

DECLASSIFIED

DECLASSIFIED

An original deposition of 2 mc of $Ce^{144}O_2$ caused the death in one of four dogs exposed 8 months ago. The lungs showed signs of radiation damage. Highest concentrations of the remaining radionuclide were found in the bronchial lymph nodes.

Two and one-half years after a dog had been exposed to $Pu^{239}O_2$, it died due to bronchiogenic carcinoma. The final body burden was 1.8 μc.

Intercellular distribution of Pu^{239} from plutonium nitrate inhaled by rats was found to be markedly different than when the oxide was inhaled.

Polycythemic mice are now being used in our erythropoietin assay to determine effects of Sr^{90} on bone marrow. However, a factor previously unrecognized may be affecting the value of this assay. Tests have shown that polycythemic mice have a greater blood volume than normal mice.

Flour beetles were exposed to neutrons in the PCTR and the effects were compared to those obtained from accelerator neutrons. It appeared that the former was more damaging on the production of progeny, but the effect of gammas in the PCTR is unknown.

By means of tracer techniques, it was shown that selenium can be taken up by greasewood shrubs from physiologically dry soils.

5. Programming

Approval of AEC was obtained to operate the Fuel Element Rupture Test Facility in PRTR using a PRTR pressure tube with PRTR fuel elements.

Approval for operation of PRCF at an increased power level, 10 kw rather than 100 watts, was requested of AEC.

HW-77770, "The Hanford Isotopes Production Plant," was revised in minor details in accordance with an AEC-DID request and released to them. The intense interest in the subject matter of this report was highlighted by the fact that summarized data from the report appeared in "Chemical Week" and "Nucleonics Week" concurrently with its release.

A review of candidate heat source radioisotopes that would be useful if they could be isotopically separated showed Tl^{204} to have outstanding potential. Cs^{134}, Ho^{166}, and Tm^{170} are also of sufficient interest to justify preliminary investigation of special isotope separation techniques such as the Szilard-Chalmers process. Cs^{134} and Ho^{166}, because of their gamma activity may show no advantages over cheap Sr^{90} or Cs^{137}; however, Tm^{170} and Tl^{204}, each with only bremsstrahlung of moderate energy, may be highly competitive with costly Pu^{238}, Po^{210}, or Cm^{242}.

Fuel cycle costs and reactivity response characteristics were calculated for a reduced spatial concentration fuel containing U^{238} and Th^{232} enriched with relatively low exposure plutonium.

Calculations were made of the incidental production of Np^{237} in the fuel of various power reactor types and also in some special types with harder neutron spectra.

Preliminary computations using the ALTHAEA code were begun in a study to determine the behavior and economics of seed-blanket systems. Natural and slightly-enriched uranium were considered for the blanket, while three concentrations of fully enriched uranium and plutonium were considered for the seed.

As part of a continuing fuel conservation study for various reactor types with varied fueling modes, a general equation was developed to express the efficiency of fissile fuel utilization in a thermal reactor.

TECHNICAL AND OTHER SERVICES

Four new plutonium deposition cases, each estimated to be less than 1% of the maximum permissible body burden (MPBB plutonium with bone as reference is 0.04 μc), were confirmed by special bioassay analysis during the month. Three of the incidents occurred in the Weapons Manufacturing Building (234-5) and one in the Plutonium Fabrication Pilot Plant (308). The total number of individuals who have received internal plutonium

DECLASSIFIED

DECLASSIFIED

deposition is 325. With the death of one employee, the number of deposition cases currently employed is 236.

The Tri-City Area economic model has been completed. Despite certain raw data deficiencies, the model reveals useful approximations of the economic interaction between HAPO and the Tri-City Area.

Detailed statistical descriptions of preirradiated dimensional measurements on fuel elements canned by the AlSi and hot-die-sizing processes were prepared for comparison purposes.

Several studies were completed of the effect on stud pull strength of cans and rails, of variations in the type of material used, and the method of processing.

In connection with final product inspection, the variability of one of the control measurements decreased sufficiently during the month to permit changing from three to two sampling positions per part.

Testing of the first magnetic tape prepared to numerically control the Sheffield rotary contour gage was completed at the Schenectady Laboratories of Equipment Development Services.

Further work was done on the least squares program to calculate individual standard gamma spectra from a composite standard.

A second phase of the Monte Carlo atmospheric diffusion problem was started. A computer program has been written to customer specifications and some preliminary data obtained for study.

The statistical analyses of data from X-irradiation experimental studies on Tribolium were completed.

Eight equations of an empirical nature were developed which predict the amount of Pu^{239}, Pu^{240}, Pu^{241}, and Pu^{242} as functions of the percent burnup.

SUPPORTING FUNCTIONS

Plutonium Recycle Test Reactor output for August was 1545 Mwd, for an experimental time efficiency of 83.3% and a plant efficiency of 71.2%. August output is the highest monthly output since startup of the PRTR. There were nine operating periods during the month, of which seven were terminated by scrams, one was terminated manually and one continued through month-end. A summary of the fuel irradiation program as of September 2, 1963, follows:

	Al-Pu		UO_2		PuO_2-UO_2		Other		Program Totals	
	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd
In-Core	17	1445.4	7	1082.1	61	5185.6			85	7713.1
Maximum		132.0		213.0		128.7				
Average		85.0		154.6		85.0				
In-Basin	26	1707.9	59	4556.2	12	232.0	1	7.3	98	6503.4
Chemical Processing	32	2309.3	1	78.0					33	2387.3
Program Totals	75	5462.6	67	5716.3	73	5417.6	1	7.3	216	16603.8

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

A total of 73 reactor outage hours were charged to repair work. Main items were:

Instrument repairs	25 hours
Valve repairs	18 hours
Modifications to process tube outlet temperature detectors	10 hours
Replacement piping on primary heat exchanger	9 hours
Shim rod repairs	8 hours

The August heavy water inventory (for a 26-day period) indicates a loss of 763 pounds.

DECLASSIFIED

DECLASSIFIED

For the entire month of August the Plutonium Recycle Critical Facility was devoted to the reactivity measurement series for Pu fuel samples. Testing of cold samples was essentially completed at mid-month, and the first irradiated sample test was complete at month-end.

Four hot operating runs were conducted in the Fuel Element Rupture Test Facility during the month, two of which were terminated for planned maintenance, one was terminated as the result of a scheduled diesel generator load test, and one was terminated because of a plugged control valve.

Total productive time in Technical Shops Operation for the period was 26,271 hours. Distribution of time was as follows:

	Man-Hours	Percent of Total
N-Reactor Department	3 763	14.33
Irradiation Processing Department	4 352	16.57
Chemical Processing Department	343	1.30
Hanford Laboratories	17 813	67.80
Hanford Utilities and Purchasing Operation	0	--

Laboratory Maintenance Operation, total productive time was 21,050 hours of 24,400 hours potentially available. Of the total productive time, 86% was expended in support of Hanford Laboratories components with the remaining 14% directed toward providing service for other HAPO organizations. Manpower utilization for August was as follows:

A. Shop Work		2980 hr
B. Maintenance		8470 hr
1. Preventive Maintenance	1850 hr	
2. Emergency or Unscheduled Maintenance	2000 hr	
3. Normal Scheduled Maintenance	3850 hr	
4. Overtime	770 hr	
C. R&D Assistance		9600 hr

The heavy water inventory at the end of August 1963 showed a loss of 763 pounds ($10,478) for the PRTR and a gain of 4 pounds ($56) for the PRCF. Heavy water scrap generated during the month amounted to 2375 pounds resulting in a $1682 charge to operating costs. Heavy water accumulated at August 31 for return to SROO amounted to 13,221 pounds ($165,811).

Visitor activity is reflected in the following data:

	Number of Visitors	
	In August	Since 6-13-62
Visitors Center	3293	54,383
Plant Tours	156	n.a.

HAPO professional recruiting activity this month is summarized below:

	Plant Visits	Offers Extended	Acceptances Received	Rejections Received	Open Offers at Month End
Ph.D.	7	4	5	4	4
BS/MS (Direct Placement)		2	0	1	2
BS/MS (Program)		0	0	0	0

Three technical graduates were placed on permanent assignment. Eight graduates joined the program, and one member terminated. Program strength at month end was 79.

Results for the 1962-63 recruting year were:

Ph.D. recruiting - 26 acceptances
BS/MS (Program) - 99 acceptances
Direct Placement - 56 acceptances

DECLASSIFIED

DECLASSIFIED

Authorized funds for 10 active projects total $6,735,000. The total estimated cost of these projects is $10,640,000. Expenditures through July are not available.

Manager, Hanford Laboratories

HM Parker:JEB:dph

REACTOR AND FUELS LABORATORY MONTHLY REPORT

AUGUST 1963

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 02 PROGRAM

1. Metallic Fuel Development

Target Element Development. Irradiation testing of 16 experimental lithium-aluminum target elements in the KER test loops 1 and 2 continued at a coolant pressure of 1600 psi and a calculated target temperature of 300 C. Sixty days of exposure at full reactor power had been accumulated by the elements as of August 20, 1963. Loop 1, which contained eight target elements, was discharged on August 22, 1963. As a result of loop operating difficulties, only the last 35 days of irradiation in Loop 1 were at elevated temperature. The eight target elements were examined in the KE basin after discharge. All of the target elements were coated with a reddish-brown deposit but otherwise appeared unchanged as a result of irradiation. Following "weaseling" and measurement of bulk density, the target elements will be shipped to Radiometallurgy for examination.

A coextrusion with Li-Al core and Al clad was prepared in the course of investigating the transport of lithium in aluminum. A continuous, unidentified phase, approximately 0.0001 inch in thickness, was observed at the core-clad interface. No grain growth across the interface or change in the morphology of the layer was observed after 24 hours at 600 C.

High Temperature Irradiation of Tubular Fuel Elements. Fuel swelling and operational data of eight 12-inch KSE-5 fuel elements which were exposed in the range 1070-1350 MWD/T at volume mean fuel temperatures in the range 433-535 C are being analyzed in terms of parameters of the uranium swelling model. Analysis thus far indicates there is good agreement between the higher temperature KSE-5 test and previously obtained KSE-3 data at 1200 MWD/T exposure. On the basis of these data (KSE-3 & 5), fuel swelling begins at a volume mean fuel temperature near 400 C and approaches a theoretical maximum, which is limited by the loop coolant pressure, at a volume mean fuel temperature of 525 C.

N-Reactor Fuel Evaluation. Inner tube (NIE) and outer tube (NOE) supports have been crush-tested in the Radiometallurgy facility after KER irradiation to approximately 85% of N goal. One outer-tube suitcase handle support was crushed without cracking or failure of the spot welds. The crushing strength and deformation characteristics were not adversely affected by the in-reactor service. Three inner

tube "buggy-spring" supports were tested and the crushing strength and deformation characteristics of these supports had not been adversely affected by irradiation. Cracking in the loops of the "buggy-spring" supports was observed after testing. However, these supports were produced from commercial strip and cracking during crushing was not unexpected. Both the inner and outer supports were removed from the fuel tubes by shearing the spot welds. No evidence of corrosion or crud buildup in the crevices between the support tabs and the heat generating surfaces of the elements was evident.

Fluted Fuel Element Irradiations. Two fluted N-Reactor size, single-tube fuel elements are currently undergoing the sixth cycle of irradiation in the ETR M-3 facility. These fuel elements are operating at a maximum specific power of 130 kw/ft with a corresponding maximum fuel temperature of 520 C. After an exposure of 1000 MWD/T a volume increase of 0.3% was indicated by density measurements.

Post-irradiation examination of the N-inner size fluted elements which were irradiated to 2000 MWD/T in cold water in the KE Reactor was started. On these six-fluted elements pronounced swelling had occurred in localized areas near the ends and appeared to be within the zone affected by the heat of brazing. On metallographic sections the swelling could be characterized as "grain-boundary tearing" in which there is a mechanical separation of the highly distorted uranium grains or subgrains. This phenomenon has been observed in high exposure elements with metal temperatures above 375 to 400 C. It was not anticipated that grain boundary tearing would occur in the elements of this test since the maximum fuel temperature was only approximately 300 C.

N-Reactor Fuel Support Development. Annealing studies on N-inner supports have established that alpha-phase annealing of the finished support will sufficiently alter the crystallographic texture of the Zircaloy-2 material to eliminate the deleterious effects of hydride formation during subsequent autoclaving. In supports which have been annealed at 675 C the hydride platelets which precipitate during autoclaving are comparatively small and randomly oriented. Such supports have ample ductility for the sizing-up operation and the possibility of embrittlement from the formation of large hydride platelets during irradiation may be eliminated. This latter aspect of service requirements for supports is the subject of a study now under way.

Cladding Deformation Studies. Thirty-six NaK capsules containing a total of 94 Zircaloy-2 clad uranium rods have been irradiated to provide data on the cladding strain capabilities as a function of cladding thickness uniformity, temperature, and exposure. Nine of the 27 capsules which were discharged at 1800 MWD/T have been opened and the fuel samples examined visually for cladding condition. Performance of these Zircaloy-2 clad fuel samples was consistent with

DECLASSIFIED

DECLASSIFIED

the results observed from similar samples irradiated to a lower exposure of 1000 MWD/T. On samples with either one, two or four 0.008-inch striations at the fuel cladding interface, necking or splitting of the cladding occurred at each striation. On the test samples with 0.004-inch deep striations at the fuel-cladding interface, necking of the cladding was observed at only part of the striations. The observations continue to indicate that plastic instability of the cladding can occur at low total cladding strain if the depth of the striations at the fuel cladding interface is 15% or more of the nominal cladding thickness. Diameter measurements have not yet been completed on these samples.

Macrophotography and some diameter measurements were completed on the three 6-inch long samples with nominal cladding thicknesses of 0.040-inch and 0.025-inch. One end of each of these rods is open. This flat unrestrained surface of uranium offers no resistance to longitudinal movement of the uranium. Measurements made on a composite macrophotograph of one sample showed the diameter increase was from zero at the open end to approximately 0.006-inch at a distance of $1\frac{1}{4}$-inch from the open end and remained constant at 0.006-inch to the closed end of the rod. The plasticity of the uranium and the low restraint at the open end thus partially relieved the swelling induced stresses in the fuel cladding for a distance of $1\frac{1}{4}$ inches.

Fuel Element End Cap Temperatures. A machine solution for the temperature distribution in the end closure region of clad fuel has been developed, and three basic geometries for N-inner fuel closures have been studied. These are the flat cap, the V-shaped cap where the cap is mitered into the uranium, and the inverted V-shaped cap. The V-shaped end cap gave the lowest end cap temperature and the greatest reduction in uranium temperatures in the end cap region. The flat end cap geometry gave the highest end cap surface temperatures.

Hot Headed Closure Studies. A series of 12 hot-headed and projection-welded experimental fuel elements were processed to determine the effects of variations in the amount of copper at the interface between the cap and the element and to determine the extent of the heat affected regions due to projection welding and variations in the amount of post-heating. These samples have been subjected to "cap-to-core" nondestructive tests using N-Reactor Department's equipment and standards. The results of these tests indicated that unbonding was detectable up to a certain level of post-heating. Above this level of post-heat (12 impulses of 60-cycle current flow at 40% phase shift) there were no significant indications of unbonded areas. These samples will be prepared for metallographic examination.

Fuel Straightening. The three roll straightener is being evaluated for use in hot straightening of NIE fuel elements. Five elements have been hot straightened after beta heat treatment. The initial double throw warp of the fuel elements was 0.185, 0.196, 0.148, 0.269, 0.244 inch. The warp after straightening was 0.011, 0.020, 0.017, 0.021, 0.022 inch, respectively. Some difficulties were experienced with roll adjustments, chute adjustments, and element collaring. Adjustments are being made and work is continuing toward evolving a process from this equipment.

Closure and Joining. As part of the program to determine the tensile and impact properties of the various brazing alloys, casting of tensile specimens by an investment casting process is being developed. This process is based on the ancient "lost wax" process where a wax pattern is cast in a permanent mold, an investment of a suitable ceramic is made around the wax pattern, the investment cured, then the wax melted out. After removal of the wax, the mold is fired to remove the volatile impurities then transferred to the vacuum melting furnace while hot. The present program is directed towards finding a suitable mold material. In general, the best results have been obtained with alundum, zirconia, and plaster of Paris liners. Of these, plaster of Paris gives the best surface but tends to break down. Zirconia is extremely inert but is very brittle in the "green" state. Alundum was reasonably strong but very porous. The best compromise seems to be 30% plaster of Paris and 70% -65 mesh alundum. A backup mold is cast around each liner of 40% sea sand, 30% plaster of Paris and 30% silica flour, with either fiber glass or fiberfrax added for strengthening.

Corrosion tests on the Zircaloy-beryllium, Zircaloy-beryllium-uranium and Zircaloy-uranium alloys have been expanded to include the pH of the corroding media. In general, the 2.5% Be alloys had corrosion rates, at 410 hours in pH-10, equal to those of the same composition at 900 hours in pH-7. The appearance of the oxide was considerably different. The pH-10 oxide for the alloys of 1.5% Be and greater was yellow in color and weakly adherent. Alloys above 3.5% Be were shite and tended to brush off easily. The effect of uranium on the beryllium containing alloys was much less in pH-10 than in pH-7. Zircaloy containing up to 1% uranium remained glassy black with weight gains only slightly higher than for Zircaloy-2 but less than for the 0.5 Be-Zircaloy alloy.

Fuel Fouling Detector. Development work on a thermocoupled one-inch diameter Zircaloy-2 clad, thorium-uranium fuel rod for use as a fuel fouling detector has continued. Several rod sections have been Be-Zr braze-closed to both establish brazing parameters and to provide experimental pieces for evaluation purposes. Radiography and autoclave testing show the closures to be sound and free of thorium contamination.

Metallographic examination of closures, including one in which a short section of a Zircaloy-2 clad thermocouple was brazed in place, is under way.

Three double-sheathed Zircaloy-2 clad thermocouples have been received and are being evaluated. Two isolated defects in the outer sheathing were located during nondestructive testing. Sections of the thermocouples are being examined metallographically to establish the nature of the indicated defects.

Fabrication of the fuel components for 12 thermocoupled fuel probes was started.

2. Corrosion and Water Quality Studies

Corrosion Testing Program for N-Reactor Steam Generators. Corrosion tests are being performed to determine the cause of the localized intergranular attack and stress cracking found in the N-Reactor steam generators. Analyses of water and residues from the various heat exchangers are being made to determine sources and types of possible contaminants. Results of chemical analyses on residues from Unit 3-B show that a neoprene purge balloon was charred in the piping of that unit. Tests are presently under way to determine if the charring of neoprene rubber in the presence of stressed sensitized stainless steel will cause stress cracking. The decomposition products given off from charred neoprene are highly acidic and contain large amounts of chloride. Some stainless steel samples were sensitized in the same manner as the tubes of the N-Reactor steam generators, stressed to yield, and exposed to the vapors from charred purge balloons. After one week of exposure at room temperature, no cracking was observed on the samples.

In another test, samples of stressed sensitized stainless steel and reject tubing from the same lot as used in the N-Reactor steam generators were exposed to charred and uncharred vapor phase inhibitor. After one month at room temperature, no cracking was noted.

Tests are in progress to determine whether localized carbon pikcup can be induced in stainless steel by heat-treating samples containing lubricating residues.

N-Reactor Secondary System Cleaning. A test has been completed to determine whether the cleaning compounds proposed for use in the N-Reactor secondary system would result in pitting in the crevice areas of the dump condenser tube sheets. Stressed crevice samples of carbon steel and 70-30 Cupro nickel were exposed up to 24 hours to Vertan 690 cleaning solution at room temperature, passivated in an alkaline nitrite, and boiled five hours in distilled water. Chemical deposits were found in the crevice areas but localized attack was not observed.

Evaluation of NH_4OH for N-Reactor Primary Coolant. In support of the program to evaluate NH_4OH for pH control in the N-Reactor primary coolant, the KER-1 loop was operated during the past month with ammoniated water at pH 10. Two high temperature runs were made. One test lasted about 12 days and the other lasted 15 days, at which time the test was terminated.

During the first test, some turbidity was encountered in the coolant while the loop temperature was being increased. This effect is also common during temperature transients with lithiated coolant in the system. The turbidity decreased rapidly during steady-state operation, and no visible turbidity was encountered during the last eight days of the test. During the last four days of the test the purification (feed and bleed) rate was decreased 50% to determine the rate of radiolytic decomposition of NH_4OH. No turbidity was encountered, and the water quality was maintained at the desired level without difficulty.

During the second test, particulate solids were again encountered during the startup period, but the turbidity did not decrease as rapidly as in the previous test. Evaluation of this problem is complicated by the fact that the purification rate was again reduced to study decomposition rates of the NH_4OH. It is possible that the turbidity remained high because of this change. It is also possible that solids deposited in the sampling line during the period of high turbidity may have "leaked" into subsequent samples giving erroneous results. This particular effect was observed in one instance, when two samples collected within a period of 30 minutes showed a wide variation in turbidity - the first sample was turbid and the second was clear.

A sample line has been installed in the KER-1 feed line to permit continuous measurement of the ammonia concentration. These results will facilitate an experimental determination of the ammonia decomposition rate in the reactor.

The rate of thermal decomposition of ammonia in dilute solution is being studied in TF-3 in connection with a special corrosion test. The coolant in TF-3 is ammoniated water at pH 10, 314 C and 2000 psi. In the test, samples of feed and bleed streams are taken every four hours and analyzed with the Technicon Autoanalyzer. Although only a few measurements have been taken, there does not appear to be any appreciable decomposition of the NH_4OH. If more complete measurements agree with these preliminary data, it may be concluded that the decomposition of NH_4OH noted in KER-1 is due almost entirely to radiolytic effects.

Corrosion Testing of N-Reactor Rolled Joint. An N-Reactor rolled joint test section has been fabricated to determine corrosive effects

DECLASSIFIED

of repeated decontaminations of the rolled joints. Testing will start next month in TF-1; the section will be examined periodically to determine extent of galvanic attack or other forms of localized corrosion.

Decontamination Studies. A new proprietary bisulfate compound has been tested and appears promising as a decontaminating agent. The decontamination factors ranged up to 50 for carbon steel (one-step process), and up to 100 for stainless steel (two-step process; alkaline permanganate followed by bisulfate). The corrosion rates are low, 0.06 mil/hr.

In-Reactor testing of Nonchromium Inhibitors. A production test was prepared to authorize testing in 1706-KE single-pass tubes, of mixtures of silicates, nitrites, and sulfites for inhibition of aluminum and steel corrosion.

Ten additional static tests were started in the 1706 Laboratory. These tests compare different concentrations of sodium silicate for inhibiting corrosion of carbon steel and aluminum. Concentrations ranging from 10 to 200 ppm added SiO_2 are being tested with filtered water, 100 ppm sodium dichromate, and 100 ppm sodium nitrite as reference systems.

Corrosion on Heat Transfer Surfaces in Process Water. High watt density electrical heaters clad with X-8001 aluminum are now operating in TF-20 to determine corrosion effects on heated surfaces in process water. Prior to installation the clad heaters were photographed, weighed, and the diameters measured with a micrometer.

Carbon Dioxide for pH Adjustment. Additional samples of aluminum and steel were discharged from the 110 C, pH 6.7, CO_2 test in the 242-B Water Studies Apparatus after eight weeks of exposure and are being processed for corrosion and film measurements.

Outgassing Rates from Aluminum-Lithium Alloys. An in-reactor capsule experiment is presently in progress in KE Reactor to measure tritium outgassing rates from irradiated aluminum-lithium alloys. At temperatures below 370 C, no tritium evolution has been measured. However, a slow leak developed which allows helium to pass from the outer to the inner capsule. It is estimated that this reduced the sensitivity by a factor of 10^3.

When the temperature was raised to 500 C, the tritium evolution rate was 3.4×10^{-2} cc T_2/min/cc alloy.

Data from ex-reactor tritium outgassing studies at duPont involving different sample geometrics indicate rates of 2×10^{-1} cc/min/cc alloy at 500 C. In the Hanford experiment the time at 500 C was limited to four minutes to prevent excessive swelling of the elements.

Prior to discharge of the capsule on August 22, 1963, outgassing runs will be made at lower temperatures for longer times. An improved capsule is being prepared for test.

Nickel Plating Uranium-Anodic Etch Bath. The etching characteristics of uranium in the phosphoric-hydrochloric acid etch bath are under study. Short pieces of typical production uranium were etched in baths containing various phosphoric and hydrochloric acid concentrations. The etched pieces were then examined microscopically at 16X for unetched areas. An etch solution composition of 6.0 M/l H_3PO_4 - 0.4 M/l HCl produced the least unetched uranium. Tests are continuing on baths with various uranium concentrations.

Zirconium Alloy Corrosion Capsule. Capsule irradiations of Zircaloys in high temperature water, which will follow the lithium-aluminum alloy irradiation in KE-2A test hole, have been delayed because of the necessity to charge a second lithium-aluminum capsule. However, equipment for controlling water quality is being assembled and tested. Water is being analyzed for dissolved oxygen and hydrogen in preparation for controlling these gases in subsequent in-reactor zirconium alloy corrosion experiments.

Effect of Beryllium and Uranium on the Corrosion of Zircaloy-2. The effect of beryllium and uranium on the corrosion of Zircaloy-2 is being jointly investigated by CMO and MFD. Zircaloy-2 samples containing varying amounts of beryllium (0-5%) and uranium (0, 200 and 2200 ppm) have been corrosion-tested in 360 C, pH-10 (LiOH) deoxygenated water for 14 days. Weight gain of Zircaloy-2 increases with an increasing beryllium concentration. For Zr-2, Be, U "ternaries" weight gain was independent of U concentration.

3. Gas Atmosphere Studies

Effect of Precipitated Hydride on Hydrogen Absorption of Zircaloy-2. This experiment will establish the effect of precipitated hydride in Zircaloy-2 on subsequent hydrogen absorption during corrosion. One group of samples were cut from a sheet of Zircaloy-2 that had been charged to 410 ppm of hydrogen and homogenized at 800 C. A second group of samples were vacuum annealed only. Both sets of coupons were exposed for 294 days in 400 C (750 F), 1500 psi steam. The pretransition corrosion rates were the same for both materials. The post-transition corrosion rate of the hydrided material was about 15% higher than the unhydrided sample. The hydrogen absorption rate of the hydrided specimens was about 20% lower than the unhydrided specimens. These results would indicate that development of a precipitated hydride phase during long term exposure of reactor components to high pressure steam should not produce any major change in either corrosion rate or rate of corrosion hydrogen absorption. Another two sets of these same samples have been exposed 270 days in 360 C (680 F)

DECLASSIFIED

deionized water and the results to date also indicate no appreciable change in either corrosion or hydriding due to hydride precipitation.

Graphite Burnout Monitoring. Small graphite burnout monitors exposed in channel 3461 at B-Reactor from June 12 to July 16, 1963, showed burnout rates below 2% per 1000 operating days (%/KOD). Monitors for discharge about the end of August were recharged into channel 3461 and also into a newly available channel, 1081. The results will be of more than usual interest in view of the new production test at B-Reactor which allows higher CO_2-to-He ratios.(1)

Monitors exposed in channel 1880 at KW-Reactor from April 3 to August 3, 1963, showed one peak of about 10%/KOD at the usual location for the graphite-O_2 reaction at 80 inches into the channel and a broader peak of about 15%/KOD at 150 inches. CO_2 is not a component of the gas atmosphere in this reactor and the water concentration was low during the period of this test. The peak may, therefore, have been due to a higher reactivity of the monitors at this position. The monitors in the first 102 inches of the channel were TSGBF graphite; those monitors in the remaining length of the channel were KCO graphite. TSGBF graphite which was used in the construction of the K-Reactors, if of higher purity than the KCO graphite, which was used in part of the construction at DR-Reactor. Laboratory tests have shown that the oxidation rate of KCO graphite is about four times faster than TSGBF graphite. Because of these results only F purified graphite will be used in future monitoring tests.

Monitors were also discharged from channel 3066 at KE-Reactor. The test period was from June 2 to July 3, 1963. Results showed burnout rates below 0.5%/KOD.

Thermal Oxidation of Large Graphite Bars by Water Vapor. The investigation of the effect of the sample size on the oxidation rate of graphite by small partial pressures of water vapor has continued. The concentration of water vapor in oxygen-free helium was maintained at about 7000 ppm by bubbling the helium through water at 0 C and was measured with an electrolytic hygrometer. The cylindrical TSX-graphite sample used was 1.5-inch in diameter and 12 inches long. The oxidation rates at 710, 757, 793 and 836 C were respectively 8.8, 18.6, 70 and 130 x 10^{-7} hr^{-1}. These values were used to derive the rate equation

$$\text{Rate } (hr^{-1}) = (4.82)(10^4)e^{-48,500/RT}$$

which, it should be noted, is applicable only to the experimental conditions used. The rates found in this study are less than those

(1) R. E. Baars, S. M. Graves and P. J. Zimmerman, "Production Test IP-593-J," July 1, 1963.

observed for smaller samples. A run at 840 C was also made using helium from which the oxygen was removed but to which water vapor was neither added nor removed. The resulting rate was only about 3% of that in 7000 ppm water vapor. This indicates that if the reaction is first order with respect to H_2O in this temperature and partial-pressure range, the rate is that expected for a water concentration of about 175 ppm. The agreement of the calculated value with the measured value of 125 ppm H_2O is surprisingly good considering the extensive extrapolation involved.

Future runs will be performed under similar temperature and partial pressure conditions with TSX-graphite samples whose diameters range from 0.5 to 3.5 inches.

Kinetics of the Reaction $CO_2 + H_2 \rightleftharpoons H_2O + CO$. Both the forward and reverse steps of this reaction are being studied for application to the N-Reactor gas-compatibility problem. The initial attempt to use available gas chromatography equipment to follow the progress of the forward reaction was unsuccessful. The reaction rate was therefore measured by collecting the water on anhydrous calcium chloride or magnesium perchlorate.

A study of the rate of the reaction as a function of hydrogen gas pressure has been completed. The plot of the rate of formation of water as a function of hydrogen concentration to the one-half power gives a satisfactory straight line; whereas the first-order plot shows pronounced curvature. At 840 C the average of seven runs gave a specific rate constant of $4.08 \pm 0.66 \times 10^{-2}$ liters$^{\frac{1}{2}}$ moles$^{-\frac{1}{2}}$ sec^{-1}. This can be compared with a value of 11.3×10^{-2} calculated from the results reported by Graven and Long (J.A.C.S., 76, 2026 (1954). The reason for the difference is not yet apparent.

4. Process Tube Development

Tube Burst Test. An alternate for the projectile method has been devised for introducing a crack into a tube section undergoing testing of its brittle fracture characteristics. This method involves milling a 1/16-inch wide slot 80% through the tube wall. A principal advantage of this method is that the length of the defect can be varied, a parameter that is fixed in the projectile method. Application of pressure causes a through-crack to form in the area of the defect. Leakage is prevented by an aluminum patch cemented to the inside of the tube at the crack site so that pressure may be increased to the level at which the crack will propagate into the body of the tube.

Single tests at room temperature have been performed on N-Reactor tube sections by this method at two crack lengths: 0.350 inch and 0.810 inch. The tube with the shorter defect failed at a hoop

DECLASSIFIED

DECLASSIFIED

stress of 89,000 psi and the longer at 75,000 psi. The lower of these values is more than eight times the operating stress.

It appears likely that information can be obtained on irradiated tube sections by this method without waiting for completion of the irradiated burst test facility.

A metallography specimen has been examined from the maximum irradiation section of the Zircaloy process tube removed after four years of exposure in KER Loop 2. There is no visual evidence of hydride accumulation in this material even at the very surface. The same heavy agglomeration of the second phase material is present, however, as that reported for the earlier Loop 3 tube (HW-75052).

5. Thermal Hydraulic Studies

N-Reactor Studies. Installation was completed of the full scale electrically heated model of a half column of N-Reactor fuel elements in the high pressure heat transfer apparatus. The test section consists of two coaxial tubes, $17\frac{1}{2}$ feet long. The inner tube is 0.440-inch ID and 1.220-inch OD and is supported with ceramic spacers inside of the outer tube which is 1.775-inch ID and 2.390-inch OD. The materials of the two tubes vary along the length to produce the downstream half of a cosine heat generation curve. Thermocouples are installed within slots milled in each tube to measure surface temperatures. The test section is designed for heat generation rates of 2700 KW under steady state conditions and 3800 KW during overload periods of one-half hour.

Initial runs with the test section were made to investigate the pressure-drop effects of the orientation of fuel support pieces in the outlet nozzle. The angular position of the most downstream support piece determines the degree of alignment of holes in the support piece with the nozzle exit. Previous experiments by Equipment Development-IPD showed that when holes in the support piece were in line with the exit nozzle, the pressure drop was considerably less than when these flow passages were not in line. As a result, additional holes have been drilled in the support pieces, and it was the purpose of the experiments with the test section described here to determine the pressure drop of the modified support pieces.

Results to date have not been conclusive. The pressure drop across the nozzle outlet was influenced much less by orientation of the modified support pieces than by the original support pieces. On the other hand, the modified support pieces produced a larger pressure drop across the nozzle outlet and also across the column of support pieces. The results to date are not complete, however, and more data will be obtained in future tests.

6. Shielding Studies

Cross-Sections for MAC Code. The program to calculate two complete sets of neutron and gamma ray cross-sections for MAC Code is well under way. Two cross-section sets are needed: one set of a coarse group structure (~ 10 groups) is needed for preliminary type design work in which small amounts of programming and computer times are desired; another set of fine group structure (31 groups) is needed for final design work in which reliable and accurate calculations are required. The 31-group cross-section set is almost complete.

Unit weighted, energy dependent (31 group structure) absorption, elastic scattering and inelastic scattering neutron cross-sections for isotopes were calculated with a modified version of BARNS Code. The elemental cross-sections were then calculated with a separate specially-written computer code.

Gamma ray cross sections were obtained from established compilations of gamma ray interactions.

Energy dependent neutron removal cross sections are currently being evaluated by one or two methods. Angular distribution of elastic scattering data given by RBU are used wherever available. Otherwise, the total cross-section of an element is converted to its removal cross-section using data presented in Price, Horton and Spinney, "Radiation Shielding." A better method of evaluating the removal cross-sections, using data presented in the Second Edition of BNL 400, is not used because it is too laborious for the present purpose.

N-Reactor Shield Experiment Preparation. An electronics system is being assembled which is designed to process signals from several types of neutron detectors into isotropic energy resolved neutron fluxes incident to the detector. The apparatus consists of a TMC 400-channel pulse height analyzer, two amplifiers, two preamplifiers, a summing circuit, a coincidence analyzer, a He^3 detector, and related equipment. The entire system was set up in the exponential pile laboratory where a plutonium-beryllium neutron source is available for testing the counting system. Initial checkout of the system disclosed that the pulse amplifiers of the coincidence signal processing system will not provide a linear gain for the signal to the pulse height analyzer. It is suspected that the pulse shape produced by the amplifier is only marginally accepted by the analyzer. This is being investigated further.

7. Graphite Studies

N-Reactor Graphite Irradiations. The last of the second-generation capsules, H-6-2, in the series of long-term irradiations of N-Reactor

DECLASSIFIED

[redacted]

DECLASSIFIED

graphite was removed from the GETR on July 16. The capsule operated successfully for 114.0 effective days (5 cycles). Maximum neutron exposure to the samples is estimated to be 5×10^{21} nvt, E > 0.18 Mev. The capsule was successfully disassembled during the week of July 22, and all 24 samples and 65 flux monitors were recovered without difficulty and were shipped to Hanford.

The second of the third-generation capsules, H-5-3, was installed in the GETR on July 17. It pressure-tested satisfactorily and on reactor startup all nine thermocouples functioned properly. Sample temperatures agreed quite well with other capsules in the series. The other capsule, H-4-3, presently in the GETR continues to operate satisfactorily with the exception of one thermocouple which failed just prior to reactor startup.

B. WEAPONS - 03 PROGRAM

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

C. REACTOR DEVELOPMENT - 04 PROGRAM

1. Plutonium Recycle Program

Fuels Development

PRTR Fuel Element Inspection. A significant modification to the PRTR fuel fabrication process was made with the adoption of homogeneous UO_2-PuO_2 fuel made by high energy pneumatic impaction (Nupak). The Nupak UO_2-1 w/o PuO_2 fuel is separately compacted to over 99% theoretical density and crushed and sized for final fabrication by swage or vibrational compaction.

An experimental process line was set up with a capacity of 100 pounds of Nupak material daily. Nupak material for five PRTR clusters is being prepared.

Impurity Content of PuO_2. Improved analytical techniques have indicated excessive concentrations of fluorine, chlorine, and water in PuO_2 powders recently prepared by oxidation of Pu metal buttons or by calcination of Pu oxalate. Impurity levels as great as 60 to 600 ppm fluorine and 100 to 1200 ppm chlorine were found in the PuO_2 powders produced from Pu metal buttons by air oxidation. This powder also contained 200 to 2500 ppm water due to inadvertent exposure to abnormally moist air during powder processing. PuO_2 powders produced by low temperature, ~ 200 C, calcination of the oxalate followed by higher temperature, ~ 950 C, calcination were found to contain fluorine in concentrations as great as 100 to 3000 ppm. Chlorine impurity was low in this type of PuO_2 powder. Two hundred to 2500 ppm water was present in the oxalate derived powder due to incomplete drying or possibly due to sorption from the atmosphere.

Purification of Impure PuO_2. Plutonium dioxide recently prepared for PRTR and EBWR loadings contained impurities in greater than normally acceptable concentrations; i.e., > 100 ppm water and > 10 ppm fluorine and chlorine. Consequently, new purification processes were developed and tested on high impurity content PuO_2 powders. A simple pyrolytic treatment of PuO_2 powders in air for 12 hours at 950 C with occasional mechanical stirring gave the best results. Fluorine, chlorine and water impurities were reduced to allowable concentrations as indicated below.

Purification of Impure PuO_2 Powders

Impurity	Before Treatment	After Treatment
Fluorine	680 ppm	4 ppm
Chlorine	1180 ppm	< 20 ppm
Water	2500 ppm	< 100 ppm

DECLASSIFIED

DECLASSIFIED

Purification of plutonium dioxide by this process is now routinely performed.

Post-irradiation Examination of PRTR Fuel Elements. High fission product activity was detected in the PRTR primary coolant shortly after startup of July 24, 1963. Stagnant water tests and visual examination in the PRTR storage basin confirmed that three new swage-compacted UO_2-1 w/o PuO_2, Mark-I (5198, 5199 and 5200) fuel elements had suffered rapid cladding failures. Small streams of gas bubbles disclosed one hole in 5198, six holes in 5199, and two holes in 5200. Since the reactor had reached only 10 MW, and since the activity was detected immediately upon startup, it is possible that the cladding was defective when the elements were charged. Subsequent helium leak checking of twelve similar, nonirradiated rods revealed one cladding defect. Leak checking before autoclaving had showed no leaks.

Post-irradiation examination of one of the defect areas revealed small radially oriented cladding penetrations. The radial direction of the penetrations is perhaps indicative of preferred orientation in the swaged Zircaloy cladding. The inner surface of the cladding in the defect area is characterized by a grey, zirconium oxide corrosion product, characteristic of Zircaloy corrosion in the presence of fluoride ion and water. The maximum diameter of the intergranular cladding penetrations is only 0.00125 cm (0.0005-inch). The cladding defects have the appearance of a stress-corrosion phenomenon and no zirconium hydride platelets were found. This is in contrast to the observations on vibrationally compacted fuel rod failures where the corrosion rate was slower, the attack was more general, and zirconium hydride was observed in some areas.

Post-irradiation examination suggested contaminants in the fuel, and chemical analyses of archive samples substantiated that unexpected large quantities of fluorine, chlorine and water were present in the PuO_2 used in the fuel.

Post-irradiation examination of the failed fuel rod from a vibrationally compacted UO_2-PuO_2, Mark-I element (5179) indicates that the failure was caused by internal corrosion of the Zircaloy cladding. The point of failure was a raised blister near the bottom of one rod. Metallographic examination of the cladding in this area shows that the failure mechanism was catastropic corrosion from within. Severe internal cladding corrosion was also observed in the end cap regions which were removed from the point of failure. Similar occurrences previously observed in other failed vibrationally compacted PRTR fuel elements (5113 and 5174) suggest related causes. Analyses indicate

that these failures also may have been caused by impurities (fluorine, chlorine and water) in the fuel. Analyses of archive samples are in progress.

PRTR Fuels Compatibility. Zircaloy-2 clad capsules of UO_2, PuO_2, and UO_2-PuO_2, containing various controlled amounts of halogen and water impurities, were held in excess of 250 hours at 400 C without any apparent sign of attack to the cladding. Impurity contents in the fuels ranged from 0 to 7000 ppm of fluoride, chloride and/or water. These results suggest that attack of the cladding will only proceed if part of the fuel is at a much higher temperature so as to release the halogen.

PRTR Fuel Element Inspection. Two swaged UO_2-PuO_2 fuel elements (5133 and 5150) were examined in the FEEF. Element 5133 was permanently removed from the charging schedule due to mechanical damage on one rod. This element may be further irradiated in the PRTR rupture loop at a later time; it now has 1560 MWD/ton exposure. Element 5150 appeared to be in excellent condition after 1200 MWD/ton exposure.

Nonautoclaved Fuel Element Irradiation in PRTR. Three nonautoclaved Zircaloy-clad fuel elements were examined in the PRTR basin. During two PRTR operating periods, the first element charged (5121) had developed a smooth oxide coating which appeared to be of a uniform thickness along the entire length. Elements 5193 and 5194, irradiated for one operating period, had developed a smooth oxide coating with thickness variations from the upper to the lower end of the element because of coolant temperature variations along the length of the element.

Fuel Refurbishing. A UO_2-PuO_2 mixed oxide fuel element was repaired by splicing and reconnecting a broken rod wire wrap in the PRTR basin. This work was accomplished under approximately eight feet of water, using seven specially designed tools. The most unique of the special tools was a split-core fixture which enabled placing the 0.183 cm (0.072-inch) wire on the fuel element into a 0.193 cm (0.076-inch) hole in the splicing connector.

Fuel Element Design Studies. The UO_2 Mark-I fuel element with ring type end brackets designed to reduce wear corrosion has accumulated 157 days of exposure in the TF-7 loop facility under PRTR coolant conditions. Only superficial wear on the contact surface of the bottom end bracket has occurred during this time.

Fuel Element Rejuvenation. The rejuvenation fuel element GEH-4-84 successfully completed its second cycle of irradiation in the MTR and will be returned to Hanford for destructive post-irradiation examination. This 1.435 cm (0.565-inch) OD element generated a

DECLASSIFIED

maximum surface heat flux of 125 watts/cm^2 (~ 405,000 Btu/hr-ft^2) during irradiation.

One rod from the first cycle rejuvenation fuel element GEH-4-81 is undergoing post-irradiation examination. Preliminary results show minimal sintering in the annulus area between the ZrO_2 center tube and the outside of the element. The ZrO_2 tube is darkened but shows little ZrO_2-UO_2 interaction and appears intact.

Irradiation of Uranium-Plutonium Oxide. Additional burnup samples were obtained from the 24 irradiated UO_2-PuO_2 capsules. Fission gas release values were calculated based on mass spectrometric analyses of these samples (and in a few cases, cesium analyses) combined with Meleager calculations. The Meleager calculations, prepared for each composition and density by R. E. Peterson, provided an accurate relationship between fuel isotopic concentrations and burnup. The results indicate that fission gas release was significantly more affected by rod power than PuO_2 concentration.

Short Duration Irradiation of UO_2-PuO_2. Flow tests with the transport vehicle for short duration irradiations in the MTR VH-4 facility are scheduled to be conducted prior to the MTR Cycle 196 startup (about August 20-21, 1963). The data obtained will permit design of vehicles that can be used for short duration irradiations throughout reactor operating periods.

Preassembled 19-rod Cluster. Dye penetrant tests were made on the ID of the tubes of the two bundles. Several flaws were detected in both bundles, but indications were not severe enough to prevent the assemblies being used in flow tests.

Fast Reactor-Thermal Reactor Exchange Element. Flow tests were conducted on the fast reactor-thermal reactor fuel element mockup. Pressure drop is such that the fuel element will operate very satisfactorily in both the PRTR and in the proposed fast reactor.

A stress-rupture test on a prototype Zr-2 fuel rod end cap showed end cap failure under a load of 385 pounds. This test indicates that the apparently delicate end cap is sufficiently strong to withstand any expected fuel element conditions.

Eddy Current Fuel Element. Special Zr-2 - 304L end caps have been fabricated for the Eddy Current Fuel Element. The facility for vibrational compaction of the fuel element rods is complete and ready for use.

Corrosion and Water Quality Studies

PRTR Thread Lubricant Corrosion. A molybdenum disulfide containing lubricant was examined for corrosion effects on stressed stainless steel. Three stainless steel (type 316) flared tubing connectors containing a short length of type 304 tubing were exposed to atmospheric pressure steam at 265 C (509 F) for 1400 hours. Two of the three fittings had been heavily lubricated with the molybdenum sulfide lubricant prior to assembly, tightening and exposure. After exposure the fittings were examined, cleaned by the alkaline permanganate-ammonium citrate process and dye penetrant tested. No stress corrosion cracking was observed and corrosion appeared light and uniform.

Electrolytic Hydriding of Process Tube Sections. In order to hydride process tube sections (for mechanical property testing) without destroying the cold worked condition, it is necessary to keep the hydriding temperature low. One possible way to accomplish this is through use of an electrolytic technique, which employs a boiling aqueous electrolyte. Six process tube sections have each accumulated the first six-hour exposure in the electrolytic cell. A total of 30 hours of exposure per tube is envisioned, with intermediate anneals, before a level of 200 ppm is attained.

Oxidation of Crystal Bar Zirconium in Carbon Monoxide Environment. A sample of crystal bar zirconium has been oxidized on the semi-micro recording balance for 52 days at 500 C (932 F) in carbon monoxide carefully purified of oxygen and water vapor. The weight gain after 52 days was 84 mg/dm^2. This compares with about 400 mg/dm^2 in one atm steam and 120 mg/dm^2 in pure oxygen at 50-day exposure.

Dynamic Corrosion Studies. The long-term test in TF-1 to determine effects of decontamination on PRTR materials has been completed After six cycles, using the same procedures used for decontaminating PRTR, there was no evidence of pitting or other type of accelerated attack on any of the samples.

Concentration of Zirconium in Primary Coolant. The program to routinely monitor the zirconium concentration in the primary coolant as a measure of fretting corrosion was continued. According to the results obtained to date, the zirconium concentrations are generally within the 0-1 ppb range and are considered "normal" for this system. There were no sustained periods of high concentration during the month that would be indicative of significant fretting attack in the system.

Investigation of a continuous, colorimetric procedure (using Arsenazo as the indicator) for measurement of ionic zirconium concentrations in water was continued. Results obtained to date indicate that

DECLASSIFIED

DECLASSIFIED

concentrations at least as low as 5 ppb can be detected. Additional studies are being conducted to determine the maximum sensitivity and degree of reproducibility of the procedure.

Tests were also conducted using D_2O as the solvent, and samples from the PRTR primary system were analyzed using this procedure to determine whether zirconium could be detected. The procedure is satisfactory in D_2O samples, although initial results indicate a slightly lower sensitivity than in H_2O. The results from the PRTR samples showed qualitative agreement with emission spectrograph results from the same samples, although the actual concentrations indicated were only 40-80% of those obtained using the spectrograph.

PRTR Boiler pH Control. Tests were conducted with PRTR boiling water samples to determine the relationship between hydrate alkalinity and phosphate concentration for neutralization of free hydroxide with disodium phosphate. The objective of these tests is to investigate procedures for controlling the high pH conditions encountered in the boiler during operation. The results indicate that the phosphate neutralized only 0.7 times as much hydroxide as was predicted on the basis of the accepted neutralization reaction mechanism.

These tests confirm that high phosphate concentrations will be required to completely neutralize the hydrate alkalinity at low boiler blowdown rates. Neutralization must be accomplisehed to permit operation within the Whirl-Purcell (pH versus phosphate) relationship recommended for this system. The use of mono-sodium phosphate or phorphoric acid as the neutralizing agent would significantly reduce the residual phosphate concentration in the water when neutralization is complete. This alternative is recommended if corrosion of the chemical injection system piping would not be a serious problem.

Reactor Components Development

PRTR Pressure Tubes. A report entitled "PRTR Zircaloy-2 Pressure Tube Surveillance Program," HW-78456, was written and published. Its content is summarized as follows:

(1) Creep of the tubes in-reactor is not occurring, (2) the hoop strength is sufficient for the tubes in-reactor to sustain about three times the PRTR operating hydrostatic pressure, (3) the tubes in-reactor appear to have a low notch sensitivity, (4) about one in every five fret marks studied had abnormal hydride concentrations, (5) there has been no case where a hydrided fret mark has reduced the load capacity of a burst test specimen, (6) on bursting at pressures three or more times greater than operating pressures, cold worked portions of the tubes appear unable to arrest a traveling crack at room temperature

whereas annealed portions are capable of arresting a traveling crack, and (7) the PRTR environment is changing the annealed material properties in the direction of the cold worked material properties.

These conclusions are the basis for proposing that the scope of the destructive examination portion of the PRTR surveillance program be changed to a minimum tube discharge rate of one tube every 90 days at rated power rather than one tube per operating month. This will allow a more detailed examination of each tube discharged in anticipation of the need for fracture propagation and notch sensitivity properties as the strength of the annealed material approaches that of the cold worked material.

Pressure Tube Monitoring. Fourteen process tubes were visually examined this month. This group of tubes included several into which fuel element 1049 (UO_2) had been charged some time during approximately the past six months. Examination of these channels together with data from previous examinations strongly suggest that this element is much more prone to cause fretting corrosion in a process tube than most other fuel elements now in the PRTR.

Examination of the remainder of this group of tubes disclosed new areas of fretting corrosion. From the standpoint of tube wall penetration there were no severely fretted areas. To date, the enlarged area fuel element support pads have caused no observed fretting attack penetration of the process tube greater than one mil.

Reactor Components Development

Shim Rod Environmental Test Facility. The glass test section, which had been fabricated previously, was installed in the shim rod environmental test facility. A first generation rod was installed in the test section and the facility and the rod have operated satisfactorily during the month. A new, stronger, permanent-magnet brake to prevent control element coastdown has been fabricated and will be tested in the near future; the original magnetic brake was unsuccessful.

Fretting Corrosion Investigation. A 5-mil amplitude at 16 cps was measured at the center of the pressure tube in the EDEL-I test loop during fretting corrosion tests. No fuel element was installed during this test. The natural frequency of the pressure tube, installed, has been measured at 16 cps. The 5-mil amplitude of the process tube was damped to approximately 1 mil by the installation of a fuel element, but the 16-cycle frequency continued to predominate at the bottom of the tube.

DECLASSIFIED

The same vibration detectors were installed on the bottom jumper of PRTR pressure tube 1558. The frequency at maximum amplitude was 25 cps at full reactor flow. Primary system temperature had no significant effect on this frequency. EDEL-I studies in 314 Building indicate that the frequency is also independent of flow. The difference in frequencies between PRTR and EDEL-I is thought to be caused by the shielding sleeves in PRTR which effectively shorten the unrestrained length of the pressure tube.

PRTR Rupture Loop Grayloc Connector. Another series of leak tests on the PRTR Rupture Loop Grayloc connector was made using the hubs and clamps furnished by Gray Tool Company and an original seal ring. Fifty thermal cycles, between 100 F and 600 F, at a pressure of 2100 psig were run with no trace of leakage. This was the first successful Grayloc test at 314 Building. All previous seal rings tested with the replaced hubs and clamps were of the "new experimental" design. The major change in this design was shorter lips on the seal ring.

EDEL-II Test Facility for Reactor Valves. A scope drawing for a test facility which would be attached to EDEL-II for testing of available helium valves for reactor service was made. Initial contact with valve suppliers has been made prior to the selection of models to be purchased for tests.

Design Analysis

PRTR-PRCF Safety Studies. Delay in the Doppler effect in UO_2-PuO_2 mixed oxide fuels because of the finite heat transfer time from the PuO_2 particle to the U-238 O_2 matrix is being investigated. A satisfactory mathematical analysis of the temperature transients upon reactor excursion was formulated in spherical geometry and is being applied to the problem.

PRTR Power Level Potential. The possibility of increasing the PRTR power level is being re-examined. A review was made of the preliminary study report of last year, and suggestions were outlined for updating this study and analyzing the key problem areas.

Thermal Hydraulic Studies

Plutonium Segregation Experiment. Fourteen experimental boiling burnout conditions were determined for an electrically-heated test section which simulated the type of heat flux distribution found in segregated mixed oxide fuel elements. The heated portion of the test section consisted of a piece of Inconel tubing, 24 inches long with a 0.820-inch OD and 0.435-inch ID. The outside of the tube was machined so that localized regions of high heat generation 0.1-inch wide occurred at 1.0-inch increments along the tube. Cooling water

was passed through the bore of the tube. Preliminary estimates of the axial heat flux distribution indicated that the maximum heat flux occurring in the 0.1-inch wide high flux regions was approximately 3.25 times the average flux and four times the minimum heat flux. The results are summarized in the following table.

Mass Velocity lb/hr-sq ft	Inlet Enthalpy B/lb	Heat Flux at Burnout Maximum B/hr-sq ft	Heat Flux at Burnout Minimum B/hr-sq ft
1.0×10^6	312	1.0×10^6	0.25×10^6
1.0	312	1.04×10^6	0.26×10^6
1.0	273	1.15×10^6	0.29×10^6
1.0	301	1.30×10^6	0.32×10^6
1.0	244	1.95×10^6	0.49×10^6
1.0	212	2.08×10^6	0.52×10^6
1.5×10^6	301	2.40×10^6	0.6×10^6
1.5	218	2.47×10^6	0.62×10^6
2.0×10^6	218	1.96×10^6	0.49×10^6
2.0	218	2.0×10^6	0.50×10^6
2.0	273	2.05×10^6	0.51×10^6
2.0	301	2.63×10^6	0.66×10^6
2.0	244	2.70×10^6	0.68×10^6
2.5×10^6	239	2.70×10^6	0.68×10^6

In most cases the burnout was detected by thermocouples located a considerable distance upstream from the test section outlet.

At heat fluxes approaching the burnout heat flux, large fluctuations in test section temperatures were observed. These fluctuations were indicative of local unstable film boiling conditions.

Preliminary comparisons were made between the above information and some limited data obtained with an earlier test section of similar dimensions but generating a uniform heat flux. These comparisons indicated that at identical coolant mass velocities and inlet qualities, the peak heat flux at burnout conditions for the "banded" test section was approximately 60% greater than the burnout heat flux for the uniformly heated rod. The minimum heat flux was about half of the burnout heat flux for the uniform rod.

2. Plutonium Ceramic Fuels Research

Mixed Plutonium-Base Carbides. The pseudo binary PuC-ZrC system is being investigated in order to develop compounds that are more

DECLASSIFIED

refractory than PuC. Samples prepared by tungsten arc melting of mixtures of PuC and ZrC exhibited melting points ranging from 2900 C for 10 mol% PuC to 2260 C for 63 m% PuC.

Plutonium Cermets. Development of methods for efficiently preparing PuO_2 cermet pellets by pneumatic impaction is continuing. A split die for preparing six pellets at once was fabricated and various parting materials to prevent the sections of the die from being cold welded together during impaction were evaluated. Steel pieces, flame sprayed with Al_2O_3, readily parted after impaction. Split dies are now being fabricated with a 0.5 to 1 mil flame sprayed coating of Al_2O_3.

A pneumatically impacted PuN-50 v/o W cermet pellet was vacuum heat treated for four hours at 1400 C and four hours at 1500 C. After each heat treatment the specimen lost 9% of its mass. Density measurements and metallographic examinations suggest an increase in closed porosity; perhaps as a result of PuN decomposition. The PuN in this specific sample was not uniformly dispersed in the tungsten matrix.

Preliminary experiments were conducted to evaluate formation of a PuS-W cermet by heating Pu and WS_2 to 1300 C in vacuum. The reaction product, formed on the surface of a Pu rod, is being analyzed by x-ray diffraction.

Cermets have also been prepared by pneumatic impaction of PuC particles in matrices of Ta, Va, and W.

PuO_2-MgO Phase Equilibria. PuO_2-MgO phase equilibria studies were completed. PuO_2 and MgO are insoluble in the solid state and form a binary eutectic system containing no intermediate compounds. The eutectic point lies at PuO_2-43 mol% MgO at about 2100 C. For comparative purposes a UO_2-53 mol% MgO specimen, reported by others to be the eutectic composition, was melted in the same manner as the PuO_2-MgO specimens. The melting point determined for this UO_2-MgO eutectic composition, 2275 C, agrees well with that published in the literature. Metallographic examinations comparing these two eutectices reveal identical microstructures.

Irradiation of Plutonium Carbide. An irradiation specimen of PuC (GEH-14-408) was discharged from ETR on August 5, 1963, with approximately 5% plutonium atom burnup (approximately 45,000 MWD/ton).

3. Ceramic (Uranium) Fuel Research

High Heat Flux Fuel Rod. A UO_2 fuel capsule designed to generate 855 watts/cm^2 (2,700,000 Btu/(hr)(sq ft)) was fabricated for irradiation in the ETR as a part of a study of improved, high rate,

heat removal technology. The capsule, clad with 1.18 cm (0.465-inch) OD by 0.099 cm (0.039-inch thick) molybdenum, contains UO_2 enriched to 50 w/o U-235 in U. The outer surface of the cladding tube is threaded [25.2 threads/cm (64 threads/inch), 0.023 cm (0.009 inch) deep] to improve heat transfer characteristics. Calculations indicate that 93% of the fuel radius will initially be molten during irradiation and the maximum core temperature will be ~ 4100 C.

Burst tests of a sample of the cladding for this capsule demonstrated that the thread did not weaken the cladding. Corrosion tests of the molybdenum cladding material showed that the corrosion rate will be approximately 4.1×10^{-4} cm/day (1.6×10^{-4} inch/day) under the water conditions anticipated for this test (260 C, pH of 10, trace H_2O_2 added).

Radiation Damage to UO_2. The lattice constant of UO_2 irradiated to 0.056 a/o burnup decreased approximately 0.065% after annealing for eight hours at 960 C in vacuum. These data are consistent with data reported in France and the United Kingdom.

Structure Development in UO_2 During Irradiation. Irradiation of three tungsten marker capsules in the MTR was satisfactorily completed. Conditions were specified to produce three different degress of core melting. These capsules will be returned to HAPO for post-irradiation examination. Six other tungsten marker capsules will be irradiated for additional cycles.

Comparison of Ceramographic Techniques. Mounted, polished, and etched sections of irradiated UO_2 fuel rods were exchanged with APED-San Jose to compare ceramographic techniques used at the two sites. Agreement among investigators at HL and APED of interpretations of irradiated fuel microstructures (and thus, thermal performance of nuclear fuel elements) has been hampered by different structures seen at the two sites in presumably similarly irradiated specimens. This cooperative exchange of specimens will resolve whether the differences are real or artificial, and will promote better agreement and understanding among investigators.

Materials and Information Exchange. In response to a request, one kilogram of pneumatically impacted UO_2 was sent to the Dounreay Establishment, Scotland, for vibrational compaction studies. Characterization tests (see table) show that the UO_2 was nearly stoichiometric and quite pure.

Characterization Test	Value	Analytical Technique
O/U atom ratio	2.0016	Coulometric titration (controlled potential)
N	140 ppm	Kjeldahl (dissolution in HCl; H_2SiF_6; $CuSeO_4$)
Cl	32 ppm	Pyrohydrolysis; coulometric titration to an amperometric end-point
F	2 ppm	Pyrohydrolysis; colorimetric analysis by lanthanum-alizarin
C	37 ppm	Combustion (LECO)
H_2O	70 ppm	Solids moisture analyzer (coulometric-electrolysis method)
Density	10.87 g/cc	Displacement - CCl_4
Metallic impurities	< 200 ppm	Emission spectrography Ga_2O_3 carrier carrier method; Co as internal standard.

Optical Properties of ThO_2. The infrared absorption spectrum for polycrystalline ThO_2 is similar to the spectrum of UO_2. The ThO_2 becomes more transparent with increasing wavelength from 2 to 9 μ. Absorption bands are observed at 9.6, 11.6, and 13.6 μ. The ThO_2 is significantly more transparent than UO_2.

Cermet Fuel Plates Containing Tungsten Coated UO_2 Spheres. Thin plates, approximately 20 mils in thickness, were pneumatically impacted using tungsten coated UO_2 spheres and tungsten powder. The 20 v/o UO_2 fraction consisted of -270 +400 mesh UO_2 spheres coated with 10-15 microns of tungsten. The remaining tungsten fraction was 5 micron tungsten powder. Metallography shows good distribution of the UO_2 spheres and good bonding at the tungsten coating-tungsten powder interface. Pneumatic impaction also eliminated many pores initially present in the UO_2 spheres but did not cause the UO_2 spheres to deform significantly.

Evaluation of UO_2-W Cermet Fuel Grids. Photomicrographs made of UO_2-W cermet fuel grids, pneumatically impacted in an expandable mandrel, revealed extensive cracking in the cermet. A definite pattern of cracking in the vicinity of the web joints was observed. Most of the UO_2 particles (-60 +200 mesh) in the path of the cracks were

cleaved. Apparently the cracks were formed after impaction of the cermet. The difference in thermal expansion between the mild steel mandrel and the cermet could cause the observed cracks; as the steel cools it would tend to pull away from the cermet. Because the cermet is bonded (cold-welded) to the steel mandrel, the cermet is placed in tension and cracks are formed to relieve the resultant stresses.

Short-Term Irradiation of Cermet Fuels. Two capsules containing tungsten - 50 w/o UO_2 cermets were irradiated for two and four hours, respectively. The vacuum insulated, tungsten-clad cermets included 4.1% enriched UO_2.

Post-irradiation examination revealed fission gas inside each outer Zircaloy capsule. One tungsten cladding had developed a large longitudinal crack and the other a microscopic, transverse crack. The tungsten surface temperature during irradiation is calculated to have been greater than 2100 C.

Physical Testing of Cermets. The effect of UO_2 concentration on the strength of UO_2-tungsten cermets is being evaluated by modulus of rupture and diametral compression tests. Uranium dioxide additions of 0, 10, 20, 30, 40, 50 and 64 v/o in pneumatically impacted cermets are being evaluated.

The diametral compression test consists of loading a cylinder in such a manner that a diametral plane is subjected to tensile stresses. This test can only be applied to brittle ceramic materials such as glass, magnesia, alumina, and electrical porcelain. Preliminary experiments indicate this test is applicable to pneumatically impacted UO_2-W cermets. The modulus of rupture test is a simple bend test. Results of both tests are reported as tensile failure in psi.

Values obtained from modulus of rupture tests on a 20 v/o UO_2-tungsten cermet are:

$\bar{x}$ of 5 samples	45,700 psi
$\bar{x}$ of 11 samples	48,400 psi.

Values obtained from diametral compression tests on the cermet are:

$\bar{x}$ of 10 samples	18,400 psi
$\bar{x}$ of 11 samples	16,700 psi.

While the modulus of rupture test yields strength values which are two and one-half to three times larger than the diametral compression test, a variation in value is to be expected. Even though

both tests produce tensile failures, the arithmetical mean highest value, and the lowest value from each test method are functions of the volume of material under stress the stress distribution of the test and sensitivity of the test of microstructure and surface effects.

Cermet Fuel Volatilization. A 50 w/o W-UO_2 cermet exhibited vaporization of UO_2 to a depth of several "UO_2 particle diameters" below the surface during heating in a purified H_2-A atmosphere between 2200 and 2500 C for four hours. Apparent interconnecting porosity allowed the UO_2 to escape from the tungsten matrix by sublimation.

Magnetic Force Resistance Welding Studies. A cap-to-tube weld was made between two pieces of tantalum. Destructive examination of the bond indicated larger recrystallized grains in the weld area than in the parent material, but other joint characteristics are good. Further experiments will be made to reduce the size of the recrystallized grains.

Plasma Electron Beam Furnace. A furnace using a new concept of electron beam heating recently developed at the GE Advanced Technology Laboratory, the plasma electron beam, was built and put into operation. This device allows high temperature heating of materials in low pressure inert, reducing, or oxidizing gases.

4. Basic Swelling Program

Irradiation Program. One general swelling capsule reached its goal exposure and was discharged from the reactor. It operated continuously at the preselected control temperature of 625 C (1157 F). Two additional capsules were charged and await reactor startup before applying external heat to bring the capsules to control temperature.

The assembly and bench testing of two more general swelling capsules is completed and the capsules are being tandemized for future charging into a single test hole.

The concept of employing a flexible bellows to transmit pressure from He to NaK for use in the in-reactor controlled pressure, controlled temperature swelling capsule was tested in a laboratory mockup experiment. Pressure was transferred without loss through the bellows indicating that the concept is sound. Equipment is being modified and parts are being fabricated to construct a trial controlled pressure capsule for test purposes.

Post-Irradiation Examination. Optical and electron metallography have been completed on three enriched (1.44 a/o U-235) uranium specimens irradiated to 0.5 x 10^{20} fissions/cc (0.10 a/o B.U.) at

respective temperatures of 640 C (1184 F), 625 C (1157 F), and 575 C (1067 F). The first two specimens were as-extruded whereas the third sample had been beta-quenched prior to irradiation. The metallography qualitatively corroborated the measured volume changes of 8, 13 and 28%, respectively. Both of the as-extruded specimens had extensive porosity at the grain boundaries with small pores scattered throughout the grains, but the 625 C (1157 F) specimen exhibited much more porosity than did the 640 C (1184 F) sample. These results are in general agreement with the previous observations made with specimens of natural enrichment and serve to emphasize the extreme dependence of the volume change on the irradiation temperature becoming less as the temperature is increased in the 575 C (1067 F) to 650 C (1202 F) range. Further evidence of this has been observed in a longitudinal section of a half hollow cylinder of natural uranium irradiated to 0.5×10^{20} fission/cc (0.1 a/o B.U.) at 625 C (1157 F). Optical examination shows small pores located at grain boundaries in the central region of the sample. Near the ends porosity is more profuse at both grain boundaries and within the grains. It is believed that the ends operated at a lower temperature than did the center due to the heat transfer conditions in the capsule. Replicas of this specimen are being processed for electron microscope examination.

The results obtained with the enriched (1.44 a/o U-235) uranium specimens indicate that the volume change versus temperature curve is shifted slightly to higher temperatures. The shift is not large, however, and it would appear that a fission rate greater than the factor of two thus far employed is needed to illustrate the dependence of the volume change-temperature relationship on burnup rate.

Post-Irradiation Annealing. A systematic study of the behavior of highly irradiated [$\sim 2 \times 10^{20}$ fissions/cc (0.37 a/o B.U.)] clad and declad uranium as a function of post-irradiation, pulse annealing has been initiated. Replicas of the "as-irradiated" specimens and of specimens annealed at 600 C (112 F) for 30 minutes have been examined in the electron microscope. The "as-irradiated" specimen disclosed a typical swirled microstructure with scattered pores having diameters of < 0.1 micron. Examination of replicas from the edge and core indicated a somewhat higher density of these small widely scattered pores at the edge of the uranium. After the annealing treatment pore diameters had increased by several orders of magnitude in both clad and declad specimens. Some areas of the sample have recrystallized and a greater portion of the porosity is located at the newly formed grain boundaries. There is no apparent difference in pore size or density from the edge to the core of the sample after annealing. No effect due to cladding restraint was apparent. Replicas from specimens which were annealed for longer periods of time are in process.

Thorium-Uranium. Three nonirradiated tensile specimens (Th - 1, 4 and 5.4 w/o U) were tested to failure using the same procedures employed with the irradiated counterparts, i.e., in-cell. Instron testing at room temperature with a 0.005-inch per minute cross-head speed. As expected, the specimens exhibited a yield point of about 2100 Kg/cm^2 (30,000 psi) and about 30% total elongation. These specimens are being processed for metallography for comparison with the irradiated specimens.

Second-Phase Distribution in Dilute Alloys of Uranium. Finely dispersed particles in dilutely alloyed uranium are being delineated by refined and modified metallographic techniques. Three samples of dingot uranium with Fe-Si additions have been successfully electropolished and electroetched in a bath consisting of five parts ethylene glycol, five parts phosphoric acid and eight parts ethyl alcohol.

Electropolishing is performed at a potential difference of 18 volts for a period of 60 seconds; electroetching is performed at a potential difference of five volts for a period of 120 seconds. Replicas prepared after a series of these repetitive polish-etch treatments revealed that best results were obtained on samples which had been subjected to three repetitive polish-etch treatments. Twelve random photographs at 15,000X were obtained for quantitative metallographic analysis of each sample.

The greatest difficulty encountered in the sample preparation occurs during the hand polishing steps. The specimen surface often becomes distorted at this stage by a layer of flow metal. Subsequent etching of such a sample will produce a very rough, granular background which obscures the intermetallic particles present in the matrix. In an effort to improve the efficiency of the polishing step, attention has been given to Syntron polishing methods. Samples taken by hand through 4/0 grit emery polishing paper are polished on three successive Syntron polishing wheels covered with (1) Metcloth, Linde A abrasive, and 2% chromic acid; (2) Microcloth, Linde A abrasive, and 2% chromic acid; (3) Gamal polishing cloth, MgO (electronic grade) abrasive, and distilled water. Preliminary results indicate that this method of polishing may be superior to the previous hand polishing techniques.

The remaining 27 dingot uranium samples and four additional samples of ingot uranium are being prepared for quantitative metallography using the techniques outlined above. The metallurgical history of the four ingot uranium specimens is as follows: U + Fe-Si and U + Fe-Al, each having had a normal beta heat treatment and then aged 24 hours at 525 C (941 F); the same two alloys, each gamma-treated at 1000 C (1832 F) for 24 hours, water-quenched, and then aged 24 hours at 525 C (941 F).

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. To date, complete shipments of 304, Inconel 600, Inconel X-750, A212B, and A302B have been received. Partial shipments of AM-350, A-286, 406, and Zircaloy-2 were received during the past month. One heat of AISI 348 S.S. was accepted, and the material is presently being processed into sheet, plate, and bar stock.

One additional quadrant of nickel and iron base alloys including Hastelloy N, Inconel 702, R-235, Inconel 625, Inconel 718, and R-27 was discharged from the ETR hot water loop facility. These specimens achieved an estimated neutron exposure of 1.8×10^{20} nvt. These specimens together with two additional quadrants previously discharged will be sent to Radiometallurgy for testing. To date, 29 specimens of these alloys have been tested. Seventeen control tensile specimens were also tested this past month.

Specimens of Inconel 600, Incoloy 800, Inconel 625, AISI 406, and Hastelloy X-280, which are candidate alloys for fuel cladding in the BONUS nuclear superheat reactor, have been fabricated, characterized, and assembled into graphite boats for irradiation. These alloys will be irradiated at high temperatures in a gaseous environment to four different exposures. Post-irradiation tests will indicate any changes in mechanical properties due to the environment and exposure.

In-Reactor Measurements of Mechanical Properties. Two additional in-reactor creep tests on 20% cold worked Zircaloy-2 were started. At 320 C (608 F) and 35,000 psi (2450 kg/cm^2) stress a creep rate of 1.7×10^{-5}/hr was found, and at 365 C (688 F) and 25,000 psi (1750 kg/cm^2) stress a rate of 2.6×10^{-5}/hr was found.

The ex-reactor tests with the same test conditions as above have not yet accumulated enough test time to quote a creep rate, but at comparable times the ex-reactor rates are less than the in-reactor rates, which is expected.

On the basis of an Arrhenius-type plot of creep rates versus the reciprocal of the test temperature, it was predicted that the creep rate of the 365 C (688 F) test during a reactor outage would increase to about 4×10^{-5}/hr in a short time due to rapid annealing of creep inhibiting defects. This prediction was confirmed when the creep rate increased almost immediately after reactor shutdown to 3.6×10^{-5}/hr.

During reactor outages the Arrhenius plot predicts that the 320 C (608 F) test would increase in rate by a factor of 10. Due to the lower annealing temperature, the increase in rate takes longer to occur and after about 120 hours the rate of the 320 C (608 F) test was 3.5×10^{-5}/hr. At this time, the test was stopped to prevent breakage of the specimen. The data from these two tests appear to confirm the creep model proposed on the basis of earlier tests.

Irradiation Effects in Structural Materials. 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 is being given to the determination of mechanical property changes produced in metals by irradiation at elevated temperatures.

During the month a total of 214 specimens were tested, which included 189 irradiated bend test specimens, notched tensile specimens, and smooth tensile specimens. Tensile specimens in the as-fabricated condition for a variety of stainless steel and nickel base alloys accounted for the other 25 tests performed. At the conclusion of ETR Cycle 56, seven quadrants containing 114 irradiated tensile and bend test specimens were discharged from the G-7 loop. Three capsules containing 24 tensile specimens were also discharged from the G-6 position.

A hanger assembly for placing corrosion specimens in the G-7 loop was fabricated and installed during ETR Cycle 56 shutdown. Zircaloy-2 corrosion coupons were placed in the upstream portion of the pressure tube in an out-of-flux position. Modifications to G-7 loop piping in the sub-pile room have been made to provide for the insertion of an out-of-flux test assembly in the downstream portion of the loop. The purpose of these test assemblies is to provide a means of obtaining corrosion rates on materials exposed in both in-flux and out-of-flux positions of the hot water loop.

Over 170 irradiated dosimeter wires were shipped to Hanford and counted. These iron, nickel, titanium, and aluminum-cobalt wires came from both central and quadrant flux monitors which had been irradiated in the ETR. The dosimeters have been counted previously by personnel of Phillips Petroleum Company in Idaho Falls, Idaho. These dosimeters were counted at Hanford to establish an additional basis of comparison as to the compatibility of flux measurements made at different sites. The prime purpose of this comparison is to determine more precisely the relationship that exists between materials irradiated to similar exposure levels in different type reactors. Preliminary examination of these data indicates that there is good agreement between the results obtained by the two sites.

Flux data from a central monitor exposed in the G-7 loop for the duration of ETR Cycle 54 has been received and analyzed. The fast fluxes determined from iron, nickel, and titanium dosimeters in this monitor are in better agreement with one another. The data obtained from the central monitor exposed in ETR Cycle 54 deviates by as much as 20% in the higher flux regions of the G-7 loop when compared with the average of data obtained from central monitors exposed in ETR Cycles 45 through 51. However, the flux values calculated, using new cross section data are known to be 5% lower than those previously reported. Therefore, all values are still within the ± 14% scatter range previously reported. Future data obtained from both central and quadrant dosimeters will continue to reflect the new cross section changes and will necessitate the correction of data obtained previously.

Several special flux monitors containing dosimeters of iron, nickel, titanium, and an aluminum - 0.1% cobalt alloy, were inserted in weep holes in the support plate for the G-7 pressure tube. These weep holes are provided for gamma heat dissipation from the support plate. The dosimeters are continuous wires extending the full length of the reactor core, enclosed in heavy wall aluminum tubing. These dosimeters will monitor the flux intensities encountered outside the G-7 pressure tube during ETR Cycle 57.

Tensile specimens of AISI 348 and 304 stainless steels irradiated in the ETR G-7 loop to exposures of 1.2×10^{21} nvt (fast) have been tested. Additional tests have been performed on control specimens of these materials which were exposed for the same time and temperatures in the out-of-reactor hot water loop.

Control specimens of AISI 304 stainless in the annealed condition show an increase of 6,700 psi (47.1 kg/mm^2) in 0.2% yield strength and a decrease in total elongation of 7.3% when exposed in the out-of-reactor loop for approximately 3,265 hours at temperatures greater than 200 C (392 F). The corresponding values for material containing 25% cold work was 9,000 psi (63.3 kg/mm^2) and 6.9%, respectively. Similar changes of the same magnitude were noted for both the annealed and 25% cold worked AISI 348 stainless with the exception of yield strength in the 25% cold work condition. The increase in 0.2% yield strength was 16,700 psi (117.3 kg/mm^2). When these contributions due strictly to thermal effects are accounted for, data acquired at two different irradiation temperatures are comparable.

Irradiated specimens of AISI 304 and 348 in the annealed condition show a four-fold increase in yield strength after exposures up to 1.1×10^{21} nvt (fast). Uniform elongation is reduced from 78.7 and 58.0% for 304 and 348, respectively, to approximately 1.0% after irradiation.

Comparison of data acquired from irradiations performed at 280 C (540 F) with that obtained at a lower irradiation temperature tends to indicate that damage due to neutron flux and environment accumulates at a faster rate at elevated temperatures for these materials. The most pronounced changes occur in the material exposed in the annealed condition.

Tests were performed on 72 notched tensile specimens of AISI 348 stainless steel which had been irradiated in the ETR G-7 loop to exposures of 2.2×10^{20} nvt (fast). Specimen conditions included annealed, 10, 20 and 40% cold work levels for both the rolling and transverse directions. Data from the 40% cold worked specimens indicated that there was insufficient ductility to prevent crack propagation from the notch prior to net section yielding. A slight amount of ductility was exhibited by the 20% cold work specimens for the highest exposure levels examined and the ratio of notch strength to ultimate tensile strength was slightly greater for these specimens. Ductility decreases with increasing exposure and an extrapolation of existing data indicates that the 20% cold worked specimens should begin to exhibit brittle failure at exposures of approximately 1.0×10^{21} nvt (fast).

The annealed and 10% cold worked specimens exhibit considerable ductility at the exposure levels examined and the notch strength was 15 to 25% greater than the ultimate tensile strength. Although decreasing ductility with increasing exposure is also indicated for these conditions, it is doubtful that the annealed and 10% cold worked specimens will exhibit a nonductile type failure for any reasonable exposure. Control notch tensile specimens exposed in the out-of-reactor hot water loop for similar times at temperature show sufficient ductility for all levels of cold work examined to prevent crack propagation prior to net section yielding.

It should be noted that specimen size and thickness play an important role in determining whether a material fails in a brittle manner. Specimens in this examination were fabricated from sheet stock 0.050-inch thick. All other conditions being equal, an increase in the specimen thickness would cause a corresponding decrease in ductility.

Damage Mechanisms. The objective of this program is to establish the nature of the interactions 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.

During this period, the iron-titanium alloys previously prepared and fabricated were examined by tensile testing. These alloys

which were prepared by adding titanium in trace quantities ($4\frac{1}{2}$ times the interstitial impurity content) to Ferrovac E and MRC zone-refined iron, showed about the same ultimate strengths as the starting material. The alloys, however, showed no trace of a yield point or inhomogeneous deformation. The interstitial impurities were presumably tied up as stable titanium carbides, nitrides and oxides and hence were not free to pin the dislocations which would produce the usual cottrell type yield point. Perhaps even more surprising was the absence of a yield point of the type described by Gilman and Johnston which is generated by the rapid multiplication of dislocations as the tensile sample is strained. The absence of this type yield point indicates there were a great number of mobile dislocations present in the sample prior to deformation.

Tensile tests of zone-refined iron obtained from Battelle Memorial Institute and Ferrovac E iron were conducted in a liquid nitrogen bath. Both samples showed an increase in lower yield stress to about three times the room temperature value. The Ferrovac E retained a fair amount of ductility and a reduction in area of about 63%, whereas theBMI sample broke after about 8% strain with a reduction in area of only 6%. Both these samples had reductions in area of about 95% when deformed at room temperature. The fracture surface of the BMI sample indicated intergranular fracture had occurred. Such behavior has been observed by other workers and attributed to segregation of an oxide phase at the grain boundaries.

A series of strain-rate tests was initiated to investigate the possible existence of a rate sensitive deformation mechanism. Four points have been obtained to date at rates varying from 3.3×10^{-5} to 3.3×10^{-3} sec^{-1}. Initial results indicate the possibility of a different deformation mechanism at the higher strain rates.

Environmental Effects. ETR G-7 loop water analyses on samples taken daily during the middle of the Cycle 56 operating period show oxygen concentrations ranging from 0.07 to 0.7 ppm; pH was controlled in the neighborhood of 9 and ranged from 7.2 to 9.6. Water conductivity values varied around 5 μ mho cm^{-1} and were reasonably well within the range expected for high-purity demineralized water adjusted to loop pH values with lithium hydroxide. Gross radiation readings taken at the surface of one-quart water volumes immediately after sampling were low, ranging from 20 to 150 mr/hr.

Design of the G-7 loop downstream out-of-flux specimen exposure chamber was completed during the report period and is currently undergoing acceptance review by MTR-ETR engineering.

In-reactor weight gain results for 24 Zircaloy-2 coupons irradiated in quadrants 85 and 86 during ETR Cycles 50 and 51; and 50, 51 and 52, respectively, were obtained during the report period. Quadrant 85 was exposed to a maximum water temperature of 271 C (520 F) for 22.6 days in an estimated fast neutron flux of 1×10^{14} nv (> 1 Mev) while quadrant 86 was exposed to a mean maximum water temperature of about 274 C (525 F) for 41.4 mg/dm^2 days at the same neutron flux intensity. Average weight gains of 49.7 mg/dm^2 and 59.9 mg/dm^2 continue to show order-of-magnitude higher average corrosion rates in the active zone of the ETR as compared with rates obtained in the absence of irradiation. Visual examination of the coupons of quadrant 85 showed surfaces similar to those of irradiated coupons described previously whereas extensive areas of quadrant 86 coupons showed a fairly rough dull grayish appearance. Oxide in these areas appeared to be slightly thicker than adjacent surfaces.

Preliminary results from the 400 C (752 F), 1500 psig steam autoclave test of samples previously irradiated in the ETR and unirradiated specimens of Zircaloy-2, indicate that the initial oxidation rate of abraded and etched irradiated material may be considerably higher than for unirradiated material similarly prepared. Fourteen-day average weight gains were approximately 50% higher for irradiated material, but the incremental weight gains during the second two-week exposure were similar for irradiated and nonirradiated samples.

Gas Loop Studies. A successful method has been developed for fabricating the large castable Fiberfrax insulating blocks to be used between the hot helium containing superalloy inner pipe and the cooler pressure bearing stainless steel outer pipe on the model gas loop. A tendency for the Fiberfrax block to severely crack when thermal cycled has been largely overcome by using stainless steel wire and mesh as a reinforcing material. In thermal conductivity tests made in air, two blocks of varying density both provided about a 982 C (1800 F) temperature drop through four inches of thickness from 1121 C (2050 F) on the internal surface to about 121 C (250 F) on the outer surface. Experiments are now being initiated to determine the outgassing and load bearing characteristics of the Fiberfrax insulating blocks at service temperatures expected in the model loop.

Insulation inside the test section of the model gas loop is to simulate that contemplated for use in the ATR gas loop test section. The insulation should, therefore, be removable and decontaminable as well as being compatible with an ultra high purity helium system.

Metallic foil, pyroceramic materials, and silicon carbide foam have been selected as possible materials. The metallic foil type

insulation appears to be the best candidate, and it will be used in the initial model gas loop test section. However, the test section is designed with one interchangeable section to permit a comparative evaluation of the other materials. All materials for testing have been ordered.

The existing design of the tensile specimen holder was reviewed and a redesign of the holder incorporating fewer parts and easier fabricability was made. Final drawings of the redesigned holder are now being made. Concurrently, further investigation of the best materials for construction of the holder is being made through a literature search.

One of the expected problems associated with a high temperature, high purity helium system is diffusion bonding between the tensile specimens and the holder. Preparations are now being made for evaluation of methods of minimizing this problem.

Radiation-Damage Exposure Units. Suitable neutron-exposure units are necessary (1) to permit correlation of radiation damage test data obtained in different reactors and (2) to permit valid application of test data obtained under certain conditions to reactor systems having quite different neutron spectra. A criterion to define such exposure units has been proposed and a test has been made in the case of several materials.

The exposure unit proposed is the number of neutrons per cm^2 having energies above a specified energy limit E_L. The energy limit must satisfy the condition that the integrated flux above E_L be proportional to gross-displacement production for the spectra under consideration. This condition requires that E_L be chosen so as to make the quantity

$$\frac{\int_0^\infty N(E)\ \sigma(E)\ \phi(E)\ dE}{E_L \int^\infty \phi(E)\ dE}$$

constant for a given material for the spectra in which radiation effects are to be compared. $N(E)$ is the number of displacements produced by a neutron with energy E, $\sigma(E)$ is the energy-dependent scattering cross section of the material, and $\phi(E)$ is the neutron spectrum. Although there is no a priori reason for the existence of an E_L which would generally satisfy these conditions, for a variety of spectra sufficiently wide to be useful, it has been found that the above conditions can be fulfilled.

Displacements were calculated for graphite in five different neutron environments: at the center of a fuel element, at the edge of the coolant-water annulus and at the cell boundary (10 cm from the fuel)

in a graphite lattice; in a test position in the GETR; and in an unmoderated U-235 fission spectrum. In all instances it was found that the displacement rate was linearly related within a few percent to the neutron flux in excess of E_L, if E_L was chosen as 0.067 Mev. In contrast, if a value of 1 Mev was used, the ratio of the displacement rate to flux in excess of that energy was found to vary by a factor of 1.5 for the GETR facility and 2.5 for the unmoderated spectrum in comparison with that of a well moderated spectrum.

The same calculations were made for other elements than carbon in these five neutron environments, and it was found that an E_L could be chosen for each element which showed a linear relationship between displacement rate and neutron flux above E_L. E_L varied monatonically by greater than a factor of 10 in going from beryllium to tungsten.

Since many of the displaced atoms will be annealed during irradiation, the persistent damage will depend upon a number of environmental factors as well as on the material. These factors are best correlated by investigators familiar with radiation effects in the material of interest. However, the gross number of displaced atoms provides a basis of correlating similar experiments conducted in other types of reactors, or applying the data to predict the irradiation behavior of the material.

6. Gas-Cooled Reactor Studies

EGCR-Graphite Irradiation. The sixth capsule, H-3-6, in the series of long-term irradiations of EGCR graphite is operating satisfactorily in the fourth and last cycle of irradiation. It is scheduled for removal from the GETR during the week of August 19. At that time the maximum neutron exposure to samples irradiated in all six H-3 capsules will be approximately 16×10^{21} nvt, $E > 0.18$ Mev.

Contraction of Nuclear Graphite. Samples of five nuclear graphites which have been used in the construction of Hanford reactors were recently discharged from a hot test hole at C Reactor with a total exposure of 15,950 MWD/AT. Length changes had been measured previously on this series of samples at exposure levels of 3025, 5110, 7585, and 11,760 MWD/AT. The irradiation temperature is in the range of 500 to 600 C. The rates of contraction for the last exposure increment between 11,760 and 15,950 MWD/AT and the ratios of these rates to the rates for the previous exposure increment, 7585 to 11,760, are given in the following table.

Graphite-Type Orientation	CSF		KC		CSGBF		WSF		[illegible]	
	T	L	T	L	T	L	T	L	T	L
Total contraction after 15,950 MWD/AT, %	0.242	0.308	0.066	0.330	0.214	0.375	0.215	0.337	0.244	[illegible]
Contraction rate between 11,760 and 15,950 MWD/AT, %/1000 MWD/AT	0.018	0.025	0.008	0.033	0.028	0.032	0.019	0.027	0.015	0.027
Ratio of contraction rate during last exposure to contraction rate during previous exposure	1.06	1.14	1.00	1.65	1.56	1.23	1.27	1.13	0.71	1.11

The contraction rates of all graphites have increased for all samples parallel to the extrusion axis. Transverse contraction remained constant for KC graphite, which has the lowest total contraction of all the test materials.

The transverse contraction rate of SGBF graphite decreased; however, this graphite had the highest total transverse contraction. The transverse behavior of SGBF graphite is perhaps explainable from the behavior noted in an experimental series of graphites having different graphitizing temperatures. Experimental graphites produced at lower temperatures contract at higher initial rates than graphites processed at higher temperatures but show a decrease in these contraction rates at lower exposures. The SGBF graphite was processed at 2450 C while the other nuclear graphites, with exception of CSGBF, were processed at 2700 to 2800 C. Thus, the higher initial contraction rate and decrease in transverse contraction rate of SGBF may be related to the lower processing temperature. However, CSGBF graphite, also processed at 2450 C, has not contracted as rapidly as SGBF and does not show a decreased contraction rate for the last exposure period, so this is not the complete explanation.

These samples will be recharged for further study of the variation of contraction with exposure.

Graphite-Water Vapor Compatibility. The oxidation of samples of TSX graphite is being studied in a helium stream containing small amounts of water vapor in order to determine the reaction rates. Previous work has established that the ultimate rates of oxidation of graphite are different from those initially observed. Therefore, the samples (0.43-inch diameter by 2-inch long) were pretreated by outgassing for 16 hours in a vacuum and then oxidized 4% weight loss in air at 570 C in order to obtain more representative long-term oxidation rates. The sample was then exposed to a helium stream containing 6000 ppm of water vapor. At each temperature studied the oxidation was continued for approximately 24 hours. Linear weight losses vs time were observed. The results are given below.

Temp., °C	Experimental Rate, g/g/hr	Calculated Rate (HW-75072), g/g/hr
760	9.60×10^{-6}	1.23×10^{-5}
778	1.55×10^{-5}	1.78×10^{-5}
810	4.14×10^{-5}	3.10×10^{-5}
838	6.89×10^{-5}	4.60×10^{-5}

The experimental rates differ somewhat from those calculated in HW-75072. Since the latter are based on a different grade of graphite, the agreement is probably satisfactory.

The apparent activation energy for the four temperatures is 59 kcal/mole. However, the plot of logarithm of the rate vs 1/T is not linear, perhaps because inhibition by water has not been accounted for.

A refrigeration unit has been added to the system to more accurately control the water vapor concentration. The effect of partial pressure and the variation in oxidation rate from sample-to-sample will be studied.

Graphite-Water Vapor Reaction Under Gamma Irradiation. Study of the effect of radiation on the rate of graphite oxidation in helium containing low partial pressures of water vapor has continued. At 680 C and 300 to 400 ppm water the presence of a 3.6×10^6 r/hr Co^{60} gamma field increased the reaction rate by about 25%. The reaction is, therefore, primarily thermally controlled under these conditions.

Inhibition of Graphite Oxidation. The oxidation rate of graphite increases with the amount of carbon burned off because pores are gradually opened exposing a larger surface for reaction as oxidation proceeds. For example, at 650 C the oxidation rate in air is given by

$$\text{Rate } (hr^{-1}) = 0.0074 + 0.0015\ B$$

where B is the percent weight loss at any time for the range B = 5 to 15%. At 425 C the rate reaches a constant value of 0.0027 hr^{-1} after a few tenths percent burnoff.

The rate at 615 C in air containing 1/2% of the oxidation inhibitor, CF_2Cl_2, is lower and is given by $R(hr^{-1}) = 0.0037 + 0.00022\ B$ for B in the range 1 to 5%. At 570 C the rate is 0.0017 hr^{-1} for B > 1/2%.

The oxidation rates in air plus 1/2% CF_2Cl_2 were measured from 570 to 660 C. The specific rate constant per unit surface area ($g/m^2/hr$) plotted versus reciprocal temperature gives an activation energy of 46 kcal/mole. This is within 10% of the value normally obtained in the absence of CF_2Cl_2.

The above rates were obtained after the sample surface had been cleaned by heating to 570 C for at least one hour in a stream of argon containing 6% CF_2Cl_2. If the surface is not cleaned, large transient effects on the oxidation rate are noted. Thus, the effectiveness of CF_2Cl_2 as an inhibitor depends on prior conditions as well as the temperature and gas composition.

7. Boronated Graphite Studies

2-C Irradiation Facility. Calculations were made using a geometry such as may exist during tests of boronated graphite capsules in the 2-C facility at KW Reactor. These calculations included determinations of hydraulic and thermal hydraulic variables such as flow rate for turbulent flow, pressure drop, the film heat transfer coefficient, capsule-wall temperature, and elevation of the capsule-wall temperature above saturation temperature for nucleate boiling.

Specific values obtained for the variables under study are as follows:

1. The flow rate for turbulent flow was found to vary from a minimum of 9.5 gpm to the maximum available flow rate of 30 gpm. This corresponds to a range in the Reynolds number of 4500 to 14,300.

2. The isothermal pressure drop along a concentric annular section was calculated to be 2.08×10^{-3} psi per foot of length at a Reynolds number of 4500 and ranged to 1.6×10^{-2} psi per foot of length at a Reynolds number of 14,300.

3. Calculations were made of the film heat transfer coefficient required for heat fluxes varying from 25,000 to 150,000 BTU/$hr\text{-}ft^2$, a bulk water temperature of 55 F, and through the

flow range under consideration. The coefficients thus calculated were found to be as low as 300 BTU/hr-ft^2-^{o}F and up to 600 BTU/hr-ft - F.

4. Capsule-wall surface temperatures under various conditions of heat flux and flow were found to range from as low as 110 F to a safe maximum of 300 F.

5. To insure that nucleate boiling would occur it was found that the wall temperature of the capsule should be higher by 20 F to 40 F than the saturation temperature of the fluid. This would dictate a wall temperature of 297 F to 340 F for system pressures of 40 psia to 70 psia.

With these data it is possible to determine safe conditions of loop operation.

Boronated-Graphite Irradiations. Calculations of temperatures in design studies have continued. The purpose of these calculations is two-fold: (1) to determine, on the basis of in-reactor heat transfer tests, the heat generation attributable to the (n,α) reaction; and (2) to develop the optimum design for the capsule to be used for full scale irradiation testing of graphite for the Fermi reactor. Some 700 cases have been calculated to date.

The first irradiation of a capsule for the determination of heat generation rates in boronated graphite took place during the month. This capsule contained 10 boronated samples of 8 w/o boron of the "black" and "grey" variety. Six thermocouple readings from various regions of the capsule were periodically recorded. Flux monitors were included to determine the exposure. Disassembly of the capsule is in progress. The results from this test should allow normalization of the heat generation rate to a specific incident-flux intensity. The capsule just irradiated will be followed with others in the next two months. All heat transfer capsules are intended to be dual purpose, i.e., give useful irradiation data in addition to capsule design data.

8. Aluminum Corrosion and Alloy Development

In-Reactor Loop for Corrosion Studies. The Design Change permitting installation of the C-1 process tube in the reactor is awaiting final approval.

Several questions arising from the installation of the C-1 Loop in C-Reactor were resolved during the month, as follows: (1) the restricting orifice after the rupture disc was removed; (2) the loop will be pressure-tested to a pressure $1\frac{1}{2}$ times the design pressure adjusted for temperature; (3) a restriction will not be placed after

the loop rupture disc; (4) a stress analysis of the loop piping is being performed; (5) the loop annunciator alarm has been altered in tone so it does not compromise the building evacuation system.

A safety analysis evaluated the consequences of failure of the backup accumulator to deliver cold water if the loop were dumped without previously scramming the reactor. In this case the reactor would scram on a low-flow trip, but the power generation before the reactor could be completely shut down is approximately that required to melt the fuel elements. To get a more reliable answer to this question, computer calculations are being made to determine the exact rate of drop-off of power at the location of the C-1 Loop following initiation of a reactor scram

Other problems which still have to be resolved prior to fueled operation are (1) finding a location satisfactory for dumping the loop, (2) providing three independent power sources for the scram relays, and (3) the adequacy of the Struther-Dunn relays.

9. Metallic Fuel Element Development

Irradiation of Thorium-Uranium Fuel Elements. Three tubular Zircaloy-2 clad thorium - 2.35 w/o U-235 - 1.0 w/o Zr fuel elements are completing their fourth consecutive cycle of irradiation in the ETR-P7 Loop. At the end of the fourth cycle the fuel elements will have achieved an exposure of 1.0×10^{20} fissions/cm^3 (2900 MWD/T). Maximum specific power is currently 63 watts per gram with a corresponding maximum fuel temperature of 555 C.

Thorium Fuel Defect Testing. Defect testing of Zircaloy-2 clad, Th - 2.5 w/o U - 1 w/o Zr core, fuel rod specimens has been initiated. The specimens are 1.3 cm OD x 5.1 cm long (0.523" OD x 2" long) with a 0.063 cm dia (0.025") by 0.076 cm deep (0.030") hole through the 0.063 cm (0.025") cladding The specimens are tested in 300 C water at 95 atmospheres pressure (1400 psig) Both as-extruded and heat-treated (750 C for one hour) specimens have been tested. In both cases hydrogen gas and finely divided oxide were noted issuing from the defect throughout the duration of the test. A small blister formed around the defect shortly after the piece reached temperature. This blister grew slowly until the test was terminated when it was approximately 1.04 x 1.51 x 0.185 cm high (0.410" x 0.595" x 0.073" high) for the as-extruded specimen and slightly higher and smaller in diameter for the heat-treated specimen. Both tests were terminated after 5.5 hours at temperature. Examination of sections through the defect area indicate no undercutting, but that the blister formed by tearing in the core material next to the Th - Zr-2 diffusion layer Preliminary testing has shown the Th - 2.5 w/o U - 1 w/o Zr alloy to be an order of magnitude

more corrosion resistant than unalloyed U in 300 C water.

Dispersed Phase Thorium Alloys. One phase of the metallic fuel development program for FY-1964 is the study of quaternary additions to thorium base, solid solution strengthened, Th-U-Zr ternary alloys for development of finely dispersed intermetallic compounds in the structure. The dispersed phases are expected to contribute to additional irradiation swelling resistance in these high strength, corrosion resistant alloys. Evaluation of the structure and structural thermal stability will be by optical and electron microscopy supplemented by hot hardness and corrosion testing.

The initial alloys being studied are aluminum or beryllium additions (500 to 2000 ppm) to Th - 2.5 or 5 w/o U - 2 or 5 w/o Zr alloys. Both series of alloys have been prepared as 100-gram arc melted buttons. This material will be hot-worked to provide sample stock for heat treatment. The alloy compositions and as-cast room temperature hardness values are shown in the following table. The aluminum additions appear more effective in second phase hardening than beryllium additions.

As-Cast Hardness of Thorium Base Alloys

Nominal Alloy Composition*

U w/o	Zr w/o	Al ppm	Be ppm	Hardness R_F
2.5	2.0	0	0	73
"	"	500		86
"	"	1000		90
"	"	2000		92
"	"		500	80
"	"		1000	81
"	"		2000	79
2.5	5.0	0	0	85
"	"	500		86
"	"	1000		86
"	"	2000		91
"	"		500	88
"	"		1000	86
"	"		2000	81
5.0	2.0	0	0	79
"	"	500		90
"	"	1000		86
"	"	2000		97
"	"		500	89
"	"		1000	77
"	"		2000	--
5.0	5.0	0	0	84
"	"	500		90
"	"	1000		88
"	"	2000		92
"	"		500	--
"	"		1000	--
"	"		2000	--

*Balance thorium.

Submicron Uranium Carbide Dispersion in Metallic Uranium. The formation of a submicron dispersion of a second phase material in uranium represents the most promising approach to imparting high swelling resistance to metallic uranium fuels during elevated temperature irradiation. To be effective, the dispersed phase must: be stable in the metal structure at elevated temperatures; not agglomerate to reduce the number of particles; be insoluble in the matrix; and not move to preferred locations in the structure such as grain boundaries.

Carbon is present at levels to approximately 700 ppm in essentially all uranium metal induction melted in graphite crucibles. In normally cast uranium this carbon is present as massive inclusions, either as the monocarbide or in conjunction with uranium nitride or oxide. The high melting point of the carbide (2390 C) and the low solubility of carbon in gamma phase uranium (110-185 ppm at the melting point) lead to the formation of primary carbide particles. The solubility of carbon decreases to 25-65 ppm at the gamma-beta uranium transformation, and carbon is essentially insoluble in both the alpha and beta uranium phases.

The uranium carbide thus formed in the melt remains essentially unchanged throughout fabrication and heat treatment, finally occurring as inclusions aligned in the direction of metal flow during fabrication. These inclusions are generally in the 10-25 micron diameter range.

When molten uranium containing dissolved carbon (1000 ppm is soluble at 1300 C) is rapidly quenched, as in the formation of uranium shot, the carbide is found precipitated as extremely fine platelets, less than one micron in width and up to 2-3 microns in maximum dimension.

Uranium shot, -20 +250 mesh, has been densified and clad with Zircaloy-2 by encapsulating the shot in Zircaloy-2 can components and coextruding at 11.5:1 and 13:1 reduction in area from 650 C preheat. The resulting clad rods, 0.310 inch or 0.590 inch diameter, have ± 0 001 inch clad thickness variation in the central portion of the extrusions, measured density of 18.89 gm/cm^3, and average carbon content of 400 ppm. The carbide particles in the extruded material are randomly dispersed and are one micron or less in size. Original shot boundaries have particulate uranium oxide, 2-3 microns in size, but have not interrupted uranium diffusion. The Zircaloy-2 uranium bond zone appears clean an subsequent beta heat treatment of the 0.590-inch diameter rod produced uniform bond zone diffusion growth. No detrimental effects were noted as a result of five water quenches from the beta phase. Samples of this material are being prepared for irradiation testing in NaK capsules.

10. Advanced Reactor Concept Studies

Fast Supercritical Pressure Power Reactor. The nuclear design of the 300 Mwe FSPPR has been completed with the exception of a quantitative evaluation of the fuel temperature coefficient. Burnup calculations indicate that an average exposure of 100,000 MWD/ton U + Pu is obtained in six years of full power operation. The control span required for burnup is a function of the initial enrichment used in the absorber regions of the flux trap because of the strong reactivity effect of plutonium formed there. Using an initial enrichment of 3.5% Pu in the absorber regions causes reactivity to increase up to an exposure of about 35,000 MWD/ton, and to decrease thereafter to the initial value by the end of core life. The total control span for burnup is 1.2% $\Delta k/k$.

The mean fission energy of the fast core region of the reactor for normal operation was found to be 82 kev, which results in a core breeding ratio of 0.908, and a total breeding ratio is 1.182. (For comparison, mean fission energy in the Fast Ceramic Reactor is roughly 120 kev.) The moderator in the flux trap appears to have little influence on the core spectrum. Coolant density coefficient calculations, on the other hand, show that the water causes a strong spectral degradation and adversely affects the breeding ratio.

The effect of the flux trap within the core on the power distribution has been investigated for the initial loading and as a function of exposure. Although power peaking occurs to some extent adjacent to the moderator region, it appears possible to achieve a radial peak-to-average ratio of 2. Initial plutonium in the absorber regions also serves to stabilize the power distribution.

An initial critical mass of 1059 kg of fissile plutonium is calculated, of which 119 kg is utilized in the flux trap to assist in power flattening during core life and to reduce control span requirements. Preliminary calculations show that it should be possible to achieve similar results by using a less effective moderator in the flux trap, e.g., graphite instead of zirconium hydride. The FSPPR core design will not be altered to incorporate this refinement but will be frozen with its present composition. Variations in flux trap composition will be made the subject of a separate future study.

Thermal hydraulic calculations and pressure tube stress analysis are nearing completion. To date, the average channel thermal hydraulic calculations have been completed, the results showing a pressure drop through the reactor core and associated fittings of 860 psi. The hot channel calculations completed so far show tube temperature in the

first coolant pass in excess of 1500 F. This condition is being corrected by lowering the heat flux slightly and increasing the mass flow rate in that pass.

A capital cost estimate for the FSPPR was prepared and the initial compilation indicates little difference in cost between the FSPPR and the SPPR (studied in 1961). Some cost changes have resulted from revisions made in the AEC Cost Evaluation Handbook. The cost estimate is presently being reviewed to determine if any design changes should be made on the basis of capital cost.

It has been learned that patent applications will be filed on the FSPPR segmented core design in the U.S. and foreign countries in time for papers describing the core to be presented at meetings in October and November.

11. High Temperature Lithium Loop

A proposal was prepared for a small scale, high temperature lithium-cooled loop to operate in the temperature range of 3000 to 3500 F. The loop is conceived as a table top size experimental and scouting facility for determination of materials and design requirements for components operating under very high temperature conditions. The primary loop would consist essentially of a pump, heater, heat exchanger, and a power conversion device (initially an MHD unit). Cooling would be via a secondary, convective-flow liquid metal loop to helium coolant.

The loop is intended as a small, comparatively simple system for initial experimentation in the higher temperature regions. Components would be made by utilizing Hanford developed extrusion technology with tungsten as the main structural material. Graphite and pyrolytic carbon would also be employed.

12. Reactor Safety Engineering Test Program

Containment Test Facility. A proposed engineering program for the Containment Test Facility study was formulated as the basis for the final Hanford Laboratories program proposal. The program as conceived is a four-step approach consisting of the following phases:

1. Leak tests of the containment structure.
2. Dynamic proof-testing of system capabilities.
3. Short-range comparative testing of concepts of major interest: dry containment, spray, suppression, and wet well-dry well pressure suppression.
4. Parametizing studies of the design and performance variables in all containment concepts of general interest.

A tentative detailed test schedule was prepared for phases 2 and 3. As a supporting program, it was proposed that existing facilities of the Thermal Hydraulics Operation be used to obtain some basic data on blowdown characteristics and to assist in the development and testing of rupture actuator, flowmeter, and void measuring apparatus for later use in the basic CTF program.

Discussions were held with AEC-DRD personnel regarding the proposed program. Assistance was rendered in scoping the facility and equipment and in preparing cost estimates.

D. DIVISION OF RESEARCH - 05 PROGRAM

1. Radiation Effects on Metals

This program is directed toward establishing the combined effect of impurities and neutron irradiation on the properties and structure of specific metals, and deducing from thermally activated recovery processes how the damage state can be altered. Present studies involve single and polycrystalline specimens of molybdenum, nickel, and rhenium.

The subgrain structures of zone-refined molybdenum single crystals are being examined by x-ray diffraction micrographs. This technique utilizes a point source of x-rays and a fine-grain photographic film to produce a representation of the crystal surface in terms of the diffracted intensity from each point on the surface. Imperfections and subgrains may be observed directly, with tilt angles between adjacent subgrains easily measurable from the amount of overlap.

The crystals purchased from commercial sources for use in the first series of irradiation experiments did not prove to be uniform in substructure. Most were very poor, with misorientations between subgrains up to one degree or more. A few were fairly good, having large subgrains with tilt angles less than five minutes of angle. Micrographs made with diffracted beams from (110) planes were quite different in appearance from those made using (200) planes. The substructure in the (200) planes was much less pronounced. This is probably a result of the relative directions of the Burgers vectors with respect to the pertinent planes, but a detailed analysis is not complete. Annealing of the crystals increases the perfection of the subgrains and reduces misorientation, but small recrystallized regions usually appear.

Electron beam zone-refined crystals grown in this laboratory have also been studied. Only one specimen crystal, cut into six sections, has been examined so far; its substructure is much less pronounced than that of the samples procured off-site. The effect of annealing has not been investigated. A total of 120 inches (305 cm) of high-purity molybdenum crystals grown in this laboratory is now on hand.

A study of x-ray line broadening of irradiated polycrystalline foils of molybdenum has been initiated, utilizing the high temperature x-ray equipment. A pre-irradiation annealed sample has been used to obtain reference peaks, and work has begun on a sample containing 10-30 ppm carbon, irradiated to 10^{19} nvt ($E > 1$ Mev). The purpose of these measurements is to secure line profile data from which additional information may be obtained concerning the formation of loops during annealing of irradiated molybdenum. These loops are present in electron micrographs of irradiated molybdenum foils after annealing at fairly low temperatures, 575 C (1067 F). Large dislocation loops are formed in irradiated molybdenum after an exposure of 10^{19} nvt followed by an anneal at 750 C (1382 F) for two hours. Such a foil containing < 10 ppm carbon has been deformed and examined by transmission electron microscopy. Channels, similar to those previously reported present in "as-irradiated" and subsequently deformed specimens, were present. Only an incomplete electron diffraction pattern was obtained, and consequently the orientation of the foil could not be determined.

Foils suitable for transmission electron microscopy have been prepared from massive single crystals of molybdenum containing 400-500 ppm carbon. Transverse sections, 0.020-inch thick, were cut on a wafering machine. Dimples were produced on these wafers with an electrolytic jet and final polishing was done by the usual window technique. Optical observations showed that the sample was a bicrystal with the grain boundary near the equator. One wafer disclosed the presence of a number of small grains in the core of the rod. Electron microscope observations are identical to those made on polycrystalline molybdenum foils of similar composition: large carbide particles are scattered throughout the crystal and dislocations emanate from the carbides. The number of dislocations present is comparable to the number observed in fully annealed foils. It was concluded that the deformed layer due to wafering was completely removed by the electrolytic jet. Techniques used to section and thin these massive specimens are applicable to irradiated single crystal tensile specimens, and will be used for studying defects formed during irradiation and deformation. The particular benefit of the wafering technique is the capability to cut a section parallel

to primary slip planes. Observations on the interaction of point defect clusters with dislocations on a selected slip plane should therefore be possible.

Cold-worked unirradiated polycrystalline molybdenum specimens containing approximately 10 ppm carbon have been tested in tension at a variety of strain rates in order to determine the temperature and rate-sensitivity of the yield stress. Separate investigations have shown that annealed, unirradiated molybdenum exhibits four distinct deformation "processes" at room temperature over a range of strain rates from 10^{-6} to 10^{-1} sec^{-1}. The cold-worked molybdenum, however, showed only two operative processes. This suggests that two of the room temperature deformation processes occurring in annealed molybdenum are dependent upon the presence of a high density of mobile dislocations. Tests are presently being continued at temperatures below room temperature.

Construction of a prototype irradiation capsule for irradiation at elevated temperatures is under way. The capsule consists of four independent heating elements operating at different temperatures up to 1000 C (1832 F). Each temperature zone is monitored by three thermocouples and contains space for irradiating 72 single crystal tensile or x-ray specimens and a large number of polycrystalline foil specimens. The concept of operation is such that heating will continue only when the reactor is at three-fourths full power or higher, holding the temperature at 60 C (140 F) when the reactor is down. Analysis of instrumentation requirements is continuing; solid-state control and recording devices appear to be optimum.

2. Plutonium Physical Metallurgy

The objective of this program is to determine some of the basic physical metallurgical properties of high purity plutonium and to establish the effect of certain specific alloying additions on those properties. Two areas are under study: mechanisms of phase transformations and mechanisms of deformation and recovery.

The completion of the installation of the XRD-5 x-ray diffraction unit in the glovebox has permitted the initiation of exploratory work on the crystallography of deformed alpha plutonium. Preliminary investigation of as-cast material indicated that the results of a computer program, utilizing the lattice structure and parameters reported by Zachariasen and Ellinger, could be experimentally verified. All lines with predicted intensities in excess of about 2% that of the strongest line can be readily identified.

In all cases the experimental pattern agrees satisfactorily in both line position and intensity with those predicted. It has been possible to resolve the $(11\bar{3})$ and (201) lines which appear at 32.19° and 32.31° two theta, respectively. Since both these planes yield relatively strong reflections, the ability to distinguish between them will aid greatly in subsequent investigations of the deformation process.

Initial precision investigations have been started on material which has been flat-rolled to 90% reduction at 70 C (158 F). Cast ingots one inch (2.54 cm) x 2 inches (5.08 cm) x 0.125 inch (0.318 cm) were rolled at 0.005 inch (0.0127 cm) per pass to a final thickness of 0.012 inch (0.030 cm).

It was immediately apparent from the qualitative diffraction pattern that significant changes in the crystallographic arrangement had occurred. For example, the intensity of the (020) and (211) reflections which are unresolved in the usual pattern and appear as the strongest line in the as-cast material are, after rolling, second in intensity and weaker than the unresolved $(11\bar{3})$ and (201) pairs which are normally the third strongest. Other lines are similarly affected. High resolution techniques applied to these two line pairs indicate that only the (211) and (201) planes are in reflecting positions in the worked material. There is some evidence of line shift suggesting lattice parameter changes. Further work in this area is required to establish this with any degree of certainty.

It is apparent that a definite texturing occurs in alpha plutonium after rolling and further work with pole figures will clarify the manner in which the monoclinic structure is able to accommodate extensive deformation.

The higher resolution techniques have also indicated the existence of a small amount of line broadening in the worked material. This is evident primarily from the inability to resolve the alpha one and alpha two reflections in cases where such resolution is possible with the as-cast sample. The results obtained from a rolled sample after annealing at 100 C (212 F) for 18 hours have not been completely analyzed so that it is not possible at the moment to evaluate the effectiveness of this treatment in relieving the residual stresses.

Fractographic investigations of unalloyed plutonium are continuing. Replicas of the fracture surface for electron metallography are obtained by a two-stage plastic-carbon technique. The surface to

be replicated is coated with a saturated solution of cellulose acetate in acetone. A strip of cellulose acetate is applied to the fracture and allowed to dry under pressure for two hours. The negative replica thus obtained is very carefully stripped from the fracture surface and decontaminated by ultrasonic cleaning in alternate baths of 6N HCl and water. A positive replica is prepared in a vacuum evaporator by first shadowing the acetate replica with palladium (at 45 degrees) and then depositing a layer of carbon. The acetate is oriented in the vacuum evaporator by first shadowing the acetate replica with palladium (at 45 degrees) and then depositing a layer of carbon. The acetate is oriented in the vacuum evaporator so that the shadow direction is parallel to the general direction of fracture. The replica is cut into 1/16-inch (1.6 mm) squares and the acetate is dissolved in acetone. The carbon replica is immersed in a series of HCl baths to dissolve any contamination it has picked up from the acetone bath. The replica is rinsed in water and picked up on 200 mesh copper grids for viewing in the electron microscope. The above procedure yields replicas with < 500 D/M detectable contamination.

Replicas of the fracture surface have been taken from "beta-pressed" alpha plutonium which had been impact-fractured at -196 C, -80 C, 20 C, 100C, and 165 C (-320.8 F, -112 F, 68 F, 212 F, and 329 F). Replicas have also been taken from samples that had been beta heat-treated at 170 C (338 F) for 30 minutes, quenched to 85 C (185 F), upquenched to 95 C (203 F) after 25%, 50%, and 75% beta to alpha transformation, respectively, and then fractured. In addition, a replica was obtained from a specimen that had been beta heat-treated at 170 C (338 F) for 30 minutes, quenched to 100 C (212 F) and fractured in the beta phase.

Initial observations in the electron microscope show the fracture characteristics of alpha plutonium to vary with temperature. At -196 C (-320.9 F) the fracture mode is entirely by intercrystalline separation with no evidence of ductile fracturing and very little transgranular cleavage. Fracture at the higher temperatures reveals some evidence of ductile fracturing and considerable amounts of transgranular cleavage. The extremely fine-grained appearance of the -196 C (-320.8 F) fracture indicates the presence of subgrains in plutonium which are not seen by optical examination. This is in support of previous reports of a subgrain network revealed by cathodic vacuum etching of alpha plutonium.

The effect of alpha to beta transformation temperature, compressive load, and grain size on the phase transformation strain in unalloyed plutonium is being investigated. Transformation temperatures under investigation are 150 and 130 C (302 and 266 F); loads are 100 and 2000 psi; grain size, varied by heat treatment of the normal as-cast plutonium, has not as yet been established metallographically.

Unit length increases at 100 psi of as-cast and delta heat-treated plutonium at 150 C (302 F) were 0.021 and 0.027, respectively. Those of as-cast and delta heat-treated plutonium at 130 C (266 F) were 0.012 and 0.025, respectively. These tests were conducted for 800 to 1300 minutes and the thermal creep rates for the final 400 to 900 minutes were less than 10^{-6} per minute. With no load the unit length increase should be 0.03. This indicates some phase transformation strain and/or creep strain has occurred. At 2000 psi and 150 C (266 F), total unit length increase during a 2800-minute test was 0.064. The creep strain of beta plutonium at 2000 psi and at 150 C (266 F), as determined from the thermal creep rate which prevailed during the final 500-minute period of the test, was 2×10^{-5} per minute. The thermal creep strain of beta plutonium at 2000 psi and 150 C (266 F) was also determined independently and found to be 2×10^{-5} per minute.

A chapter entitled "Metallurgical Properties" for the Plutonium Handbook edited by O. J. Wick, Hanford Laboratories, has been completed. This chapter consists of discussions of the Transformation, Kinetics, Heat Treatment, and Recrystallization of Plutonium and Plutonium Alloys.

E. CUSTOMER WORK

1. Radiometallurgy Laboratory

Examinations and Measurements. Routine examinations and measurements are or will be reported as part of the sponsoring research and development programs; however, items of unusual interest or representing new or different laboratory techniques will be briefly described.

During visual examination of a ruptured N-Reactor fluted inner element, the uranium ignited in the rupture area and burned partially.

Heat flow patterns were evident in two nonruptured N-Reactor elements of the same type.

In microdrilling a 2000 MWD/T UO_2 fuel element, a greater tendency to crack and powder was experienced than in previous instances.

Radiation readings were made on an Inconel thermocouple sheath from 105 KE. Metallography disclosed a 0.001-inch reaction layer on one specimen.

Rockwell hardnesses were run on irradiated uranium wafers after cooling to liquid nitrogen temperatures. Compressive strengths were also measured on cooled samples.

Physical and Mechanical Testing Cell. Installation of the annealing furnace is complete and testing has started.

Several minor deficiencies in the dilatometer unit have been referred to the vendor for disposition. Testing has been interrupted pending his action.

Pinhole Autoradiographic Camera. Equipment was fabricated to permit evaluation of the pinhole camera design for gamma autoradiography of irradiated fuel materials.

Orbital Grinder and Polisher. Test grinding and polishing of metallographic samples were continued to evaluate the effect of operating speed on rate of metal removal. Increasing the orbital speed by one-third decreased the grinding and polishing time by one-half. Further increases in orbital speed will be investigated until no further improvement in metal removal time or adverse preparation effects are observed.

Vacuum Induction Furnace. The remote vacuum induction furnace has been mocked-up and test runs completed. The vacuum integrity of the furnace and vacuum system under static conditions is excellent; the leak rate is only 2.4 μ/hr for a 1440-ml system. Calibration of the pyrometer and recorder for zirconium hydriding studies is planned this month. Thirty-gram samples of uranium metal have been heated from room temperature to melting in 30 seconds; still less time is required to heat zirconium samples to 1400 C.

2. Metallography Laboratories

During the report month 401 samples were processed, a total of 929 macrographs and micrographs taken, 2138 negatives printed, and 6741 prints processed.

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

An investigation of reject welds in N-Reactor elements verified that residual salt from the heat treatment step vaporized in the welding process and was responsible for the failures.

In the electron microscopic examination of Zircaloy-2 fracture specimens, only regions preselected by optical means were replicated. One stereo-electron micrograph was prepared.

3. N-Reactor Design Testing

N-Reactor Charging Mockup. A mockup facility was designed, fabricated, and installed in 314 Building. This mockup is used to evaluate various charging hardware and fuel element configurations. The mockup has been tested and operation was satisfactory. To date, several magazine-to-nozzle adapters have been tested and several columns of fuel have been charged successfully.

4. High Temperature Lattice Test Reactor Prototype. Preliminary design was started on the safety rod assemblies for the HTLTR. Primary effort has been directed toward development of scram devices for the rods. Currently, a magnetic clutch is being considered for scram initiation and a mechanical brake for deceleration of the falling rod.

Detail design of the graphite core blocks and keys continued with 88% of the design now completed. Provisions to permit removal of the center four core blocks without tearing down the stack have been incorporated in the design. Design of the insulation details is approximately 10% complete.

A taper joint design contemplated for joining sections of the heating elements has been tested in the laboratory. It was found that the joint could sustain a 45-pound load. Samples of the material proposed for use in the heating elements have been obtained and will be checked for electrical resistivity at temperatures up to approximately 500 C.

5. EBWR Plutonium Fuel Elements

Pneumatic Impaction of PRTR and EBWR Fuel Material. One hundred pounds of UO_2-1 w/o PuO_2 were pneumatically impacted in a single day.

Difficulties experienced with equipment in 308 Building for pneumatic impaction of PuO_2-UO_2 for PRTR and EBWR fuel elements were corrected. The most serious problem with the pneumatic impaction machine was failure of the set pressure unit, which gradually became deformed with use. Replacement parts of improved design were fabricated, installed, and tested.

A minor but persistent problem, breakage of the ejection ram, was caused by recoil of the piston in the ejection cylinder during impaction. The problem was eliminated by replacing the relatively massive 50-ton capacity ejection cylinder with a light weight, 10-ton cylinder. After modifications were completed, the equipment operated satisfactorily.

EBWR Plutonium Fuel Element Fabrication. The new vibration compaction facilities (Project CGH-992) in the Plutonium Fabrication Pilot Plant are lacking only the floor and wall coverings to complete the Minor Construction work. Check out of various process equipment such as feeders, bag sealers, alpha monitors, etc., is proceeding on schedule, and lines should be fully equipped and in developmental use in September.

Development of the "fishtail" vibratory compaction technique has progressed to the point where EBWR-size fuel rods can be multiple loaded and compacted to UO_2 fuel densities of 88-89% of theoretical in less than 30 minutes. No beam or tube breakage problems have been encountered.

Cladding Alloy Procurement. Approximately 4% of 600 EBWR Zircaloy tubes tested with ultra-sound were rejected. A selected ten of these rejects were confirmed by independent methods. The rejects appear to be of three types: outer and inner surface cracks, impressed metal on the outer and inner surfaces, and pits or voids. Slight hydriding was observed; nearly all of the hydride appeared as single needles or as star-burst patterns oriented in a longitudinal direction. The over-all hydride concentration is 8-10 ppm. Small areas of 200-400 ppm were also observed at 750X. Inspection methods were improved to increase the inspection rate to 180-200 tubes/day.

Fifty tubes were scanned at ANL with the eddy current detection equipment developed there. Forty-eight of these tubes had passed the ultrasonic test at HAPO and two were known rejects. The rejects were detected with the eddy current equipment, but the equipment did not indicate as large a defect as had earlier ultrasonic tests. Also, some defect indications were registered with

tubes that had passed the ultrasonic test. The areas were marked for re-examination at HAPO to determine if the signals were valid or were related to the surface finish.

Irradiation Testing Program for EBWR Prototypic Fuel Rods. Fabrication of $U^{D}O_2$-1.5 w/o PuO_2 capsules (GEH-14-494 through 529) was halted because of the impurity content (fluorine, chlorine, and water) of the PuO_2 being used. Twelve capsule irradiations in the MTR-ETR were postponed one cycle.

Two designs for EBWR fuel rod assemblies to be irradiated at Hanford were completed. Drawings and specifications were circulated for review by reactor, physics, and heat transfer groups.

J W Wordfield

For Manager
Reactor and Fuels Laboratory

PHYSICS AND INSTRUMENTS LABORATORY

MONTHLY REPORT

AUGUST 1963

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

N-Spline Measurements

The analysis of the PCTR data on the reactivity worth of splines in wet and dry N-cross coolant channels and in wet process tubes was completed. An informal report, HW-78352, "Spline Worths in N-Reactor Mockup," has been distributed which describes the work.

N-Reactor Temperature Coefficients

A series of calculations of temperature coefficients for the N-Reactor lattice have been performed and the results have been reported.

Optimization of Re-tubed Lattices

The exponential pile which mocks up C-pile has been loaded with both enriched and natural overbore fuel in a super cell array of one natural fuel column surrounded by eight enriched fuel columns. The material buckling will be measured and the results used to determine the ratio of enriched to natural fuel necessary to duplicate the buckling of CIIN fuel in the C lattice.

K Lattice PCTR Experiments

Work has continued on the analysis of the experiments to obtain absolute k_∞ values for KVNS fuel pieces in Zr-2 process tubing in the K lattice. All experimental data have been reduced to equivalent "1/v" detector data, normalized, and plotted, and these results have been compared to thermal flux traverses obtained using computer program IDIOT. An effective neutron temperature for each case has been calculated by the methods of D. E. Wood, and IDIOT cases have been run to verify their validity.

The only problem remaining in the work is the calculation of the epithermal flux in the fuel region. This has been attempted with program G-2, but the results were not satisfactory.

Instrumentation

Review was completed on a multichannel analyzer specification for the N Reactor Fuel Rupture Monitoring System and calibration tests continued on one gamma energy spectrometer of the system at KE Reactor test loops. Instrument drift over several weeks was determined to be less than one percent. Gamma energy spectra obtained from both test loops showed Loop 2 to have considerably higher Mn-56 content. Sample delay measurements showed the Loop 1 delay time to be about 9 minutes while the Loop 2 delay is about 2.5 minutes; consequently, no N-16 peaks appear in the obtained spectra. Plans were made to change thewater sampling system and to improve the detector shielding during the next outage.

A general review was made regarding the developmental fast scan method for reactor fuel rupture monitoring. A number of changes were deivsed to improve general system performance.

Detailed instrumentation work was carried out for the forthcoming N Reactor startup tests. Several meetings were held to establish the required measurement methods. Cable specifications were written and development of the cable and thimble jigs with gas seals was started. The assemblies will permit detector cooling where necessary.

Developmental work continued on specialized preamplifier techniques suitable for high temperature and high radiation environments. The use of General Electric TIMM components is being investigated.

Technical progress was achieved on the experimental reactor fuels testing loop being installed at PRTR. Installation work is proceeding and sketches were prepared covering further work. A majority of the flow sensing, recording, and controlling equipment was received. Delivery of the control valves, however, was delayed for at least one month. The rack cabinets were received and are being modified to suit requirements. The main chassis of a GM count-rate-meter channel was completed; this unit will match the N Reactor Fuel Rupture Monitor GM units. Revised specifications were prepared for the gamma energy spectrometer electronics since no conforming bids were obtained on the original issue.

System Studies

A simulation for N Reactor primary loop flow control system is being prepared which includes two groups of primary pumps with facility for grouping them as necessary to provide the correct total number of pumps, i.e., 1 and 4, 2 and 3, etc. Two loop flow control systems and one master flow controller will be used to simulate various operating conditions and equip-

ment malfunctions. The transient effects on flow pressure and reactor outlet temperature due to pump drop out and certain operational procedures will be studied.

Methods of using the transfer function analyzer to obtain N Reactor process data during startup are being investigated. The second return signal analyzer was installed and calibrated for this purpose. The system now provides capability for simultaneously analyzing the frequency spectrum of input and output process signals in terms of signal amplitude and phase.

Reactor control studies have continued with the development of an equivalent reactor structure designed to facilitate the investigation of manual, automatic and control optimization techniques. In essence, the structure describes a reactor in terms of transfer functions and interaction components in a way that permits ready measurements. Computing techniques are being investigated for methods which will allow the direct and continual measurement of the reactor parameters with regular operating data so that the need for special tests will not exist. Results have given strong indication that the local control method is far superior to the over-all total power control as was proven by the tests on the D Reactor.

A technique was worked out on the analog computer which automatically determines the optimum value of three dependent parameters. A simple quadratic function was used in which each parameter caused about the same effect on the performance. In trial tests the performance function reached about five percent of its optimum value within ten seconds.

A digital computer program was written which reduces the number of delayed neutron group constants, from a given set, such that the low and high frequency response of the transfer function is approached asymptotically. The program computes a group of constants of a size between 1 and n for a given delayed group constant set of size n. The technique applied to transfer function computations has revealed negligible error.

The problem of specifying minimum rod oscillator size which will provide a measurable perturbation in large reactor is being studied. This problem is relevant to the study of NPR kinetics since oscillator size may be a limiting consideration in making kinetic measurements in this reactor.

Considerable assistance was provided N-Reactor Department on the N-Data System Task Force. A draft of a report has been prepared.

SEPARATIONS

Critical Experiments with Plutonium Oxide-Plastic Mixtures

Approval of the Hazards Summary Report (HW-66266, Sup 1) for performing critical experiments with PuO_2-plastic mixtures utilizing the Remote Split-Table Machine was received during the latter part of August.

A critical quantity of the PuO_2-plastic material was assembled for the first time on August 21 with the RSTM.

The PuO_2-plastic mixtures are being used in the experiments to cover the range of densities of plutonium precipitates and polymers which can form in separations plants. The critical data are applicable to the intervening systems of wet powders, precipitates, slurries, polymers, etc., and are of use in the design and operation of chemical plants which process reactor fuels.

The assembly consisted of a non-reflected "checker board" array made up from alternate blocks (2-in. cubes) of fuel and Lucite. The concentration of Pu in the fuel bearing blocks was 1.14 g/cc with an H/Pu atomic ratio of 15. The over-all H/Pu ratio for the matrix of fuel bearing cubes and plastic cubes was $\sim$35. The base dimensions of the rectangular prism were 14-in. x 14-in. The effective height at delayed critical was 10.43-in. The critical array contained 19.1 Kg of Pu. Criticality was obtained when a fuel bearing control rod was partially inserted in the assembly after closing the faces of the RSTM.

Pulsed neutron source experiments were conducted concurrently with the critical experiment. After obtaining criticality, the assembly was maintained critical at low power for about an hour during which time a recording of the "pile noise" was made. The pulsed neutron source experiments and pile noise analyses gave independent but similar answers for the mean generation time of the neutrons in the system (both $\sim 8.4 \times 10^{-6}$ seconds).

A considerable quantity of useful information is obtained about the critical parameters of the system by means of the pulsed neutron source experiments. A plot of the decay constant of the prompt neutrons vs. array height during the critical approach yields a value for the height at which the assembly will become prompt critical. From measurements of the decay of the prompt neutrons near delayed critical, a value for the neutron lifetime can be determined. The control rod worth can also be evaluated from the decay constant with the rod in and out of the assembly.

The temperature coefficient of reactivity was observed to be negative during the experiments. No attempt was made to measure the coefficient.

The safety rod was evaluated by means of the "rod drop" technique (in this case by rapid removal of fuel on scramming the fuel bearing safety rod). The value was ~ $2.70.

During the critical experiment the operation of the RSTM and the safety circuits was without fault.

Subcritical Experiments with PuO_2-Polystyrene Compacts

Prior to the conduct of the critical experiment, a series of subcritical neutron multiplication measurements were performed with the RSTM using the PuO_2-polystyrene compacts as the fuel (H/Pu atomic ratio of 15; Pu concentration of 1.14 g Pu/cc). Data were obtained with the cubes stacked in various rectangular arrays.

The critical values for the reflected assemblies were used to determine that a reflected cube would be critical if 8.48 inches on a side and would contain ~11.4 kg of Pu.

The experiments performed with slab geometry (a 2-in thick slab fully reflected, and a 4-in thick slab reflected on the base) indicate these thicknesses to be subcritical for slabs of infinite dimensions reflected as above. When the 4-in x 24-in slab was essentially fully reflected, the critical length was determined to be 11.4 inches. These data are of use in setting limits for the accumulation of Pu solution in sumps and in areas where spills or leaks would result in the Pu being in slab geometry, such as on the floor of a hood.

The critical length of a 6-in square column reflected with Lucite was estimated to be 53 inches, or to be essentially infinite from the viewpoint of criticality. The 6-in square column was used to approximate a 6-in diameter cylinder, many of which are located in CPD. Information on the criticality of the 6-in square column can be used to establish limits and re-evaluate the nuclear safety of these vessels.

From the very limited data available, paraffin appears to be a slightly better reflector than Lucite. The critical height was reduced by about 4% when the assembly was reflected with paraffin on top instead of Lucite.

The subcritical measurements (limited to 85% of the estimated critical mass) predicted a critical mass about 3½% less than the true critical value. This difference is due to the curvature in the inverse multiplication curve in a

range below 85% of critical. If errors of this nature prevail for the other subcritical experiments, their true critical mass values will thus be slightly larger than reported; thus being conservative from the nuclear safety viewpoint.

Pulsed Neutron Source Experiments

Data reductions and analyses were completed for those pulsed neutron source experiments with plutonium solutions in which control rod worth measurements were made. The pulsed neutron source experiments were carried out concurrently with the criticality experiments in the 11½-inch sphere. The worth of the system's single control rod was determined over a range of solution concentrations. The results from the pulsed neutron source experiments were compared with values obtained from neutron multiplication curves during the critical approach. Both methods are in agreement for the rod's worth in terms of solution volume except for one case in which the pulsed neutron data were questionable.

Pulsed neutron source experiments on PuO_2-polystyrene compacts were conducted concurrently with the Remote Split-Table Machine subcritical experiments. Pulsed neutron source measurements on a reflected horizontal prism (a 6-in x 6-in square column of PuO_2-polystyrene blocks) predicted prompt criticality for a length of 61 inches. Multiplication measurement predicted delayed critical at a length of 53 inches. The difference between delayed and prompt critical is reasonable for this system.

Critical Mass and Buckling Values for One Percent Enriched Uranium; Comparison Between Solid Rods and Tubes

Safe mass and geometry limits for slightly enriched uranium rods in light water are given in the Nuclear Safety Guide (TID-7016 Rev. 1). There has been some question as to whether or not these limits are also applicable to all sizes of uranium tubes in light water (i.e., for a given enrichment might there be a tube size which has a smaller critical mass than the minimum critical mass for optimum size solid rods). To help answer this question, a criticality study has been made for 1.03 w/o U^{235} enriched uranium tubes in light water. Maximum bucklings and minimum critical masses were calculated for tube diameters ranging up to 1.2 in OD, using the 7090 IDIOT code for lattice parameter calculations. It was concluded from this study that for tubes up to 1.2-in OD; a) there is no tube size that will result in a higher buckling than the maximum buckling for solid rods, and b) there is no tube size that will result in a lower critical mass than the minimum critical mass for solid rods. Thus, it appears that the safe limits given in the Nuclear Safety Guide for solid uranium rods may safely be applied to uranium tubes.

Critical Mass, Buckling, and Minimum Critical Slab Thickness for 1.6 Percent Enriched Uranium Tubes and Billets

Critical approach and exponential measurements have been made in previous years on 1.6 w/o U^{235} K-V-E uranium tubes, which are very similar in size and enrichment to the K-SE-III and K-SE-V. The measured results for the K-V-E uranium tubes have been compared with the computed values for this tube size by means of the IDIOT code.

Computational Methods

A more exact approximation for axial leakage has now been programmed into the INTERSET cylindrical interaction code. A rough approximation which has been in use for several years consists of multiplying the interaction probabilities by the ratio of (fast) radial buckling to total buckling. Considerable error may accompany this assumption for various systems. The new approximation makes use of the transformation from the albedo α to k_{eff}. A linear fit of $k(\alpha)$ is used in conjunction with the eigenvalue solution (k) to the interaction matrix. Iteration is continued until the linear fit is in the region of the eigenvalue solution. The code is operating and ready for distribution.

The code which calculates and solves the large eigenvalue matrix approximation to the wave equation is very nearly debugged. Once this code, which has been written subroutine by subroutine, is linked properly, solution to the problem of the buckling of bare truncated spheres will follow.

Instrumentation

Additional data were taken at the Critical Mass Laboratory to determine safety circuit delay time. Positive exponentials of periods .1, 1 and 10 seconds were applied to the input of channels 1 and 4, the channels which use the Beckman Model VDC Electrometers. The time delay for trip varied from essentially zero to 3.6 seconds, depending on channel range setting.

A spare neutron accelerator head was received for use with the pulse neutron generator system. Comparison of the new accelerator head with the old one indicated that the new head produces larger pulses or bursts of neutrons and about twice as many counts per channel on the pulse analyzer system.

Consulting Services on Nuclear Safety - Criticality Hazards

1. Nuclear Safety in HL

   A new series of specifications were issued to cover nuclear safety in Ceramic Research and Development, 308 Building. These specifications

are as follows:

Q-1 Guidelines and General Rules
Q-2 Rules for Plutonium Handling in Wet Glove Boxes
Q-3 Rules for Plutonium Handling in Dry Glove Boxes
Q-4 Rules for the Transportation and Storage of Plutonium
Q-5 Rules for Preparing Plutonium Metallographic Samples
Q-6 Rules for Plutonium Handling in Dry Glove Boxes with Internal Water Lines

2. Nuclear Safety in CPD

Participation on the Hazards Review Committee for the new Recuplex Facility (CAC-880) was completed this month. The results of the review are reported in documents HW-77503 and HW-78880. The project will next be reviewed by the HAPO Hazard Committee and then by the General Electric Technological Hazards Council in September.

Appointment to the Hazards Review Committee for the Waste Treatment Facility (CGC-912) was accepted. This facility will receive low level plutonium waste from the 234-5 Building and the new recuplex facility. The Hazards Review is to be completed in time for the THC review in September.

Participation on the Recuplex Deactivation Hazards Review Committee continued. Procedure A-46, "Removing RB Hood Floor Crust", was issued; however, the work authorized by the procedure has not yet been completed.

3. Nuclear Safety of Off-Site Shipments

A shipment of 2100 lbs. of 1.6 w/o enriched uranium in the form of reject K-SE-III and K-SE-V fuel tubes, and 525 lbs. of uranium in the form of two billets to National Lead of Ohio was reviewed and approved.

Nuclear safety approval was also given to transfer 1787 lbs. of 1.008 w/o U^{235} enriched UNH solution from Critical Mass Physics - HL, to Production Fuels - IPD, for caustic neutralization for later shipment to ORNL.

NEUTRON CROSS SECTION PROGRAM

Scattering-Law Measurements for H_2O at 95°C

Measurements of the scattering law for slow neutrons from H_2O at 95°C using

improved energy resolution were completed during the month. These measurements covered the range of neutron energy changes of 0.075 to 0.25 eV in steps of 0.025 eV. Preliminary analysis of these data indicates that some differences were observed compared to the results obtained with lower resolution, particularly in the region of low values of α (scattering angle). Reduction of these data is in progress.

Analysis of the Scattering Law

The computer programs LEAP and ADDELT for computation of the neutron scattering law $S(\alpha,\beta)$ are now in operation. Successful investigations have been made of the effect on LEAP calculations of the higher vibrational levels of water and also of the correction due to diffusion. Other investigations have better defined the types of structure in $p(\beta)$ which can be successfully calculated by LEAP. A FORTRAN program SPLOT, for plotting the scattering law from LEAP output, has been written and partially debugged.

Time-of-Flight Spectroscopy for Slow Neutrons

Work continued on the development of equipment for a rotating-crystal spectrometer for the measurement of slow-neutron inelastic scattering by time-of-flight. Measurements were made during the month to aid in the specification of radiation shielding both for biological safety and for neutron detector backgrounds in the spectrometer installation at 105-KE. Studies were also made of the problems of electrical noise in the components of the neutron-detection system. Further tests made of the phase stability of two rotors driven by Hypersync motors gave satisfactory performance results, and more powerful versions of this motor were ordered in the hope of attaining the desired rotational speeds.

Fast Neutron Cross Sections

Analysis of the total cross sections from 3 to 15 MeV measured in June for 22 elements has been continued. The data obtained on Li, Na, Ge, and La could not be made internally consistent and have been discarded. In addition, the data obtained for Be and Si appear to suffer from a systematic error in energy scale and will require repeated measurements. The measurements which were made on a sample of radiogenic lead show that much of the fine structure seen in the total cross section of natural lead is due to Pb^{207}.

Several minor modifications to the time-of-flight instrumentation were completed, and eight new samples have been prepared for total cross-section measurements.

A new series of total cross-section measurements has just been started at the end of the month.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

High Exposure PuAl Lattice Studies

A summary report on the data from these experiments is included in the Physics Research Quarterly Report HW-77871.

Approach to Critical Experiments

The results of experiments with 2.0% PuAl (16% Pu-240) fuel which are available to date are as follows:

Lattice Spacing (inches)	Extrapolated to Critical (Rods)	Buckling (10^{-6} cm^{-2})	Extrapolation Length λ (cm)
0.66	1,378	4,346	8.021
0.80	796	-----	-----

The data for the extrapolation to the critical number of rods for the 0.800" lattice are being analyzed with a computer code. The data for the buckling and the extrapolation length has not been taken.

Five of the six templates for these experiments have been completed by the Technical Shops.

Plutonium Recycle Critical Facility

Spectral index and sensitivity measurements were performed in the center of a Pu-Al cluster and in a single Pu-Al rod in a thimble in the center of the PRCF. The thimble contained no moderator when the cluster was used and contained light water moderator when the single rod was used. The spectrum measurements were made by irradiating lutetium foils, bare and cadmium covered, in a plutonium-aluminum sample which was placed in the vertical and radial center of the reactor. The sensitivity measurements were performed by measuring the reactivity worths of three PuAl samples which had different concentrations of plutonium. Measurements were made with and without polyethylene strips which were impregnated with ~8 mg and ~16 mg of boron. The experiment is 60% completed and the results are being analyzed. A measurement of the reactivity worth of a sample is not

reproducible within the error of each measurement. The worths of samples measured at the beginning of the day and six hours later differ by between 12 μk and 23 μk while the error on each measurement is between 0.1 μk and 3 μk. The sensitivities to absorption will be obtained with an uncertainty of about 10%. This error corresponds to approximately 6 $\mu k/cm^2$ of boron.

Relative flux distributions which were measured radially and vertically in the PRCF have been calibrated in terms of an absolute flux. This will allow an absolute power calibration of the PRCF. The calibration of the relative fluxes was accomplished by irradiating gold foils in a known thermal flux in the Sigma pile. The activity from the foils was counted with the same counting system as the foils which were irradiated in the PRCF. This with the known flux in the Sigma pile allowed calibration factors to be computed for the counters. These factors were used to convert the counting rate from one foil from each of the radial and vertical traverses to neutron flux. The foil for the radial traverse was located at a height of 61 in. and 3.46 in. radially from the center. The absolute flux at this point was 1.36×10^8 n/sec-cm^2. The foil for the vertical traverse was located at a height of 54 inches and 4.62 inches radially from the center. The absolute flux at this point was 2.32×10^8 n/sec-cm^2.

The analysis of the PuAl substitution experiment which was conducted in the PRCF has been completed. The results will be used to evaluate hazards analyses, plan measurements of irradiated fuel elements and to arrive at a computational scheme for making reactivity estimates for future experiments. The values were determined by subcritical multiplications measurements. The final results and errors for changes to heavy water are:

Fuel Element	Worth (mk)
PuAl at radius zero	-14.40 ± 0.19
UO_2 at radius zero	- 1.44 ± 0.11
3 PuAl at radius 32"	- 1.96 ± 0.20
3 PuAl at radius 32"	- 2.89 ± 0.25

The differences in the worth of PuAl elements at a radius of 32" is a result of a lower worth for PuAl elements near control rods as opposed to their worth near other PuAl elements.

The combined worth of the three control rods was measured at various stages of the experiment to determine the effect of the control rod environment on the worth of the control rods. The results of the control rod measurements are:

Loading No.	Worth (mk)
42-43	5.47 ± 0.18
44-47	5.14 ± 0.16

These results show the decreasing worth of the control rods as their environment is changed from surrounding cells of 3 PuAl and 3 UO_2 fuel elements (Loading 42-43) to 2 PuAl, 3 UO_2, and 1 heavy water cell (Loading 44-47).

The worth of safety rod No. 2 was measured at various stages of the experiment to determine the effect of environment on the worth of the safety rods and also to be assured that requirements of Process Specifications were met. A value of 24.8 ± 1.7 mk was measured for the worth when safety rod No. 2 was surrounded by 4 UO_2 and 2 PuAl fuel elements. This is compared to 28.9 mk which was measured when the safety rod was surrounded by 3 UO_2 and 3 PuAl elements.

Measurements on Irradiated Fuels

The 1/4" long Co-Al flux monitors from PuAl fuel elements which have been irradiated in the PRTR have been counted. The detector used to count the radiation from the monitors is a "Cutie-Pie" (CP) chamber. The signal from the detector is amplified by a DC amplifier and is then recorded automatically by a digital voltmeter and typewriter unit. An error analysis of the data is being made to determine if the monitors have been counted with sufficient accuracy.

Scattering Law for Water

Program 2GUSWI, which calculates the scattering law for water from an analytical fit to the spectral function for water, produces results considerably closer to experiment than its predecessor GUSWI. 2GUSWI treats two resonances exactly, while GUSWI treats only one resonance exactly. However, the running time of 2GUSWI is considerably greater than that for GUSWI. Work on reducing the running time for 2GUSWI has been started.

PRTR Burnup Experiments

Analysis of the burnup data from elements 5051, 5092, and 5095 is continuing, along with the analysis of the reactivity experiments to be conducted using elements 5052, 5093, and 5094. The reactivity experiments will be modeled analytically using the GAM, THERMOS, and 9-ANGIE computer programs with a four-energy group structure.

The preliminary cross section ratios obtained from the analysis of 5051, 5092, and 5095, using the methods discussed in HW-77935, were somewhat erratic. This was the result of fitting the curves used in the computations by inspection, and the round-off errors associated with reading these curves. To eliminate this problem, least-squares fitted functions will be used to calculate the data points and first differences used to calculate cross section ratios. This method should produce better results. The average values of α^{49} and α^{41} obtained from the first calculations were $\bar{\alpha}^{49} = .42$ and $\bar{\alpha}^{41} = .25$. These values are lower than those quoted in HW-77935. The α^{49} tend to support the use of B. R. Leonard's basic cross section data in the Westcott model rather than the Westcott or Sher data, but the α^{41} is lower than would be expected.

Modified Heavy Gas Equation

Debugging of program PHI is complete. PHI permits the study of the modified gas equation when applied to uranium, plutonium, or plutonium-uranium systems. The flux distributions for a graphite moderator produced in preliminary runs of PHI are being compared to the more exact flux distributions produced by SPECTRE S.

Truncation Bias Accrual

As machine analysis capabilities have improved, reactor physicists have become forced to live with the frustrating observation that more refined theoretical predictions frequently show seemingly perverse and erratic tendencies to give less satisfactory agreement with experimental measurements.

A multi-energy diffusion representation of a thermal reactor, for example, seems to yield consistently better criticality predictions than those given by a multi-energy transport analysis. Such anomalies commonly have been interpreted as evidence of fortuitous cancellation of errors in the less refined theory. However, an absurdly simple but strikingly effective improvement of the level of reliability of numerical integration methods, achieved recently, suggests that part of the trouble may be at once both far more widespread in its effects, and far less profound in its origin, than has been thought.

Consider the effect of the accrual of truncation bias in integrating a typical reaction rate term in a neutron balance equation, by summing a large number of positive-definite contributions all of the same general magnitude. As each element of integration is accrued, eight-decimal-digit truncation biases the partial integral low by a probably fractional error of $\epsilon = 10^{-8}/2$. For n elements of integration, such bias accrues to the

probable fractional error ϵ_n in the integral, bounded by a sequence of the sort

$$1 - \epsilon_n \geq (((1/n + 1/n)(1 - \epsilon) + 1/n)(1 - \epsilon) + 1/n)(1 - \epsilon) + \ldots, \quad (1)$$

which, after geometric-series summation and simplification to first order in ϵ, becomes

$$\boxed{\epsilon_n \leq \epsilon \left[(n + 1)/2 - 1/n \right] .} \quad (2)$$

The last expression has been encircled to emphasize that a truncation-bias fractional error limit of this sort, rising linearly with the level of refinement of the analysis, acts, in effect, as a "numerical uncertainty principle" which implies the existence of a point of diminishing returns in analysis refinement.

Assessment of the typical importance of this uncertainty shows that for a one-dimensional diffusion analysis, the balance-equation term has 4-figure firmness; but for a more refined one-dimensional transport analysis, the balance-equation term is quite shaky in the third figure. Furthermore, in reactors designed for efficient fuel utilization, the sensitivity of criticality predictions to balance-equation terms frequently is such as to multiply these balance-term uncertainties by another factor of 10, in translating them into critical fuel-loading fractional errors. Refining the analysis further, to include either two-dimensional detail or isotope burnup detail, also can be expected to raise the accrual of truncation bias by another factor of 10.

Because the basic IBM-7090 machine operations generate double-precision residues ($\epsilon = 10^{-16}/2$), it is possible to reduce these disturbingly large truncation errors by 8 orders of magnitude, simply by using rounding logic throughout the sensitive innermost loops of an analysis.

The effectiveness of rounding, in comparison with truncation, was evaluated empirically by formulating a Simpson integration routine employing rounding logic, for comparison with the Simpson integration routine from the Hanford machine library (which uses truncation logic). Typical integrals, for a wide range of integrand shapes, showed:

(a) 4-figure limiting accuracy for truncation logic, with the best accuracy occurring for about 4096 integration elements, and with poorer accuracy for more than 4096 integration elements, in close agreement with the behavior predicted by the uncertainty principle (2) for $\epsilon = 10^{-8}/2$.

(b) Nearly solid 8-figure accuracy for rounding logic, attained with about 64 integration elements, and showing no signs of accuracy deterioration for up to 8192 integration elements.

These results seem indicative of a general need for incorporating rounding logic within the innermost loops of almost all of the more elaborate machine analyses now in use or under development. Immediate applications have evolved in improving the level of computational reliability in evaluating fission spectrum integrals and cross section resonance integrals. A one-card Fortran subroutine has been formulated, which provides access to the rounding logic of the machine circuitry in Fortran language. Modification of Hanford's multi-energy transport analysis (GE-HL program S) is under way, to provide rounding logic throughout the entire calculation.

Code Development

RBU

Work continued on the writing of the RBU system documents, the writing of the RBU output program, and the checking out of the two-mass modified gas kernel.

Plans are nearly final for the ANS Master Cross Section Library subcommittee meeting to be held at Hanford, September 18, 19, and 20.

The nuclear data for the Lu-175 and 176 isotopes have been added to the RBU Basic Library to fulfill on-site requests for the nuclear data for these isotopes.

The first RBU problems to be run in the theory-experiment correlation work are ready to run (spheres and cylinders, plutonium and uranium homogeneous systems). The PRTR 19-rod cluster is currently being simulated in the RBU Problem Input. This latter problem will be used in a detailed burnup study before the theory-experiment program begins.

HOP

A 7090 Fortran program called HOP has been written and is presently being debugged. This new program combines the codes DELES, SIGGAS, EMASS1, EMASS2, RETCH, and VALUE into one unit, thus enabling the user to analyze data by any or all of these codes with one set of input and one machine run. Card handling and the possibility of input error are thus greatly reduced. HOP calculates the modifying function for the modified heavy gas equation, re-thermalization cross sections, and the modified Hurwitz function from any scattering kernel supplied as input.

Physics Chain

Preliminary steps have been taken toward adding the codes CLERKII and DUAL NORMAL MODES to the Physics Chain. At present, a complete reactor fuel burnup calculation with these codes is possible only by making three consecutive 7090 machine runs. By adding CLERKII and DUAL NORMAL MODES to the Physics Chain, it will be possible to make this type of calculation in one 7090 run, thereby greatly reducing elapsed time, data handling, and possibility of human error. The projected addition will involve revisions in three codes already on the Physics Chain: GAM, SIGMA-3H, and HFN, as well as changes to the two codes being added.

BARNS

The specialized version of BARNS to prepare 68-group nuclear data from the RBU Basic Library in HRG format was found to be in error in combining individual isotope data to produce cross sections for elements. In making the necessary changes, advantage was taken of the opportunity to break into subroutines in the unwieldy chain link preparing the HRG data tapes. A more understandable printout of the data is also being made. Debugging of the revised version is under way.

Phoenix Reactor Concept Studies

Initial calculations have been completed for the study of Sm-149 and Xe-135 poisoning effects in a Pu-fueled core. The Mark-I geometry ($Zr-2/H_2O$ ratio = 1, 70 kg Pu loading in 500 liter core) was made for this study. A fully enriched U-235 core, very similar to the 80 a/o Pu-239, 20 a/o Pu-240 core, was used for comparison purposes. The Pu-240 content was varied from 5 - 20 a/o. The results are tabulated below. As can be seen, the Xe poisoning in the U-235 core is considerably higher than in the Pu cores.

Xe and Sm Poisoning ($\Delta k_{eff}/k_{eff}$)

a/o 240 Type	5 a/o	10 a/o	20 a/o	U-235 Fully Enriched
Xe-135 PEAK	0.0169	0.0179	0.0202	0.0561
Xe-135 STEADY STATE	0.0155	0.0162	0.0177	0.0279
Sm-149 PEAK	0.0080	0.0078	0.0073	0.0085
Sm-149 STEADY STATE	0.0073	0.0071	0.0066	0.0073

Evaluation of MTR as a possible irradiation facility for a complete Phoenix core is continuing. Analysis of a simple, augmented one-dimensional model of the MTR has been completed; the system studied included a core of plate-type plutonium-aluminum elements containing 4.3 isotopic percent Pu-240. In 1958, prior to full-scale operation and burnup of a similar loading, zero power critical tests and control rod calibrations of this "reactor physics" core were conducted. A "rods-out" experimentally deduced multiplication of 1.200 was reported. In the present analysis, nuclear constants in four-group form were obtained using THERMOS and GAM. A k_{eff} of 1.209 was estimated by means of an HFN diffusion calculation.

The Physics Chain calculational methods used in the Phoenix fuel studies are being evaluated using the 5 w/o Pu-Al light water experimental data. The same data were used in an earlier theory experiment correlation for the MELEAGER code. The study compares diffusion theory and S-4 thermal flux depression factors as well as an over-all evaluation of the Physics Chain accuracy. Preliminary results show that the k_∞'s behave in a similar manner to those calculated in HW-73653 with an underprediction of reactivity at both high and low M/W ratios and an overprediction in the neighborhood of M/W = 0.3.

Fast Reactor Plutonium Utilization Studies

The nuclear parameter survey for compact cermet reactors has been completed. An informal document, HW-78756, "Nuclear Parameter Survey for Some Compact, Fast Spectrum, Cermet Reactors," is in preparation.

Mass Spectrometry

Isotopic analyses in support of Plutonium Recycle Program studies were continued. The isotopic analyses for PRTR fuel element No. 5108 were completed with 32 analyses performed during the reporting period. The 18 uranium isotopic analyses required for PRTR fuel element No. 1501 were also completed. Six analyses were repeated for previously reported PRTR samples to check on sample identification and reliability of analysis.

Instrumentation and System Studies

Plans were partly established for detailed testing of a number of different types of phototubes to determine possible application in the present PRTR Fuel Failure Monitor.

Most of the required electronic and detection components have arrived for use in the PRTR Underwater Gamma Scanner for fuel rod and activated wire measurements.

Although the three PRTR liquid effluent monitors for gamma emitters continue to perform correctly, several months operating experience indicates that minor high range modifications would be useful in providing more readable data. The changes have been developed and will be incorporated in the units as time permits.

A simulation of the Plutonium Recycle Test Reactor is being patched on the Applied Dynamics portable analog computer for use in preliminary automatic controller tests in the PRTR control room. It is planned to test the controller at various power levels using the computer as a reactor and using the actual moderator level control as a computer.

The Plutonium Recycle Critical Facility transient study is presently under way on the EASE analog computer as part of a hazard analysis. It is desired to conduct experiments with the PRCF at power levels above the high power level safety trip point. However, in order to operate at higher power levels it must be shown that an accidental nuclear excursion could not occur. The purpose of the present analog study is to determine whether the scram system, if set to trip at a higher power level, would terminate a nuclear excursion soon enough that safe operating criteria would not be violated. It is further desired to determine whether such an excursion occurring at higher power levels would be more severe than one occurring at presently established operating levels. The analog circuit which was prepared earlier for a one-node PRCF model was used to simulate the behavior of the PRCF when loaded with plutonium fuel only. The temperature equations were changed in an effort to obtain a more realistic model. The analog circuit was altered to accommodate changes and a study was made representing operation with uranium fuel only. The present simulation incorporating both uranium and plutonium fuel is essentially complete.

HIGH TEMPERATURE REACTOR LATTICE PHYSICS PROGRAM

All necessary parts have been constructed for the test equipment in which the corrosion of various alloys, considered for use in the HTLTR, will be studied. The rectangular containment vessel, constructed of Hastelloy-X, will permit the passage of nitrogen at 1000 C over the alloy samples. One set will be held in contact with graphite during the test and another, similar, set will not be in contact with graphite.

The HTLTR mockup has been modified to include a removable graphite test section $7\frac{1}{2}$" x $7\frac{1}{2}$" x 10'. This will provide for temperature and atmospheric testing of items too large to fit into the holes that have already been provided for the heaters, thermocouples, control rods and safety rods.

A three-group diffusion theory calculation has been made for a geometry approximating a possible loading of the HTLTR. The reactor is considered to be an infinitely long, graphite-reflected cylinder with a central, cylindrical cavity. For 10% enriched UO_2 fuel elements consisting of tubes 1.4 inches O.D. and 0.8 inches I.D., the critical loading is about 14 elements. The purpose of the calculation was to determine the reactivity variation for a temperature rise of 1200 C and the amount of gadolinium necessary to reduce the reactivity variation to zero. For the 10% enriched fuel it has been calculated that the amount of gadolinium required would be 0.0045%, by weight, of the uranium. Additional calculations are being made for 5% enriched fuel so that a suitable combination of both enrichment and gadolinium concentration can be selected by interpolation.

Since the introduction of gadolinium into the fuel reduces the neutron temperature coefficient of the reactor, strong reliance will be placed on the Doppler coefficient for reactor shutdown in the event of an excursion. The question as to whether or not the gadolinium resonances will add significantly to the negative Doppler coefficient is therefore of concern also. An experiment in the PCTR has been planned to answer this question. It will consist essentially of observing the reactivity changes that occur when fuel elements are heated in the center of the reactor. One fuel element will contain gadolinium; one will not. Equipment for measuring temperatures, heating the fuel element to about 1000 C, and thermally insulating the hot fuel from the rest of the reactor have been assembled and tested.

Several techniques of neutron spectroscopy are being studied to select a spectrometer for the HTLTR. The reported performance of instruments now in operation is being evaluated to see if an existing design can be adapted to our needs. The existing designs presently under consideration specify conventional time-of-flight spectrometers which use a rotating chopper to pulse the neutron beam. Two new designs have been devised for mechanical velocity selectors which would produce a continuous beam of monoenergetic neutrons and eliminate the long flight path. Calculations are being carried out to determine resolution and intensity and to state the relative merits of the different techniques for the HTLTR application.

Preliminary studies were begun to determine specifications for the oscillators in the HTLTR.

The neutron response of the reactor to square wave and sinusoidal reactivity inputs is being studied by examining the neutron density equations for a bare, homogeneous, thermal reactor. The IBM 7090 code TRIP-005 has been modified to TRIP-105 to allow a sinusoidal reactivity input.

References on existing oscillators are being collected.

NEUTRON FLUX MONITORS

An interim report detailing experimental work performed in obtaining values for r, T, and ø in support of the Phoenix detector program is nearing completion. The document will emphasize experimental technique and data obtained.

The computer program used for evaluating the regenerating detectors was again utilized to calculate the anticipated behavior of three detectors now undergoing irradiation in KE Reactor test facility. As the actual composition of the detectors is slightly different than theoretical optimum, the effective lifetime is expected to be reduced by about five percent; however, these results will not affect the validity of the test. Several sample units have been discharged from the facility and have been prepared for mass spectrometer analysis.

Five assembled Boron-11 beta current neutron flux detectors have completed necessary laboratory tests and are ready for in-reactor testing as soon as a water-cooled facility becomes available. Supplemental information was prepared to support the pending patent disclosure. Necessary readout instrumentation for the detectors is being assembled and tested and the development test prepared.

Mockup testing was carried out on all three loops of the experimental microwave in-core neutron flux monitor. All three loops will fit inside the available reactor test facility. Standard copper tubing of 3/8 inch diameter was used for the loops. Attenuation measurements indicated that losses of about 2 db per foot were obtained and a more powerful microwave source is being ordered. This will provide an adequate signal for the in-reactor tests.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

The signal analysis theory concerning information contained in multifrequency and single-frequency eddy current test data was further explored. A theoretical analysis of a bridge circuit similar to that used in the multiparameter eddy current testing device was made. The theoretical operation of this bridge circuit, when balanced at three different positions, agreed closely with the operation of the multiparameter tester when operated on one frequency using three channels. Additional theoretical analysis showed that this type of operation is capable of separating three parameters when they occur one at a time but not when they occur simultaneously.

A graphical model of a generalized signal analysis situation was evolved to show the basic similarities between the time domain, the frequency domain and other function domains in which signals are viewed or displayed.

Zircaloy-2 Hydride Detection

The new comparator eddy current hydride detector circuit was modified to reduce the cost of subsequent units. Eighty-five samples of N-Reactor process tubes were mapped. A memorandum was prepared to define characteristics of a proposed in-reactor eddy current hydride detection system for process tubes, and to list information and samples needed for design and evaluation.

Addition of two silicon transistor amplifier stages to the comparator eddy current circuit and modification of the output circuit to obtain single-ended output eliminated the need for a high gain differential amplifier that was used in the original single channel prototype. Since one such amplifier had previously been required for each channel, the cost of the field prototype will be reduced by approximately $1,000 per channel (the number of channels required in the final instrument has not been determined).

Although stability of the new circuit can be improved by placing the reference bridges in a box to reduce heat transfer from power components and other environmental thermal factors, it has more than adequate stability for mapping relative resistivities of laboratory samples of Zircaloy-2. Maps were made of the relative resistivities of 85 N-Reactor process tube samples. Differences were found in a small area of one sample, and it is now being metallographically examined to determine the reason. Maps of the other samples will be compared to those that will be obtained after some of them have been hydrided. Changes in resistivities can then be correlated with results from destructive determinations of the hydride content.

A number of questions pertaining to the Zircaloy-2 process tube monitoring system are listed in engineering memorandum CISA 63-1 (Research and Engineering Section, N Reactor Department). Physical Measurements Memo 63-12 was written in reply. The necessity of mapping the relative resistivities of the process tubes prior to any possible hydriding was explained both in Memo 63-12 and in a meeting with N Reactor Department personnel.

During the meeting, the present status of the eddy current hydride detection instrument was described. The equipment was then demonstrated to the attendees by mapping the resistivities of a hydrided section, a welded section, and a normal section of N Reactor process tubing. The hydrided area gave an output signal-to-noise ratio of approximately six to one, and is believed to contain about 1000 ppm hydride. The resistivities of the two halves of the welded

section were slightly different from each other. One half had a circumferential variation of resistivity, and the other had a lower more uniform resistivity. A smooth transition between the resistivities of the two halves occurred through the weld zone.

Incipient Failure Detection

Bar samples fatigued in four-point bending are to be used during initial studies on fatigue detection. Arrangements have been made to obtain samples having a carefully controlled and monitored history from Reactor Metals Research, HL. They have agreed to fatigue the samples to various degrees after initial nondestructive tests have been made.

Fundamental Ultrasonic Studies

Efforts continued to determine the feasibility of using inhomogeneous boundary waves for new ultrasonic tests. Special equipment for these tests was designed and is being fabricated. A report on the fundamental ultrasonic studies concerning liquid-liquid boundaries is being prepared.

The inhomogeneous boundary waves generated at ultrasonic incident angles beyond the critical angles are mostly a surface effect. The boundary waves should, therefore, be sensitive to changes in the acoustic properties of interfaces between dissimilar materials. To determine if interface inconsistencies can be detected with boundary waves, surfaces of zirconium samples having different amounts of hydride, samples of metals having different surface stresses and metal-to-metal bond layer differences are being studied experimentally. Initial studies were undertaken to determine the best methods for measuring these effects by using thelateral beam shift, the change in reflected beam amplitude distribution, and the reflected pulse shape changes. When samples are sufficiently thin, the generation of Lamb waves or other resonance type waves can cause similar beam shifts, etc. These effects are also being considered as possible ways to measure surface interface changes. Special equipment for these experiments was designed and is being fabricated.

A report which includes the preliminary theoretical analysis of ultrasonic wave propagation at liquid-liquid interfaces and some experimental results for kerosene-glycerine is being prepared.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Fabrication of the prototype 17 mil wall tubing tester has been completed. The entire test station including electronics, mechanical tube scanning system, and analog recorder readout has been assembled. Portions of both

the mechanical and electronic systems have required redesigning. However, the basic system appears quite workable. The tubing is scanned at eighteen inches per minute. One, two, and three mil outside and inside notches were reliably located with fairly linear amplitude relationships.

The mechanical system consists of a small tank mounted on a carriage which translates via a lead screw. The tubing is internally chucked to a synchronous rotation motor on one end and to an idler on the other end. While the tubing is rotated at the motor synchronous speed of 1800 RPM, the tank traverses along the tube length thus producing a rapid accurate scan over the entire tube length. The internal chucks on both the drive and idler ends are mounted on a shaft extension which is equal to the tubing diameter. These shaft extensions are of sufficient length to protrude through the tank so that at the end of each tube scan, the tank is positioned over an end shaft. This permits easy loading and unloading of tubing without draining the ultrasonic couplant tank or allowing the inside of the tubing to be wetted. As an added feature a short piece of tubing with the ultrasonic notch standards is inserted over one shaft extension, thus for each tube tested the ultrasonic standard is also tested to insure proper tester operation.

Electro-machined notches have been used for the standard flaws which are satisfactory once fabricated; however, they are both expensive and difficult to manufacture with the accuracies required. A method of fabricating shallow flaws in tubing with a precisely machined punch provided satisfactory results in preliminary tests. A high speed steel tool was machined to the desired notch dimensions and used as a punch. When the tool was used with a proper tube holder and tool guide, imprints up to three mils in depth were readily obtained in Zircaloy tubing. With the tool machined to a triangular tip, both transverse and longitudinal notches are made simultaneously. One tool can also be used to produce several notches so that a complete set of outside standards was manufactured with one tool within a short time. These notches also give satisfactory ultrasonic response in that the one mil punched and electro-machined notches produce equal ultrasonic signals. A thorough evaluation of the punched notches is planned.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Further reduction of data from the cooperative field release experiment conducted on July 25, 1963, involving the release of approximately 100 mc of Iodine-131, continues to confirm that the experimental objectives were met. Peak exposures at the centerline of the plume compared favorably with average values obtained over the same grid using zinc sulfide as the tracer

material. Grass samples from plots located 200 meters' distance downwind from the source received activity depositions of ten thousand disintegrations per minute per square foot. These levels were near optimum for the vegetation studies. Further data analysis was in progress at month end.

Plans progressed toward another field release of Iodine-131 in which 1-3 curies would be dispersed. Air and vegetation samples will be taken over a much larger distance range, increasing the complexity of operation considerably. It is hoped to discern the change in form of the Iodine-131 with travel distance.

Four field tests were conducted during the month. One test was conducted during unstable daytime temperature gradients and low wind speed with the source height at 61 meters to simulate a tall stack. Air samples were collected to a distance of 1600 meters. Three tests were conducted during early morning stable conditions using a source height at 6.7 meters with air samples collected on both the horizontal and vertical sampling grids to a distance of 3200 meters. The total number of diffusion experiments for the year to date was 9 unstable and 15 stable trials.

Further microscope work was completed in our precipitation scavenging research permitting calculation of scavenging efficiencies. Inherent in the air sampling technique is the anisokinetic error due to wind speed effects. Efficiencies for 1.0 and 1.56 mm diameter drops for experiments 3 and 15 were found to be 42 and 79 percent, respectively, for 10-micron zinc sulfide particles. These values are less than those predicted by Langmuir, but must be adjusted upward by a factor of 2 or more to account for the sampling error. Efficiencies in excess of 100 percent have been found by others in the field and in the laboratory.

In instrumentation work, the photographically recording tri-level micrometeorological recording system was brought into full use. Methods of data tabulation and reduction were arranged with Electronic Data Process personnel. A change was made in the anemometer recording system to provide continuous, rather than stepped, operation. The 3 direct current amplifiers for use with the tape recording system for the wind component meter were received. Checking and wiring of the equipment for experiments during September were begun.

A draft of "The Ocean Breeze and Dry Gulch Diffusion Programs - Volume I" - HW No. 78435 received all necessary clearances at Hanford and was forwarded to the Air Force for printing.

Radiological Physics

Whole body counting in Alaska continued. Counting was completed at Barrow and Point Hope. The field party is now at Kotzebue.

More complete analysis of the data taken at Anaktuvuk shows that all the people counted both last year and this showed increases and that the average percentage increase is even higher than reported last month. It was observed that the relative amount of Cs-134 in these people has gone down. The actual amount of Cs-134 present has decreased slightly while there has been a considerable increase in the Cs-137. The average body burden found at Fort Yukon was considerably less than that predicted by a urinalysis survey made last winter. This is probably accounted for by the fact that the people there ran out of meat in February or March and had not been ingesting much Cs-137 between that time and the time they were counted with a consequent gradual reduction in their body burdens. It is interesting to observe that the body burdens of men and women were about equal at Fort Yukon while at the other villages studied those of the women were about two-thirds those of the men.

More urine samples were obtained from Eskimos of known body burden of Cs-137. The rate of elimination of Cs-137 in the urine was the same as that found last year. As another application of this method of measuring Cs-137 burdens the average body burden of people in Richland was determined. Our Industrial Medical Operation saved the unused portions of the urine samples of 150 people. These were pooled and a 500 cc aliquot was counted in a large well crystal by Radiological Chemistry. Using the data obtained from the Eskimos as a calibration, we found an average body burden of 11.5 nc for these Richlanders; the average body burden found at the Hanford Whole Body Counter during the same period was 10 nc. This is quite good agreement.

The price of xenon was recently reduced considerably so we are now studying its use in proportional counters for counting plutonium X-rays rather than continuing with the development of the gas germane.

The positive ion Van de Graaff performed satisfactorily during the month.

We still have not been able to confirm the calibration of other laboratories for the spherical moderator-lithium iodide scintillation counter for low energy fast neutrons. We have been able to duplicate the calibration, within 10 to 20%, but by a method that apparently was not used by the others. In an attempt to improve the interpretation of the data obtained with this instrument, the sensitivity curves used by the other laboratories were assumed to form a complete set of functions for our purpose.

These were orthonormalized and then used to construct neutron spectra using the counting data obtained in some measurements at the PRTR. The results indicated many more high energy neutrons than other methods of interpretation had. We have no objective criteria that permit us to choose between the different methods of interpretation. In an associated theoretical study, the flux density spectra obtained for some criticality accidents were converted to dose spectra. For a well-moderated spectrum 57% of the dose came from neutrons below 1 Mev, 39% below 0.5 Mev, and 10% below 0.1 Mev. For a partially moderated spectrum 23% below 1 Mev, 11% below 0.5 Mev, and 2% below 0.1 Mev.

The AEC Radiological Physics Fellows completed their summer assignment with the Radiological Physics Operation. Part of this assignment was to take the course Radiology 485 (instead of taking it at the University as ordinarily done). The other part was to acquire work experience in neutron measurements. The latter was done by making measurements applicable to the University of Washington spermatogenesis study. Measurements were made of gamma rays accompanying the neutrons, neutron spectra under various conditions, and of neutron shielding. Technical review of the bids for the neutron generator for the study was completed.

Instrumentation

Using the Van de Graaff accelerator as a source of neutrons, continuing tests have been made to optimize response of the three individual detectors being developed as a composite neutron sensor. Over-neutron energy ranges from 45 keV to 5 MeV, integrated response of a Lithium ion detector with Rh-103 and Cd foils (for intermediate energies) and a surface barrier detector with U-238 foil (for fast neutrons) closely approximated theoretical predictions. Some discrepancy was noted, however, in the high and low cut-off response of the mid-range detector. Efforts are being made to develop a total sensor response weighted in terms of true biological effectiveness as a function of neutron energy.

Instrumentation being developed to monitor animal functions such as breathing rate and heart beat is showing improved performance by amplitude modulation of the carrier frequency at 1.5 kc and 10 kc, in place of the frequency modulated system previously investigated. Emphasis is being given the development of selective frequency filters needed to effectively separate temperature, breathing rate and pulse rate data from the telemetered signal.

Cooperative testing with Biology Laboratory personnel was carried out on a newly-acquired transducer used for animal blood pressure measurements. Initial testing appeared satisfactory; however, the physical size of the

transducer may not be acceptable for general use.

Fabrication was completed on a special respiratory control valve to be used at the Biology Laboratory in the measurement of radionuclide deposition in animal lungs. The necessary electronic controls for the valve are nearly completed in breadboard developmental form. It is planned to use a bellows surrounding the chest of the animal to activate the valve.

The three curie Po-210 alpha source was received for use in measuring the amount of beryllium entrapped on air filters. An exhaust hood is being modified to hold the detecting head apparatus. A positioning stand for four filters was completed with disk rotation being used to position the filters for counting. The hood exhaust fans and filters must be repaired before tests can commence. The neutron and gamma detectors and all necessary electronic coincidence circuitry are assembled and ready for use.

Field testing was carried out on the experimental miniature solid state integration circuit that was developed for use with portable radiation detection instruments, especially for the neutron dose rate meters now in use on plant. Equipment performance fully met design requirements.

A special solid state gate circuit was developed for use in aural indicating portion of the line-operated Scintran alpha and beta-gamma monitors. The new circuit provides an improved audible signal. Some 95 Scintrans are in use at HAPO.

Experimental fabrication was accelerated on the final production prototype personnel, rechargeable signaling dosemeter. Most of the required components have been received and a new printed circuit board layout was completed.

Experiments with the developmental wide-range scintillation radiation monitor were carried out with calibration sources and the electron Van de Graaff. Single meter scale coverage in a logarithmic form was achieved for six decades from 1 mr/hr to 1000 R/hr. Initial testing of the solid state instrument has proven satisfactory, and testing will be completed when the Van de Graaff is again available.

Development work continued on the data station circuits of the HAPO Radiotelemetry System and the three-decade range radiation level measuring portion was satisfactorily tested. The scanner-timer, which drives the data station stepping selector switch, was improved with the development of a complete new circuit. Satisfactory testing was achieved from -30°F to +150°F. In addition, a voltage regulator circuit was developed for the -12 VDC supply and changes were made to the supply circuit for the wind direction sine-cosine potentiometer to eliminate a relay and to reduce voltage

transients. All changes have been incorporated into one of the 19 data stations and similar modification was started on a second station. The two will then be tested together to assure validity of the changes. The first data station is now undergoing continuous testing.

Check-out continued on the positive ion accelerator 400 channel analyzer which has capabilities for accepting both pulse height and time-of-flight data. The analog to digital circuits have been installed and partially debugged. Further evaluation is under way.

Wind speed and direction were measured and recorded using a spherical coordinate system in an attempt to evaluate methods of handling meteorological data. For evaluation of the data the recordings were transformed into a cartesian coordinate system on the EASE analog computer. Evaluation of the recorded transformed signal on a tape recorder disclosed a discrepancy between two different tape recording systems.

The analog computer simulation of the decay of iodine and tellurium in the body and milk of sheep was revised to make better use of available information. It was found that the Donner 3500 computer was not able to reproduce data as a result of the high amount of switching logic required for the problem solution. The simulation has been placed on the Goodyear analog computer with apparently good results. The problem now is one of obtaining more data for different animals and different parent-daughter radioisotopes.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses of program samples were provided at a normal rate during the month. Purchase orders were placed for the fabrication of alumina parts for a vacuum lock sample changer for the mass spectrometer. Testing has continued on a scintillation-type ion detector for use as a detector for the mass spectrometer. A satisfactory tunnel diodie pulse-height discriminator was developed for use with this detector. An intermittent source of background ions has developed in the detector tests and has not yet been located.

TEST REACTOR OPERATION

The PCTR was operated for one week during the month. There were no unscheduled shutdowns. Groups of rats were irradiated for the Biology Laboratory in essentially a fast neutron flux. The reactor was down for three weeks for maintenance on the control rod system.

The TTR was operated once during the month for foil normalization. There were no unscheduled shutdowns.

CUSTOMER WORK

Weather Forecasting and Meteorological Services

Consultation continued on the meteorological and climatological aspects of oxides of nitrogen release in the 300 Area. The weather watch for cross-arm changes for Electrical Distribution continued. Assistance was given to Design Analysis and Nuclear Safety in answering question raised by the ACRS in their hazards review for NPR.

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

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	93	87.2
24-Hour General	62	86.4
Special	169	92.9

August was warm and dry with no notable extremes.

Instrumentation and Systems Studies

The feasibility of detecting the presence of uranium in N fuel element support clip welds with beta counting techniques has been given detailed analysis in a report recently issued. This problem is of concern since uranium in the weld nugget is indicative of an over-heated weld and a corresponding reduction in the corrosion resistance of the zirconium cladding. Unfortunately, the results of the study show the technique would at best be only marginally successful.

Considerable changes were incorporated in one coincidence-count type alpha air filter counter being used by Radiation Monitoring in the 325 Building. Some difficulties had been experienced with certain commercially-fabricated portions of the system. Revised operation was successful.

Bids were reviewed for the commercial portions of two coincidence-count alpha air filter counters being assembled for use at Purex and 234-5 Building by CPD.

Preventative maintenance procedures were revised for the long-service gross or high level alpha air monitors being used at 234-5 Building.

Initial assembly was completed on the plutonium waste container monitor for Plutonium Process Engineering, CPD. Testing procedures were established for the use of known Pu-239 sources in sample waste cans. Further developmental work will depend upon the test results.

The soil moisture measuring probe being developed for Advanced Technical Planning, CPD, and Chemical Effluent Technology, HL, is about fifty percent completed.

A low energy beta monitor, specifically for Pm-147, was assembled for use by Fission Product Chemistry, HL. Tests were successful and the instrument was delivered.

An aural signaling alpha-beta-gamma instrument was modified for special use by the Biology Laboratory.

Reactor Metals Research has requested the development of an electro-optical displacement measuring system capable of resolving 50 microinches per inch at temperatures up to 3000°K. Development will proceed along the lines of a prior proposal recommending the use of the relationship

$s = \int v dt$, where s is displacement, v = velocity, and t represents time.

Holding velocity constant, the displacement becomes directly proportional to time. Electronically, time can be measured very accurately (one nanosecond resolution). The proposed system incorporates these relationships by measuring the width of a light pulse produced as a velocity controlled shutter passes two flags mounted on the specimen; the width of the light pulse is proportional to the displacement. System logic will average a number of readings, and the average will be read out. Five readouts per second is sufficient to measure the dynamic specimen displacement.

The micropositioner control unit used on the creep program has been redesigned. The new design incorporates redunancy in critical circuits and a fail-to-stop philosophy. The new unit will be tested in the laboratory and should be on-line within a month.

Debugging of the C-column simulation pointed out difficulties in relay closure timing and in scaling. The MIDAS digital computer program for the C-column simulation was made operational and is being used to assist the analog computer simulations.

Design work for Chemical Effluents Technology's ground water analog has been started. The system under consideration is to be a passive resistive network that will simulate ground water conditions in the Hanford Area, utilizing a 15,000 node passive simulation. Construction plans for a smaller (600 node) test model are now under way. The purpose of the preliminary model is for evaluation of the design for the larger simulation. The data logging equipment for the simulation has been received and assembled.

Optics

Relative threshold measurements on lasing in both U^{3+} and Nd^{3+} lasers at liquid nitrogen temperature have been made in the Optical Shop. Similar measurements were made after several exposure times in a 3×10^6 r/hr Co^{60} radiation field. During this exposure, the Nd^{3+} rod was surrounded with P-20 phosphor and a filtered detector was incorporated to measure any output. No output was detected.

Considerable efforts were made to standardize experimental variables to obtain repeatable threshold measurements. Use of a memory screen oscilloscope greatly facilitiated these efforts and a satisfactory technique was developed.

A Thermodot Model TD-5 radiometer was modified to detect lasing action in the U^{3+} laser. Output was taken from the instrument preamplifier to obtain fast time response. The instrument has a cooled InSb detector which has a response time less than a microsecond compared to the PbS detector response time of 400-100 microseconds. The Nd^{3+} laser output was detected by a silicon photo-voltaic cell which has a response time of about one microsecond. This cell also has high detectivity at the Nd^{3+} output wavelength.

Both rods have a total of 30 minute exposure in the radiation field. Data analysis is not complete but a quick look seems to indicate a rise in lasing threshold with increasing exposure in the field.

Several things must be realized in connection with the absence of lasing due to the phosphor pumping. The temperature in the field is sufficiently above room temperature to produce significant population of the terminal level of the laser transition. This leads to a pump energy requirement before lasing can occur. Secondly, the detector was filtered heavily so that only energetic laser energy could get through. Finally, the efficiency of energy conversion from one MeV gamma photons to phosphor fluorescent photons is not well determined.

It should be noted that the Nd^{3+} laser used in these experiments was ground and polished in the Optical Shop with the end optically flat and sufficiently parallel to the chisel edge to allow a gain in the laser mode to be greater than one per pass.

Physical Testing

Six hundred Zircaloy-2 tubes for fuel sheathing of the Experimental Boiling Water Reactor have been inspected by both ultrasonic and eddy current nondestructive tests. The 0.427 inches in diameter by 0.027 inch wall tubing was tested for defects exceeding 0.001 inches deep, which is believed to be the highest sensitivity and accuracy achieved in production inspection. Four percent of the tubes were rejected, many of these have been examined metallographically. Correlations between destructive and nondestructive tests have been excellent.

A continuing major effort is being applied to the nondestructive inspection of N Reactor steam generators on a continuous basis. The ten steam generators, each containing 4000 welded tube ends, have been statistically sampled with fluorescent penetrant and with eddy current tests to assess the magnitude of the corrosion problems in the tubing. Complete examination of each tube within certain generators has been made to identify for repair the defected areas. In addition to the steam generating inspection program, continuing tests are being made on other reactor components.

An increased use of radiographic techniques for quality inspection of the more unusual materials has occurred recently. Radiography was used to locate the exact position of a series of irradiated tungsten-uranium wafers buried in a bucket of concrete prior to breakup of the concrete to salvage the plates. Varying percentages of beryllium-molybdenum alloy buttons were radiographed to determine their homogeneity prior to further fabrication. The integrity of tungsten welds was determined by radiography on high temperature test capsules for fast reactor studies. Radiography is being used as a referee test in the progress of N Fuels ultrasonic test for the zirconium-beryllium-uranium fuel element end closures.

Other inspection techniques were applied to the examination of pressurized piping system for safety compliance, inspection of repair procedures on pump impellers, and conformance to specifications in the construction of shipping casks by subcontractors to the AEC.

ANALOG COMPUTER FACILITY OPERATION

Analog computer problems considered during the month were:

1. PRCF Hazards.
2. C-Column.
3. Two-Dimensional Reactor Model.
4. Biology Problem.
5. Optimizing a Function of Three Variables.

Computer utilization was as follows:

GEDA	EASE	
276	304	Hours Up
16	16	Hours Scheduled Downtime
36	8	Hours Unscheduled Downtime
0	0	Hours Idle
328	328	Hours Total

Liaison reports indicate that the new computer is currently under construction and the expected delivery date is unchanged from November 15, 1963. The Acceptance Tests are being revised or adapted to fit the equipment and conditions. This work is now approximately 20% complete.

The following programs for solving analog simulations by digital computer methods were reviewed:

1. DynaSar (Dynamic Systems Analyzer)
2. FACE (Fortran Analog Computer Equipment)
3. DAS (Digital Analog Simulator)
4. MIDAS (Modified Integration Digital Analog Simulator).

Of the four, DAS and MIDAS appear most suitable for Hanford use. Complete copies of the DAS and MIDAS programs were received from the Martin Company and Wright-Patterson Air Force Base. The DAS program is being checked out on the IBM 7090 computer.

The MIDAS program was debugged and found to work satisfactorily. Use of this program, however, will entail relatively high costs. It is estimated that the average cost per problem per run for problems of the size simulated on our EASE analog computer will range from $20 to $50. Experience has shown that programming errors can result in costs of about $100 per run.

The programming languages used for both DAS and MIDAS are essentially the same, although there are some internal differences between the programs. The programs also are different externally in that MIDAS will allow the use of only 100 integrators and the total number of elements in the program

cannot exceed 1000 (i.e., summers, multipliers, etc.). The DAS program, on the other hand, allows a program of 3000 elements with no specific limitations on the number of integrators used.

At the present time Data Technical Processing is running an accuracy check to compare the DynaSar, DAS, and MIDAS programs. Their results are not yet available.

The text books to be used for the analog computer course have been received on plant. The books to be used are:

Johnson, C. L., Analog Computer Techniques, McGraw-Hill, New York (1956).

Truitt, T. D. and A. E. Rogers, Basics of Analog Computers, Rider, New York (1960).

INSTRUMENTATION EVALUATION

Final acceptance test procedures were established for the seven combination alpha-beta-gamma hand and shoe counters now fabricated offsite by Instrument Laboratory, Seattle. The same procedures will also be applied to three more units being fabricated by Tracerlab, Inc. A report of pre-acceptance test results was completed. Complete electromechanical drawing revisions were made in preparation for a forthcoming order.

Generally successful operation has been achieved for a month at 100-B Area on the experimental gamma background compensated beta-gamma hand and shoe counter. Daily source tests are being made and the results recorded. The instrument will detect satisfactorily Pm-147. This detection capability solves a recent requirement just established.

Following successful completion of laboratory tests on the experimental, solid state, rechargeable, portable G.M. instrument, extensive field testing has been fully successful and enthusiastic replies have been received from all radiation monitoring personnel.

General evaluation testing was continued on a commercial Victoreen Model 440 portable beta-gamma dose rate meter.

R S Paul

Manager
PHYSICS AND INSTRUMENTS LABORATORY

RS Paul:mcs

DECLASSIFIED

CHEMICAL LABORATORY

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Water Treatment Pilot Plant Studies

The test using aluminum nitrate as coagulant at pH 7.0 was completed on August 2, 1963. Water of excellent quality was produced throughout the test. The concentration of P-32 in the effluent was somewhat less than one-half that resulting from the use of aluminum sulfate as coagulant. This shows that sulfate ion is a source of P-32 by the n,p reaction, or that exchanging most of the sulfate ion for nitrate ion so affects the sorption-desorption process that less P-32 is produced, or that both of these factors are operating. As indicated in the July report, reductions were also observed in other radionuclides. Of special interest was the reduction of As-76 by a factor of 2.

On August 2, 1963, the water treatment process in the experimental unit was changed to produce water at a pH of 6.6 using aluminum nitrate as coagulant.

SEPARATIONS PROCESSES

Purex Diluent Studies

Passage of Texaco C-11, C-13 and Soltrol-170 through silica gel columns removes impurities from these hydrocarbon mixtures and greatly improves their stability against degradation. The resistance to fission product retention by Soltrol-170 was improved by a factor of 1000; thus, a diluent with approximately the same fission product retention as n-dodecane was obtained. A similar behavior was noted for Texaco C-11 and C-13.

Thus, silica gel treatment of Soltrol-170 before introduction into the Purex Plant may offer an inexpensive route to improving the decontamination performance of Purex solvent.

Rapid and convenient certification tests were developed using silica gel and concentrated sulfuric acid. These tests may offer advantages for assessing new diluent batches before they are put in service.

Solid State Electromigration

Work on the purification of plutonium metal by solid state electromigration has been terminated. A final report, "Kinetics of Solid-State Electromigration of the Transition Elements in Cerium, Uranium and Plutonium," is in preparation.

Plutonium Electrowinning

Continued evaluation of parameters which affect the operation of the plutonium electrowinning cell has shown that the cathode current efficiency rises rapidly to a maximum, as a function of cathode current density, and then drops off slightly. Undoubtedly the drop in current efficiency is due to the inception of reduction of the alkali metal cations of the solvent.

Use of Uranium(IV) in the Purex Partition Column

Adequate reduction of plutonium(IV) to plutonium(III) was obtained in batch contact studies simulating conditions at the bottom of the Purex 1BX column when operating at 45 C. The plutonium remained reduced for at least 10 minutes, the estimated residence time of the organic phase in the 1BX column. Uranium(IV) produced by reduction of uranyl nitrate with sodium sulfoxalate formaldehyde (Discolite) was somewhat less effective for plutonium reduction than uranium(IV) prepared with aluminum metal as the reducing agent.

Sodium sulfoxalate formaldehyde was tested similarly as a plutonium reducing agent. Reduction was essentially complete in two minutes' contact time. However, some reaction, as yet unidentified, resulted in complete re-oxidation of the plutonium within 10 minutes.

Trilauryl Amine Recovery of Neptunium and Plutonium from Purex 1WW

Removal of ruthenium from TLA-Soltrol solutions by washing with alkaline permanganate was discussed earlier (HW-78025 C). These studies were extended to develop a wash procedure suitable for use in plant operations where phase separation would be by settling rather than by centrifugation. In the currently most promising procedure, the TLA-Soltrol phase is contacted for one hour at 60 C with one-fourth volume of 0.2 M NaOH - 0.015 M $KMnO_4$ solution. The mixture is allowed to settle four hours or more. Ruthenium decontamination factors of 70 - 100 were obtained in traced experiments. Similar ruthenium DF's were obtained when the organic was contacted first with $KMnO_4$ solution and then with NaOH solution. Settling of solids formed (MnO_2) was more rapid when the contacts were made at 60 C than when made at 25 C.

DECLASSIFIED

DECLASSIFIED

Disposal to Ground

Water samples collected from well 699-2-33, located eight miles south and about two miles east of 200 East Area, following the installation of three piezometer tubes, showed a total beta concentration in the free ground-water aquifer of 7×10^{-7} μc/cc. The concentration decreases downward such that in the lowermost aquifer the concentration is $< 8 \times 10^{-8}$ μc/cc. Initial fluid head measurements showed that water in the free ground-water zone probably did not enter this well prior to the installation of piezometer tubes, due to higher heads at intermediate depths. Resampling of the tubes in this well is scheduled.

Shallow wells in and adjacent to the 216-Z-1 and -2 tile field were surveyed with the gamma scintillation probe to determine the extent of radiocontaminants in the subsoil. This facility was used to dispose of overflow from the Z-3, and earlier Z-1 and -2, building-waste crib. The gamma radiation from trace amounts of fission products which might have been present in the waste and the soft gamma radiation from plutonium (in the case where Pu contamination is high) can be detected with the probe. The tile field wells showed no detectable contamination. This tile field, following some modification, is being considered as a disposal site for CAW waste from the new Recuplex facility.

I-131 in Plant Processes

Measurements were made of I-131 appearing in Redox ventilation air ahead of the sand filter and downstream of a charcoal bed installed earlier this year. The study was undertaken to establish the cause of the very low removal efficiency of the installed charcoal bed for I-131 in the exhaust air.

The independent sampling performed confirmed that the installed charcoal bed was ineffective for collecting I-131. Further observations were as follows: (1) Sodium hydroxide sampling scrubbers, proven to be 90 percent efficient for removing laboratory-generated I-131, were found to be only 45 - 60 percent efficient for I-131 in the Redox air entering or leaving the sand filter. (2) Of the I-131 which was retained on two fresh charcoal capsules in series, 90 - 99 percent was found on the first capsule. The charcoal capsules used have efficiencies of 99.9+ percent for molecular I-131. A caustic scrubber behind the second charcoal capsule retained less than 1 percent of that retained on the charcoal preceding the trap. (3) A charcoal capsule in service for 15 days currently is permitting about a 20 percent breakthrough of I-131 which can be collected on fresh charcoal. (4) Condensable vapors in the air stream which are adsorbed on charcoal were removed by heating and vacuum outgassing of a charcoal capsule. Nearly 3 ml of liquid for a

240 cubic foot air sample was collected from one charcoal capsule. The liquid recovered was virtually pure hexone, and only 5 - 10 percent of the I-131 collected on the charcoal was removed with the hexone.

Although results to date must be considered preliminary, there is strong indication that: (1) a fraction of the I-131 in the air passing through the sand filter is not elemental molecular iodine, and (2) a charcoal trap placed in the stream will adsorb a large volume of hexone which would likely further lower the relatively low initial adsorption capacity of the charcoal for iodine present.

WASTE MANAGEMENT AND FISSION PRODUCT EXTRACTION

Cesium Removal from Wastes

Two additional waste solutions have been tested for ion exchange removal of cesium. The first was a synthetic Purex "1965 FTW" neutralized to pH 10, centrifuged, and the supernatant solution passed through an AW-400 zeolite bed. A highly favorable loading of 130 column volumes to 1 percent cesium breakthrough and 170 column volumes to 50 percent breakthrough was obtained at a flow rate of four column volumes per hour. This process could be useful, as an alternative, for cesium removal from Purex acid waste following strontium-rare earth removal and neutralization, either before or after interim tank storage.

The other waste tested was a synthetic Redox acid waste (D-9) with clinoptilolite exchanger. A poor loading of 10 column volumes to 50 percent cesium breakthrough was obtained in this case.

Soil Permeability of Alkaline Waste

Additional soil permeability data have been obtained on a representative soil from the bottom level of the 241-A Purex waste tank, with a synthetic Purex stored waste. The permeability values listed below have been corrected for viscosity. Permeability values reported earlier for room temperature and unit gradient were not corrected for viscosity.

	Permeabilities $cc/cm^2/sec$ at the End of 24 Hrs.	
	Solution Used	
Conditions	Whole Waste	Supernate
25 C	1.6×10^{-5}	3.3×10^{-4}
80 C	2.8×10^{-5}	1.7×10^{-4}
16 lbs/in^2 at 25 C	1.9×10^{-6}	3.4×10^{-4}

DECLASSIFIED

The reversal of the effect of temperature between the whole waste and the supernate may not be real. Difficulty in achieving identical packing of the soil could account for the factor of two difference. The large decrease of permeability for the whole waste under increased pressure is probably due to solids from the sludge being forced into the soil and clogging up pathways. The lower permeability for the whole waste is due to a large degree to the presence of a layer of sludge on top of the soil. The permeability of this sludge was found to be 6.24×10^{-5} at the end of the first 24 hours, and in eight days dropped to 1.86×10^{-5}. These values correlate very well with the values calculated from the permeabilities of the soil alone and the soil and sludge together.

Distribution coefficients for Cs-137 and Sr-85 were determined for the soil waste system. Cesium-137 K_d values were < 1 for both whole waste and supernate. Strontium-85 K_d values were 2000 for the whole waste and 6 for the supernate. The high value of K_d for Sr-85 with whole waste may be due to the phosphate in the sludge which produced a phosphate-calcite reaction with the soil.

These results indicate that if supernatant waste contacts the soil, the Sr-90 and Cs-137 will spread as far as the waste but because of low permeability of the soil this spread will be limited.

Extraction of Aluminum from Redox Acid Waste

Removal of aluminum from Redox process acid waste is desirable to make the waste more compatible with the current waste management program. Recovery of the aluminum nitrate in sufficient purity for re-use as salting agent in the Redox process may be a secondary objective. When simulated Redox acid waste (1.8 M Al) was adjusted to pH 0.4 and contacted for 30 minutes at 60 C with five volumes of 0.55 M NaD2EHPA - 0.25 M TBP-Soltrol, about 65 percent of the aluminum was extracted. Aluminum was quantitatively stripped from the solvent during a 30 minute contact at 60 C with an equal volume of either 10.3 M HNO_3 or 10 percent NaOH.

Strontium-Lead Sulfate Carrier Precipitation

Work on the lead sulfate carrier precipitation process was continued. In this process, a synthetic Purex-type waste was digested at 80 C with 0.7 to 1 M sulfate at pH 1. Lead nitrate was added and the precipitate centrifuged in a 26-inch solid bowl centrifuge at 1800 rpm (1200 G's). Variables investigated this month were feed composition and method of lead addition. Highlights of the findings were as follows:

1. Increasing the iron concentration in the feed from 0.05 to 0.25 M had little or no effect on the strontium loss to the supernate. The loss increased from the previously reported range of 1.5 - 3 percent to 3.6 percent. The concentration of sulfate was decreased to 0.72 M for this run.

2. Lowering the feed temperature during the above run from 80 to 40 C increased the loss to 4.4 percent.

3. Addition of 0.5 M lead nitrate directly to the centrifuge or line-blending with the feed just before entering the centrifuge vice pre-digestion increased strontium losses. In one run, as the lead concentration in the blended feed increased from 0 to 0.011 to 0.02 M, the losses decreased from 32 to 13 to 6 percent. However, attempts to duplicate this run gave losses generally in the range of 18 to 22 percent at 0.02 M lead.

4. The supernate of one of the runs described in (3) above was successively reprocessed twice through the centrifuge with continuous lead addition to the centrifuge. The strontium loss on the first pass was 52 percent at ca. 0.01 M lead, 40 percent at 0.02 M lead. The loss on the second pass was 60 percent at 0.01 M lead. The cause for these unusually high losses is not known. The strontium concentration in the supernate feed to the first pass was estimated to be 1.1×10^{-4} M, that to the second pass was 7×10^{-5} M. It is conceivable that the high losses were due to the low strontium concentration; however, the low losses reported last month were achieved at strontium concentrations as low as 4×10^{-4} M. The feed for these runs was not digested after the initial pass. No difference in losses was found with the feed entering at 80 C or at 25 C.

Waste and Fission Product Packaging

A Fortran program was written to aid in designing containers for air-cooled storage of heat generating fission products and radioactive wastes. The program provides means for computing the maximum heat generation rate which can be tolerated in a container without exceeding specified surface or internal temperatures. Containers may be of circular or annular cross section and cooling may be by any combination of natural convection (from inner and/or outer walls) and radiation (from outer wall). Output from the program includes:

1. Maximum permissible heat load in container.

2. Corresponding volumetric heat generation rate.

DECLASSIFIED

DECLASSIFIED

3. Wall temperatures.

4. Maximum internal temperature and its radial position.

5. Convection and radiation heat transfer coefficients.

6. In addition, the code develops specific information for the case of strontium fixed on inorganic zeolite materials.

The application of Glueckauf's plate theory(1) to strontium ion exchange on columns of 4A zeolite was investigated. Strontium liquid diffusion coefficients were calculated from conductance data for a simulated D2EHPA-1BP solution was found to vary with temperature from 0.74×10^{-5} cm^2/sec at 25 C to 1.85×10^{-5} cm^2/sec at 80 C. Strontium particle diffusion coefficients were calculated from column data, using Glueckauf's equation. Average particle diffusion coefficients varied from 1.7×10^{-7} cm^2/sec at 25 C to 5.2×10^{-7} cm^2/sec at 80 C. The liquid and particle diffusion coefficients may be used with column parameters (particle size, flow rate, and K_d) to predict strontium breakthrough curves for 4A zeolite columns.

The precision of the particle diffusion coefficients calculated from column data was poor. Although the columns of 4A were vibrated to a standard compact volume, a significant part of the variance is believed due to packaging irregularities.

Cerium Promethium Separation

Need for a radiation-resistant process for the separation of cerium from the trivalent rare earths (and vice versa) was described last month as were the results of preliminary "cold" laboratory testing of a D2EHPA solvent extraction process and several candidate oxidants. Full-level hot cell experiments were initiated in August and culminated in a fully successful, large-scale, plant test (by CPD) at the Hot Semiworks. In the Semiworks demonstration, over 60 kilocuries of trivalent rare earth product were produced with an overall recovery $\geq$ 80 percent and containing only 200 curies of cerium. The product was also free of any detectable traces of yttrium, a fact which may significantly simplify subsequent ion-exchange purification of promethium (by permitting use of simplified processes which do not yield a yttrium-promethium separation).

---

(1) Glueckauf, E. Soc. Chem. Ind. Monograph I, Ion Exchange and Its Applications, 1955.

The process recommended for Semiworks testing involved (1) extraction of all of the rare earths (including cerium) into 0.4 M D2EHPA, 0.2 M TBP, Soltrol ("Semiworks solvent") at pH 2 - 4, leaving most non-rare earth impurities behind; (2) batch contacting the organic with an oxidizing solution containing 0.2 M potassium persulfate, 0.02 M $AgNO_3$, in 2 M HNO_3 (this treatment oxidizes the cerium and causes its retention in the organic while the trivalent rare earths partition into the aqueous); and (3) phase separation followed by stripping of cerium (after auto-reduction from the IV to III state). The cerium is recovered free of other cations. The promethium fraction (trivalent rare earths) contains only silver nitrate and sodium sulfate. It can be readily purified of these by adjusting the pH, re-extracting with D2EHPA, and stripping into dilute acid.

In liter-scale hot-cell experiments with Semiworks feed, cerium extraction coefficients (E_a^o) of 100 to 200 were found, and favorably high values were maintained for several hours or more, depending on concentration of oxidant, catalyst, acidity, and temperature. After final exhaustion of the oxidant, high extraction coefficients were restored simply by adding additional persulfate.

Other oxidants tested included permanganate and dichromate. Cerium extraction coefficients were somewhat lower with permanganate, and the reagent was rapidly destroyed by radiation. Dichromate, although promising in preliminary "cold" studies, gave unsatisfactory cerium distributions and large amounts of solids with actual feed. Besides full-level experiments supporting the above process, several other possible applications of persulfate oxidation - D2EHPA extraction were scouted in the hot cell. Addition of the oxidant directly to the acidic feed followed by D2EHPA extraction of cerium alone (leaving the trivalent rare earths behind) also worked well as did extraction of cerium from Purex 1WW. In the latter case, about 95 percent of the ruthenium co-extracted with the cerium; however, no ruthenium was volatilized into an air stream which was swept through the reaction tank (simulating the plant vessel vent system). Removal of cerium at Purex Head-End by this means prior to sulfate precipitation of strontium and rare earths would greatly increase plant capacity by removing the (cerium controlled) heat-load limitation on the centrifuges. The strontium and promethium could then be stored together (at B Plant) for aging prior to further processing.

EQUIPMENT AND MATERIALS

Corrosion of Mild Steel in Contact with a CO_2-NO_2-H_2O Vapor Phase

The corrosion of mild steel in contact with a CO_2-NO_2-H_2O vapor phase is of interest in connection with radioactive waste storage and waste

DECLASSIFIED

DECLASSIFIED

sludge treatment. Mild steel coupons, two as hot-rolled, two polished and one partially covered with Teflon tape were exposed for 72 hours to the vapor phase above a 1 M $NaNO_3$ solution sparged with CO_2 and NO_2 at flow rates of 40 and two ml/min/l solution, respectively. Temperature of the vapor phase near the coupons was about 75 C. Weight loss corrosion rates varied from 14 to 19 mils/mo. All coupons were heavily pitted; maximum pit depth was about 20 mils.

Scaling of Titanium and Stainless Steel Heat Exchanger Tubes

Two additional titanium bayonets were tested for boiling of simulated Purex 1WW solution. Simulated 1WW containing 0.4 M iron and periodically butted with colloidal silica was boiled by the first bayonet. No increase in temperature difference across the bayonet wall was observed as was the case in a previous similar test. A massive section of weld metal formed when welding the end cap was left on this bayonet. All boiling nucleation appeared to originate in this area. The second bayonet boiled simulated 1WW containing 0.6 M instead of 0.04 M iron. No increase in wall temperature differential with time was observed but a high internal temperature (180 C) was required to boil the solution.

Massive deposits of solids (largely silica) present on the Purex H-4 titanium bundle were removed (by Chemical Processing Department, Waste Handling and Decontamination Operation, at the 271-T Building) by immersing the bundle in hot 10 w/o NaOH. Some deposit remained at the top of the bundle because it was not possible to immerse it completely in the tank used. Also, some deposit remained in the bottom few inches of the bundle. The titanium tubes and tube sheets show no significant corrosive attack. Some of the welds on the 304-L stainless steel shroud legs appear to have suffered significant attack in the heat affected zone.

Stress Corrosion Cracking of Mild Steel

Seven 12-inch by 12-inch mild steel weldments 3/8- to 5/8-inch thick (three submerged arc and four metal arc) were exposed to 40 percent $NaNO_3$ - 10 percent $NaNO_2$ solution at about 90 C for 22 days. The weld metal on one side of one of the submerged arc weldments was almost entirely corroded away; the other side was not visibly corroded. No preferential attack occurred on the other two submerged arc weldments. All welding in these weldments was done with Lincoln L61 wire and Lincoln 760 flux. Microstructure of the attacked weld metal resembled sensitized base metal rather than typical weld metal. No explanation for either the unusual microstructure or the preferential attack is as yet apparent. The face bead on two of the metal arc weldments was run in two stringers. After exposure, a 1/16-inch groove or crack separated the two stringers. Weave type face beads were made on the other two

weldments; no preferential attack occurred. A 3-feet by 3-feet by 1/2-inch weldment fabricated from ASTM A-201-B (A-300, Class 1) steel and 6010 welding rod was cracked after 75 days' exposure to alkaline simulated Purex waste. Cracks appeared to originate at a weld repair area. A cathodically protected (60 μ amp/sq.cm) 3-feet by 3-feet by 3/8-inch weldment fabricated from ASTM A-283, Grade C steel has not cracked after 45 days' exposure to 40 percent $NaNO_3$ - 10 percent $NaNO_2$ at about 90 C.

Effects of Welding Parameters on Corrosion of 304-L Stainless Steel Weldments

Weldments prepared by shielded metal arc welding for this study have been exposed in 65 w/o HNO_3. No significant differences in corrosion rate among the various weldments was observed. Also, rates observed were comparable to those observed previously for similar welds produced by the tungsten gas procedure.

Non-Metallic Materials

Chemical compatibility tests were completed on another nine rubber samples -- two nitriles, four neoprenes, one urethane, one fluoro and one natural rubber. Chemical compatibility tests were also completed on the styrene-acrylonitrile rubber cited last month as having remarkably good mechanical strength and electrical resistance after 10^9 R gamma irradiation. Total immersion for 28 days in caustic soda, Purex HAX and D2EHPA caused no apparent damage. Damage ranging from severe to minor occurred in hexone, 60 percent HNO_3, Recuplex CAX, CCl_4 and 25 percent DBBP - 75 percent CCl_4. Damage to irradiated samples was less severe than to non-irradiated samples.

PROCESS CONTROL DEVELOPMENT

Pulse Column Capacity Control

In the Plutonium Reclamation Facility the capacity of the solvent extraction columns is to be controlled by means of feedback of a measurement indicative of total dispersed phase holdup. Total dispersed phase holdup is inferred from a dynamic pressure measurement after the oscillatory effects due to the pulse are filtered out. Analysis of this measurement method shows that significant errors are introduced by the assumption that averaging of dynamic pressures removed all effects of cartridge design, flow rate unbalance, pulse unsymmetry, amplitude, and frequency. In order to improve the accuracy of the measurement, recent effort has been concentrated on rejecting the dynamic components of the pressure measurement rather than filter averaging.

DECLASSIFIED

DECLASSIFIED

Tests using a strain gage pressure transducer to measure the dynamic pressure showed a day-to-day stability of better than 0.05 percent of the gage full span after being subjected to normal column pulsation pressures. Expected stability was better than 99.8 percent based on power supply stability, gage hysteresis and variations in ambient temperature. If similar accuracies can be obtained in plant installations, a measurement indicative of dispersed phase holdup could be accurate to 1 percent.

C-Column Facility

The nitric acid concentration in the organic feed was one of the variables designed into the recently completed 48-run C-Column experiment. The 48 runs were split into three 16-run groups with 1CF acid concentration design levels of 5 - 8 grams nitric acid per liter, 0 - 1 gm/l and 10 - 15 gm/l. Preparation of the 1CF solutions was subject to error because a reliable method for the analysis of nitric acid in an organic-TBP solution containing uranyl nitrate was not available and the feed preparation procedure had to rely on KAPL curves relating the equilibrium distribution of uranyl nitrate and nitric acid between aqueous and organic solutions. While the two low acid feeds were within the design range, the acid in the feed for the last block of runs was higher than the design limit. For this and other reasons, a new batch of feed was prepared and the last block of runs was repeated.

Current studies of the acid-in-organic analysis by analytical laboratory personnel have resulted in a much improved procedure that promises far greater precision than has been obtained. In addition, the results of a recently completed experiment show clearly that the concentration of nitric acid in 1CF solutions steadily decreases with time when the samples are contained in clear glass sample bottles and are exposed continually to sunlight. Simultaneously, their color gradually changed from a clear yellow to a cloudy brown. This light-activated breakdown of the 1CF samples explains the major cause of the inconsistency in the acid found in the 1CF samples that were submitted for confirmation of feed specifications. The samples varied in age from a week to several months when they were submitted and had had various exposures to sunlight and fluorescent lamp light in the laboratory when they were being analyzed for their uranium content.

ANALYTICAL AND INSTRUMENTAL CHEMISTRY

Determination of Cobalt in Graphite

Neutron activation analysis readily served to determine 0.0001 to 0.01 percent cobalt in graphite lubricant. Precision was within 5 percent. Cobalt was estimated from abundances of coincident Co-60 gamma photopeaks, separately and in summation.

[redacted]

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Engineering Development

Analyses are now available for 410 pounds of electrolytic UO_2 produced in the LiCl-KCl salt system and delivered in July to the Advanced Fuel Development Operation. The product was 36 percent -4 +10 mesh, 37 percent -10 +35 mesh, 12 percent -35 +65 mesh, and 15 percent -65 mesh. The lithium and chloride contents of the water-washed products were 15 - 30 ppm and 100 - 200 ppm, respectively, independent of particle size. About 300 pounds of the material had an oxygen-to-uranium ratio averaging 2.005. The O/U ratio of the remainder of the UO_2, produced from wet melts, varied from 2.007 for -4 +10 mesh to 2.040 for -65 mesh particles. The more porous initially-produced deposit nearest the cathode becomes fines on crushing and has a higher O/U ratio because the melt dries during electrolysis. The carbon content was 20 - 30 ppm for UO_2 produced on pyrolytic graphite cathodes and 150 - 500 ppm for UO_2 produced on nuclear grade electrodes.

Salt Cycle Flowsheet

Attempts to reach the process goal of recovering at least 99 percent of the plutonium present in molten chloride solutions with a minimum uranium contamination, by electrodeposition of UO_2-PuO_2 solid solutions, have continued. In the course of these studies it has been ascertained that enrichments as high as 44 may be obtained on a platinum cathode and cathode products containing as much as 36 percent PuO_2 are achievable.

Processing PRTR Uranium Dioxide Fuels

Proposed decladding (Zirflex) and core dissolution procedures (for the Redox plant) were tested with two 10-inches long sections cut from irradiated PRTR elements 1006 and 1501. Element 1006 was from a 19-rod assembly irradiated to 150 MWD while element 1501 was from a vibrationally compacted tube-in-tube assembly irradiated to 78 MWD. Zircaloy cladding was removed completely during six hours at boiling from Element 1006 in solution initially 4 $\underline{M}$ NH_4F - 0.36 $\underline{M}$ NH_4NO_3 and from Element 1501 in solution initially 6 $\underline{M}$ NH_4F - 0.5 $\underline{M}$ NH_4NO_3. The UO_2 core material, in both cases, was left as pieces 0.5 to two inches in length. Very little, if any, of the core material distintegrated to a "mud" as occurred with similarly declad non-irradiated PRTR uranium dioxide fuel.

DECLASSIFIED

Dissolution of Element 1501 core material in solution initially 4.2 M in nitric acid to produce a dissolver solution 1.5 M in UNH and 0.2 to 0.5 M in HNO_3 proceeded smoothly and without incident.

RADIOACTIVE RESIDUE PROCESS DEVELOPMENT

Calcination Studies

Major activities of the month included several runs in the "cold" 8-inch spray calciner to define conditions for a hot-cell run with phosphate additives; a run with an iodine-131 "spike" to determine the path of iodine in waste calcination; and at month's end, a successful, full-level, A-cell spray calciner "phosphate glass" run.

"Cold" calciner runs with phosphate added to a feed, which simulated the aged Purex 1WW which has been used in all the previous hot-cell calcinations showed that a slight increase in sodium content was required to yield a melt at less than the ca. 900 C temperature of the melt-pot furnace. A full-level run was then made with a phosphoric acid addition sufficient to form ortho phosphates of all the metallic cations present and with a Na/Fe/Al ratio of 0.7/0.22/0.08 (vice 0.6/0.29/0.11 prior to sodium addition). This was the first run in the spray calciner with acidic feed since discovery of a leak in the shell, all intervening runs having been with alkaline feeds which evolve very little ruthenium. Great care was exercised to avoid pressurizing the column at any time during the run and release of activity to the cell filters and stack was negligibly low (and no greater than routinely experienced in previous spray and pot calcination runs). Although analytical results were incomplete, preliminary indication is that fission product behavior was about the same as that of other acid-side runs and that the presence of phosphate had little effect on the behavior of ruthenium or other fission products.

An iodine evolution run was performed in the "cold" spray calciner with a synthetic, acidic, Purex waste containing a concentration of cold iodine equivalent to the total iodine in actual waste plus a spike of I-131 (50 microcuries) sufficient for analytical purposes. Although in operation of a plant-size calciner, waste would normally be cooled for a sufficient period of time (to minimize heat dissipation problems and maximize storage container geometry) such that iodine-131 would be essentially absent, it was considered of interest to determine what the behavior of iodine would be, were very short-cooled or blended waste to be calcined. Although the iodine material balance was rather low (due to poor efficiency of the charcoal absorbers used for analytical purposes), indications were that only about 20 percent of the iodine was retained in the calcine, that another 20 percent was retained in the condenser and condensate, and that the remainder escaped and passed through the final filters.

Eighteen-Inch Radiant-Heat Spray Calciner

Further runs were made to determine the capacity and the mechanism of heat-transfer within the calciner. An analysis was made of the radiant-heat transfer to an absorbing gas flowing through the annulus with the aid of a computer program implemented by personnel of Applied Mathematics Section of Hanford Laboratories.

The analysis of radiant-heat transfer to the gas recycled through the annulus indicated that the heat flux between the wall and the gas in a two-inch annulus would be approximately 25 percent of that expected between two black bodies. Secondary radiation from the draft tube to the spray in the center of the column was estimated at about 17 percent of that expected between black bodies if the draft tube were at the reactor wall temperature (the draft tube absolute temperature was calculated to be about 90 percent of the reactor wall temperature). Using these estimates, the capacity of the calciner for water evaporation at wall temperatures of 580 to 750 C and lengths of 4.75 to 8.5 feet could be predicted with an error of less than five percent.

The capacity of the reactor, with the adjustable draft tube and bottom cone raised to form a reactor of minimum length (4.75 feet), was measured. The average maximum heat flux calculated for the whole effective surface area of the reactor was increased by approximately 35 percent compared to those measured at an 8.5 foot length, that is, at a wall temperature of 730 C the flux, Q/A, increased from 7,900 Btu/hr.ft^2 @ 8.5 ft. reactor length to 10,700 Btu/hr.ft^2 @ 4.75 ft. reactor length. This is due to the contribution of direct radiation to the spray in the top of the column.

The capacity of the reactor when operating with a typical simulated waste solution was found to be about 80 percent of that with water under otherwise identical operating conditions. At least five reasons for this lower capacity can be advanced: (1) the heat required for evaporation and calcination of the waste is about 7 percent higher than water on a per volumetric basis, (2) the boiling temperature of the drops of waste is somewhat higher than water, (3) complete drying-calcination involves a more difficult mass transfer problem than just evaporation, (4) atomization is undoubtedly poorer because of the higher viscosity of the solution, and (5) heat-transfer from the walls of the reactor is somewhat impaired by dust. A figure of approximately 50 percent the capacity with water is thought to be a reasonable design figure.

Calcine Powder Melter

As part of the accelerated development program in support of the Waste Solidification Engineering Prototypes, a continuous melt system for

DECLASSIFIED

DECLASSIFIED

powder calcines has been installed and successfully operated in the Cold Semiworks. The melter is induction heated, about 8 inches in diameter by 11 inches high, and constructed of Inconel. It is designed to produce a phosphate "glass" product at instantaneous rates equivalent to processing waste from a separations plant handling two tons of uranium per day. Steam can be admitted to the off-gas stream to expedite removal of any SO_2 that may be evolved from the melt. The unit will be directly coupled to the 18-inch radiant heat spray calciner at a later date.

Three experimental runs have been completed using a calcine powder feed prepared from a synthetic Purex waste with lithium, calcium and phosphate added before calcination-drying in the radiant heat spray calciner. The calcine formed a free flowing liquid at the melter temperature of 900 C. Visual observation (through a quartz window) of the melting process showed little foaming or spattering of the melt and indicated a very rapid melting of the feed powder. The melt was intermittently discharged through a bellows actuated valve in the bottom of the melter. Upon cooling, the product was a hard, homogeneous, dark-brown, seemingly deliquescent solid. It had a density of 2.7 gms/cc.

Denitrator Evaporator

One of the solidification schemes proposed as part of the Waste Solidification Engineering Prototypes is the continuous phosphate glass process under development by the Brookhaven National Laboratories. All current concepts of this phosphate glass process incorporate a denitration-evaporation step prior to the high temperature melter.

Studies of the heat transfer coefficients in such an evaporator have been initiated in a small unit using a steam heated bayonet. Simulated Purex waste with appropriate phosphate addition was concentrated in the evaporator by a factor of 7 to a boiling point of 135 C. At this temperature the concentrate has a viscosity of 300 centipoises and a specific gravity of 1.85. During concentration excessive foaming occurred at high boil up rates. Initial data, using a ΔT of 40 F, show the overall heat transfer coefficient to the concentrated waste is 190 $Btu/hr.ft^2.^oF$. Water in the same evaporator has an overall coefficient of 305.

Viscosity Measurement

Various methods of measuring viscosity of molten glass at high temperatures have been investigated. The two methods that show the most promise are (1) measuring the torque of a rotating spindle in solution, and (2) measuring the time that it takes for an object to fall in the solution through a known distance.

Both methods are satisfactory but the falling sphere method will be used because of equipment available. The sphere is suspended in the solution by means of a wire over a pulley and attached to a scale-pan to partially counterpoise the sphere. The times of fall are noted with two different weights in the scale-pan. The viscosity can then be evaluated from the formula

$$n = \frac{W_2 - W_1}{1/T_1 - 1/T_2} K$$

where T_1, T_2 are times corresponding to loads W_1, W_2 and K a constant found by observing times of fall with a liquid of known viscosity. A viscosity curve of glycerine from less than 1 poise up to 87 poises has been obtained in the laboratory. The factor, K, was found to vary slightly with viscosity but this can be taken into account in the final calibration.

Corrosion of a Calcined Waste Melting Pot

An Inconel melt pot failed in service in the 321 Building due to localized corrosion which penetrated the pot wall. Tentative conclusion based on observed increase in magnetic character of the metal near the failed area is that preferential leaching of nickel from the metal occurred at the failure.

Thermodynamic Properties of Zeolites

The self-diffusion coefficient of cesium-134 and strontium-85 in the presence of sodium ion was determined for 13X and 4AXW. A continuously-monitored shallow bed with an infinite solution volume condition was used Half-gram beds were first equilibrated with a solution containing 0.1 N NaCl plus 0.1 N CsCl or 0.1 N NaCl plus 0.1 N $SrCl_2$. Cesium-134 or strontium-85 tracer was added and the buildup of radioisotope on the bed recorded.

The concentration of cesium or strontium at the zeolite sphere (or particle) surface and in the solution was always the same, so that diffusion within the zeolite particle was rate-controlling. Consequently, the relationship

$$t_{1/2} = 0.03 \frac{r_o^2}{\bar{D}} \quad (1)$$

(1) Helfferich, F., Ion Exchange, p. 261, McGraw-Hill, 1962

DECLASSIFIED

DECLASSIFIED

could be used to determine a self-diffusion coefficient. The $t_{1/2}$ is the time to 50 percent loading in seconds, r_o is the zeolite sphere radius in centimeters, and $\bar{D}$ is a diffusion coefficient of the traced cation within the zeolite sphere in cm^2/sec.

The determination of self-diffusion coefficients at two different r_o values showed that the exchange kinetics of 13X and 4AXW obeyed the above rate law. The self-diffusion coefficients for Sr-85 were generally a factor of 10 less than for Cs-134. The zeolite 4AXW, for example, yielded a Cs-134 self-diffusion coefficient of 2×10^{-7} cm^2/sec in the presence of sodium at 50 C and 3×10^{-8} cm^2/sec for strontium.

Ion Exchange Properties of Ferrocyanide

Work continued on characterization of the transition metal ferrocyanides. Elution of cesium from columns of copper and nickel ferrocyanides was shown to be possible with 1.6 molar hydrochloric acid. The best elutions were obtained at 80 C.

A 0.01 M CsCl plus 1.0 M HCl solution was used to determine partial capacities of copper and nickel ferrocyanides. The high acid concentration was necessary in order to stabilize the salts. The nickel ferrocyanide columns reached 50 percent breakthrough at 182 column volumes or 1.55 meq/gm. The copper ferrocyanide column reached 50 percent breakthrough at 261 column volumes or 2.39 meq/gm.

Distribution coefficients indicated that nickel, copper and cobalt ferrocyanides would exchange cesium from a basic waste solution (pH 10-11) with copper showing the highest capacity. However, the copper ferrocyanide physically disintegrated when used in a column with a neutralized Purex synthetic waste. This disintegration was attributed to the high concentration of carbonate present. The copper ferrocyanide remained physically stable in neutralized Redox waste (SX), which did not contain divalent anions.

Low-Level Waste Treatment

Current efficiencies and desalting rates have been determined for a laboratory scale electrodialysis unit over a narrow range of salt concentrations. Current efficiencies from 60 - 90 percent were obtained for deionization of 0.05 N solutions of NaOH, $NaNO_3$ and Na_2SO_4. In general, current efficiency decreased with an increase in applied current and with decreasing salt concentration. Current efficiency increased with increasing flow rate through the unit. These effects are to a large degree due to polarization at the solution-membrane, solution-electrode interfaces.

The electrodialysis unit consists of 20 concentration-dilution cell pairs each having a working area of 220 cm^2 and a desalting capability of 0.00017 equivalents of salt transfer per ampere second at 90 percent current efficiency. The maximum flow rate obtainable with the unit was 2 liters/minute at a hydraulic pressure of 15 psi. Maximum current was 5 amps for a 0.1 N NaOH solution at 50 volts. The current efficiency was 94 percent at a desalting rate of 0.00085 equivalents of salt per second. Twenty-seven percent of the salt was removed in a single pass through the cell. For the same solution at a flow rate of 180 ml per minute. maximum current of 1.9 amp, 90 percent of the salt was removed but the current efficiency was only 60 percent.

Based on these limited studies electrodialysis shows promise for low-level waste treatment. One particular area where it may be used efficiently is in desalting solutions prior to ion exchange treatment.

Condensate Treatment

The small steam stripper is being used to produce 20 gallons of stripped Purex Tank Farm condensate for use in process research and development studies. This stripped condensate will be used to explore methods of increasing the ruthenium capacity of anion and cation ion exchange resins.

DECLASSIFIED

DECLASSIFIED

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

Fifty-two piezometer tubes were installed in 34 wells during the current month utilizing the sand-pack method These tubes will provide potential measurement data needed for the ground-water electrical analog and also permit samples to be obtained from the various ground-water zones. Considerable head differences (up to 34 feet) were noted in some wells, but in many cases the water levels have not yet reached equilibrium. Other wells are scheduled to receive piezometer tubes following cleanout and/or perforating.

The method reported last month for obtaining ground-water flow paths from the potential distribution has shown good accuracy. The agreement with an exact analytical stream function for a three-dimensional source and sink in a uniform stream was excellent.

The statistical study of well surveying accuracy was completed by the Applied Mathematics Section. The average error in well elevation, E, was found to be 0.055 foot with a 95 percent confidence interval of $-0.019 < E < 0.129$ foot. The elevation is the critical value as concerns electrical analog input data, and this finding relieves much of the concern expressed earlier on elevation accuracy. The average errors in horizontal surveying control on the wells were 26.2 feet and 15.9 feet for the north and west plant coordinates, respectively.

The Fourier series boundary value problem for potentials in the test analog model was abandoned when it became apparent that convergence could not be reached with 100 terms of the series. More terms were not practical in both time and computer capacity. A reformulation of a new boundary value problem which causes the series to degenerate to two terms was made, and a new computer program was written in which no convergence problems will be encountered. The computed results will be compared with potentials obtained from the preliminary analog, and the program will be expanded to study the effects of grid size warping on a finite grid.

The Prosser interbed, which has the potential to transmit separations plants effluents into the upper part of the basalt series, is now known to correlate with the upper part of the Selah Formation in the Yakima Valley area. The interbed is more extensive westward than previously recognized. The overlying basalt flow, termed the Saddle Mountain No. 3 flow, also is now correlated with the Elephant Mountain flow of the

Moxee area, and similarly is known to extend over a larger area than earlier believed.

The recent correlations will better permit determination of the probable origins of the interbed and overlying basalt flow, their manner of deposition, their hydrologic characteristics where they are exposed, and the potential of movement of waters from the interbed into the basalt flow, where the interbed either pinches out or becomes less permeable than the basalt flow. To the southeast the interbed pinches out north of Wallula Gap. The overlying basalt flow, although it continues south of the Horse Heaven Hills, rises above the river level at Wallula Gap. Hence, the contained waters evidently enter the Columbia River, the ground water table base level, north of Wallula Gap.

RADIOLOGICAL & HEALTH CHEMISTRY

Iodine-131 Studies

I-131 counting measurements were completed on the filter cartridge, vegetation, and soil samples taken subsequent to the I-131 dispersal test. Because of the small release sensitive counting techniques were required. Data are being collected, computed and evaluated. Several interesting findings will affect the conduct of the next test. Although the iodine was presumably released as I_2 vapor, about 10 percent was found to be associated with particulate material even after only a few hundred meters travel. Some 15 percent or greater was also found to be in a non-reactive form not collected by silver-coated copper turnings. These physical and chemical form changes will have a bearing on the deposition and uptake of these materials by plants.

On studies of the I-131 released in the Redox stack gas, it was found that eight charcoal filters in series only collected 40 percent of the I-131 although in laboratory tests with I_2-131 one charcoal filter was sufficient to remove almost 100 percent of the I-131. Further studies to identify the chemical form of this I-131 containing material are in progress.

Radiation Chemistry

Hydrogen peroxide is one of the products formed during the radiolysis of water which can be expected to react with other solutes which may be present. Since these reactions may be quite slow when the reactants are present at low concentrations, they may account for post-irradiation effects in irradiated systems. The study of long-lived free radicals requires one to distinguish these slow peroxide reactions from those of the radicals and the immediate impetus to study this problem was the

DECLASSIFIED

slight post-irradiation effect noticed in erioglaucine solutions. Commercial AR grade H_2O_2 was added to solutions of erioglaucine at various pH values and the reaction kinetics were followed with the Beckman DK-2 recording spectrophotometer. Excess H_2O_2 was added in all cases to change the reaction from second order to pseudo-first order. In spite of the paucity of data, one can estimate that even at pH 8, the time for the post-irradiation bleaching of erioglaucine to go 10 percent toward completion will be around 1000 days, assuming both reactants start out around 10^{-5} M. It thus appears that the post-irradiation effect on erioglaucine is not due to H_2O_2, although other solutes might affect the rates greatly.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Aerosol Sampling Studies

Study continued of errors resulting from sampling air streams at rates much lower than isokinetic. At particle transporting velocities near two mph in the wind tunnel it was found difficult to produce a uniform velocity across the duct in the region chosen for withdrawing a sample. Satisfactory sampling reproducibility on isokinetically-obtained samples, was not achieved. At velocities greater than about 7 mph, isokinetic samples cannot be obtained with the collection filter exposed directly to the air stream because of the relatively high pressure drop of the membrane filters used at the required flow rate through the filter. Inlet nozzles for permitting the higher inlet velocities, yet allowing lower volumes through the filter were tried and found unsuitable because of particle retention on the inner surfaces of the inlet nozzles.

For 10 - 12 micron ZnS particles moving at 7 mph, and sampled at an air flow of 1/30 of isokinetic, it was determined that 2.2 times as many particles will be found on the non-isokinetic filter as would be expected from the volume rate through the filter. Further verification of this and other correction factors is to be obtained.

Generation of Monodisperse Aerosols

The spinning disc aerosol generator was operated over a range of disc speeds and dye feed concentrations to determine the effect of these variables on particle size and concentration. Near monodisperse aerosols were consistently produced using an alcohol solution of methylene-blue and uranine.

The average density of particles produced was measured and found to be 70 - 80 percent of theoretical density of the organic compounds used. The lower density is likely due to voids within the particles developing as the last traces of solvent evaporate.

ISOTOPES DEVELOPMENT - 08 PROGRAM

Fission Product Encapsulation

Two successful strontium oxide compacts were made during the month by high energy compaction. In the first, SrO was prepared by decomposing $SrCO_3$, at 1300 C, followed by loading into a can, heating to 1200 C for 30 minutes, and compacting. Use of a new tungsten carbide punch resulted in a pressure of 413,000 psi. Measured density was 5.0 g/cc, compared with 4.7 g/cc theoretical. The second compact was prepared by mixing strontium oxide with 25 w/o powdered iron prior to compacting. Although a "first try", a reasonably uniform cermet was achieved and the density was comparable to the previous compact.

A single attempt to prepare a strontium fluoride compact was unsuccessful, due to excessive outgassing during a-tempted weld closure. The fluoride had been prepared by aqueous precipitation and probably not adequately dried. Future experiments will be made with powder dried at a higher temperature or prepared by anhydrous techniques.

Bremsstrahlung Shielding Calculations

Because of increasing interest in the heat-source application of beta-emitting fission products, the problem of calculation of bremsstrahlung spectra has been reinvestigated, and a computer program is being written which should result in more rapid and more accurate results than those obtained by the approximation methods employed earlier. The program will be used to obtain bremsstrahlung spectra for $SrTiO_3$, SrO, Sr solutions, Pm_2O_3, CeO_2, etc., and certain other radionuclides of interest. Once the spectra are obtained, a computer code for bremsstrahlung shielding calculations will be written.

W. H. Reas

Manager
Chemical Laboratory

WH Reas:cf

DECLASSIFIED

DECLASSIFIED

BIOLOGY LABORATORY

A. ORGANIZATION AND PERSONNEL

Dr. William H. Matchett joined the Radioecology Operation as a Biological Scientist on August 27, 1963.

GENERAL

Background counts in 100-F have risen to such an extent as to make impossible the counting of low level samples. Presumably this is due to Ar^{41} from reactor effluent vapors permeating from breaks in the line.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Dowell F-33

A surfacant with an alkyl phenol derivative as an active constituent, Dowell F-33, is being used to clean and condition the pipes at NPR. To help evaluate the effect of river disposal of this material on fish, acute toxicity tests on juvenile trout were conducted. Some of the key observations are summarized below:

Concentration of F-33, ppm	No. of fish	Per cent mortality	Per cent loss of equilibrium
90	20	100 in 1 hr.	100% in 10 min.
60	20	100 in 2 hrs.	100% in 10 min.
40	20	100 in 2 hrs.	100% within 30 min.
30	20	100 in 22 hrs.	100% within 2 hrs.
20	20	50 in 22 hrs.	100% within 2 hrs.
15	20	0 in 22 hrs.	None
10	20	80 in 5 hrs.	100% within 3 hrs.
5	20	0 in 24 hrs.	None
1	20	0 in 24 hrs.	None
0	20	0 in 24 hrs.	None

Columnaris

Observations on the mortalities of juvenile trout from the endemic fish disease, columnaris, in hatchery troughs under different water temperatures continue. Little difference in mortality is observed between groups of fish in regular river water temperatures and in waters chilled 4 F cooler. On the average, the total mortality for these groups for the eight-week period is 4%. The groups in waters heated 4 F warmer than river temperatures, however, suffered a greater mortality of about 20%. The frequency distributions of the weekly mortalities of different age groups reared in

river waters indicate that the peak or mode tends to occur earlier in the season for the younger fish than the older fish. The mode for the youngest group occurred at week three in waters of 63-64 F compared to the mode for the oldest group which occurred at week eight in waters of 71 F. The results for eight weeks of test suggest a more complex relationship than a simple direct correlation between water temperatures and mortalities from columnaris.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Strontium

A series of Sr^{90} sources for use as standards for autoradiographic-dosimetric studies were prepared from plaster-of-paris to simulate Sr^{90} contaminated bone. Surface radiation dose rate measurements were made on nine of these sources, using an extrapolation chamber with a vibrating reed electrometer to measure the ionization current (in cooperation with F. L. Rising, Radiological Development, RPO). Volume and weight determinations were made and radioanalysis are in progress on a portion of these sources. These data will permit calculation of the surface dose rates for comparison with the measured dose rates. This series of Sr^{90} sources will be used in autoradiographic studies to determine the radiation dose rates to bones from miniature swine fed Sr^{90} and should provide more accurate estimates of the bone radiation dose rates than those previously used which depended on calculations to estimate dose rates.

Iodine

Concentration of I^{131} in thyroid, milk, and plasma of cows fed 5 μc/day with no supplemental stable iodine was measured. Thyroid levels of I^{131} showed gradual increases of 40-80% in the three cows during the month, possibly reflecting a seasonal or lactation-influenced change in thyroid function. Milk level of I^{131}, however, remained nearly constant. Iodine-131 plasma/milk ratios were determined twice during the month. In the first study, one cow showed no increase in milk levels over that observed in plasma; another cow showed a 50% increase in milk concentration and the third cow had a three-fold increase. Two weeks later, milk/plasma ratio had increased slightly in all three cows.

Promethium

Two young adult female miniature swine were fed 100 mc Pm^{147} and excreta collected for a ten-day period after which the animals were killed and all tissues radioanalyzed. This experiment is a repeat of an earlier study in which lower levels of Pm^{147} were fed. The higher level was administered so that tissue concentrations could be more easily determined by radioanalysis and the very low gastro-intestinal absorption determined more accurately to verify previous results (less than 0.5%).

DECLASSIFIED

Radium

Three female miniature swine administered 6.4 µc Ra^{226} intravenously per kg body weight at about three years of age have exhibited a progressive nephritis at 7-9 months post-injection. One animal succumbed nine months after injection and grossly was observed to have a severe nephrosclerosis. One of the surviving animals manifests a very serious nephritis, the other survivor is in an earlier stage as evidenced by increased serum levels of creatinine and urea nitrogen. A hippuran-I^{131} renogram showed the latter animal had more severe renal damage than the serum chemistry indicated. This suggests that the animal is in a compensated state, maintaining blood urea nitrogen and creatinine at only slightly elevated levels, yet probably having severe structural alterations.

All six of the female miniature swine injected with 6.4 µc Ra^{226}/kg body weight at about three years of age have shown renal damage. One of four females injected at one year of age has shown renal damage and succumbed. No definite renal damage was observed in nine male miniature swine injected at one year of age or younger.

Neptunium

In efforts to determine the mechanism of fatty liver production in Np^{237} poisoned rats, C^{14}-labeled palmitate was fed and the production of $C^{14}O_2$ measured as a function of previous Np^{237} administration. Preliminary results suggest a decreased CO_2 production in female Np^{237}-treated rats, but additional studies will be required to establish significance. If borne out, these results suggest that at least part of the fat accumulation is due to a decreased rate of oxidation in the liver.

Plutonium

Techniques are being developed to permit the transplantation of Pu-burdened adrenals to adrenalectomized, Pu-free rats, in order to study the effect of Pu in the adrenal independent of other Pu effects. This is a technically difficult procedure and no results can be reported at this time.

A group of rats were prepared with biliary fistulae draining into glass containers implanted within the abdominal cavity. Eight such animals were injected with Pu^{239} to determine comparative excretion via biliary, fecal, and urinary routes. The effect of DTPA treatment will also be studied. Results are not yet available.

Four miniature white swine were injected subcutaneously on each foreleg with 5 µc of Pu^{239}(IV) nitrate. This was followed by intravenous injections of $Na_3CaDTPA$ at a level of 0.0014, 0.014, or 0.14 mg/kg to determine the effect of the quantity of the chelating agent injected on the amount of Pu^{239} excreted. No results are available yet.

Inhalation Studies

One of the four dogs exposed to aerosols of Ce-$Pr^{144}O_2$ eight months ago has died, apparently as a result of heavy lung irradiation damage. Of the initially deposited 2 mc, approximately one-half remained at time of death, with 21% of the total body burden in the lung, 27% in the skeleton, and 44% in the liver. The highest concentrations were found in the bronchial lymph nodes, followed by liver, lungs, and skeleton.

The sacrifice schedule (four and eight weeks post-exposure) for dogs exposed to Pm^{147} perchlorate (inhalation or intravenous injection) has been completed. Tissue analyses of the animal which received 9.1 mc by injection showed a body burden at the end of four weeks of roughly 8 mc, of which 47% was found in the liver, 43% in the skeleton, 4% in the kidneys, and 2% in the muscle. The femur and rib sections showed the greatest concentration of skeletal activity.

The presence of a bronchiogenic carcinoma in the left diaphragmatic lobe has been confirmed in a dog which died two and one-half years post-exposure to $Pu^{239}O_2$. The final body burden was 1.8 µc. Two additional dogs have died two and three-fourths years post-exposure to this same isotope, with the lungs in both cases showing possible tumors. Death in all cases was probably due to radiation damage to the lung.

Studies of rat lung and liver tissue fractions at two and one-half months after exposure to Pu^{239} nitrate has shown activity spread throughout the various cellular fractions. The nuclei contained about 30-40%, while mitochondria and microsomal fractions showed 10-20% of the total activity. These findings are markedly different from those of previous studies using Pu^{239} oxide (May 1963), which indicated 72% of the activity in the nuclear constituents.

Copper and Radiation Effects

The LD_{50} for subcutaneously injected cupric acetate in rats appears to be in the range of 20-40 mg/kg. A 40-50% decrease in erythrocytes, packed cell volume, and hemoglobin was noted two days after such doses. Hemoglobinuria was observed within 24 hours, reached a maximum on the third day, and had almost disappeared by the sixth day.

Erythropoietin Assay

Previously, we reported that high levels of erythropoietin (a red blood cell stimulating factor) were noted in the serum of a F_1 generation pig on the 625 µc Sr^{90}/day level shortly before death. In an extension of this study, various quantitative tests for erythropoietin have been investigated. In the present determination, serum from miniature swine are injected into "polycythemic" mice. These mice are subsequently injected with Fe^{59} and the relative uptake of Fe^{59} by the erythrocytes is used as a measure of the "erythropoietin" level in the injected serum. To calculate the percentage uptake of the Fe^{59}, the blood volume of the mice must be known. Investigators using this assay method have generally

assumed that the blood volume of the "polycythemic" mouse is the same as that of the "normal" mouse. To determine if this assumption is valid, blood volume determinations were made on young CF_1 female mice at ten days after they were removed from a reduced atmospheric pressure chamber (~38 cm Hg) in which they had been held for four weeks. Mice that were not exposed to hypoxic were used as controls. The blood volume determinations were made using Fe^{59}-labeled cells.

The preliminary results suggest that the blood volume of the "polycythemic" mice is greater than that of the "normal" mice, and the increase is due to an increased blood cell volume. The increased blood cell volume results from the exposure to hypoxia.

(This study is being repeated in an attempt to verify these results. Additional studies are in progress to determine the blood volume in mice exposed to hypoxia for shorter periods of time.)

Drugs

Clinical evaluation of an "investigational-use" tranquilizer were performed on 24 miniature male swine. This agent (RO5-2807) appears superior to any ataractic agent used in the past when given parenterally. Side effects were minimal with ataxia being the most prominent. Further studies in combination with various anesthetic agents are planned.

An anthelminthic (Thiabendazole) currently used on sheep here was administered to two miniature swine and proved to be very efficacious. This agent was highly palatable to these two animals. It is planned to reproduce this study in more animals.

Cell Proliferation in Fish

Groups of young silver salmon acclimated to 10 C fresh water, isotonic salt solution, and hypertonic salt solution were injected intraperitoneally with about 5 μc C^{14}-labeled thymidine and sampled serially. About 650 tissue samples from 50 fish have been prepared for radioanalysis. Peaking of C^{14} activity in tissues taken from fish in fresh water occurred during the first hour, with the exception of blood and gills which showed peak activity at four hours. At 288 hours post-injection the tissues ranked by decreasing C^{14} content were: head kidney (pronephros), gonads, spleen, mid-kidney (mesonephros), gill, liver, mid-gut, pyloric caeca, pseudobranch, stomach, heart, and blood. Head kidney and gonad contained roughly twice the C^{14} that other tissues contained. These results are part of a cell proliferation study in which finer structure will be studied by using tritiated thymidine.

Radiation Effects on Insects

Flour beetles were exposed to neutrons in the PCTR (average energy 800 kev) and to accelerator neutrons (average energy 4.8 mev). The damage, as evidenced by decreased progeny produced, was greater from the PCTR

than from the accelerator neutrons. However, it is possible that the substantial gamma exposure within the PCTR was responsible for the difference in effect. No supplementary gamma exposure was given with the accelerator neutron exposures. It appears that about 460 watt hours of exposure to the neutron-gamma radiation in the PCTR produces complete sterility in this insect.

Plant Nutrition

Data from the field release of I^{131} are as yet incomplete. It is, however, clear that I^{131} is released back into the atmosphere from the leaf surfaces. This was indicated by a less than eight day half-life on the vegetation harvested, and by direct observation of I^{131} in a filter placed in the exhaust line leading off from the chamber in which a plant from the exposure had been placed and air drawn through. The test was not adequately controlled to define the percentage of the deposited I^{131} which is released, but the amount recovered in the filter appeared adequate to account for the short half-time for the I^{131} on vegetation.

Plant Ecology

Selenium-75 placed in the soil at 1 meter and 0.5 meter depths beneath greasewood shrubs was detected in greasewood leaves two months after treatment. The shallower placement contributed the greater amount of Se^{75} to leaves. Uptake of Se^{75} from 1 meter depths was below the depth of annual moisture penetration, indicating uptake from physiologically dry soil.

HA Kornberg
Manager
BIOLOGY LABORATORY

HA Kornberg:amh

TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORY

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposiums

Bair, W. J. Factors affecting retention, translocation, and excretion of radioactive particles. IAEA Symposium on Radiological Health and Safety in Nuclear Materials and Mining, Vienna, Austria. August 30, 1963.

Clarke, W. J. Pathologic effects of inhaled plutonium particles in dogs, The Institute for Animal Pathology, University of Munich, Munich, Germany. August 21, 1963.

Clarke, W. J. Pathologic effects of inhaled plutonium particles in dogs, World Association of Veterinary Pathologists, Hanover, Germany, August 15, 1963.

b. Seminars (Off-Site and Local)

Bair, W. J. Particle inhalation. University of Washington Summer Institute, August 13, 1963. (108-F Building, 100-F Area)

Bustad, L. K. Radioiodine studies in large animals. University of Washington Summer Institute, August 13, 1963. (108-F Building, 100-F Area)

McClellan, R. O. Bone-seeking isotope studies in miniature swine. University of Washington Summer Institute, August 13, 1963. (108-F Building, 100-F Area)

c. Seminars (Biology)

Conte, F. P. Osmoregulation of migratory fish, August 8, 1963.

d. Miscellaneous

Eberhardt, L. L. Population dynamics. Presented at in-service training school, Montana Fish & Game Department, August 26, 1963.

Mahlum, D. D. Metabolic studies with small animals. Yakima County and City Pre-School Workshop, Eisenhower High School, Yakima, Washington, August 29, 1963.

Uhler, R. L. Accumulation of fallout nuclides by plants and environmental interactions. University of Utah, Salt Lake City, Utah, August 8, 1963.

II. Articles Published

a. HW Documents

Nakatani, R. E. and D. H. W. Liu. Bioassay of a descaling alkaline cleaner on young trout and salmon. HW-78446, Unclassified, August, 1963.

b. Open Literature

None

III. Visits and Visitors

a. Visits to Hanford

8/8/63 - R. A. Hammerstrom, J. F. Conrad, D. A. Leith, J. W. Westgate, and D. W. Chapman from Oregon State Fish Commission, Clackamas, Oregon, discussed research with R. E. Nakatani, F. P. Conte, P. A. Olson, and F. P. Hungate.

8/9/63 - G. D. Wendlandt from Economics Labs, Inc., Spokane, Washington, demonstrated products to J. F. Park.

8/13-14 - 30 Summer Institute Students, University of Washington, Seattle, toured and listened to lectures given by F. P. Hungate, et al.

8/14/63 - James F. Young, General Manager, Atomic Products Division, toured by H. A. Kornberg.

8/14/63 - Harry L. Brown, Louis Werner, and Jared J. Davis, Hazelton Nuclear Science Corporation, Palo Alto, California. Investigated business potential, toured by H. A. Kornberg.

8/10/63 - Dr. William Brown, University of Minnesota, St. Paul, discussed research with L. K. Bustad.

8/13/63 - Group from University of Washington, columnaris sampling, visited with R. Wildman and H. Jones.

8/19/63 - A. Chambers and J. McKell and Julia Harden, U. S. Public Health and AEC Trainee, toured by R. F. Palmer.

8/22/63 - Prof. Mangelson and Nolan Mangelson (G.E. summer employee), Utah College, toured by R. F. Palmer.

8/23/63 - Dr. David Moore and two graduate students (B. J. Mason and E. B. Maas), Oregon State University. Presented seminar. Toured by H. E. Erdman.

8/23/63 - Joel McKenney, University of Washington, visited facilities and discussed research with L. K. Bustad and R.O. McClellan.

b. Visits Off-Site

8/1/63 - W. J. Bair presented seminar at University of Utah, Salt Lake City, Utah.

8/1-2 - F. P. Conte and R. E. Nakatani visited the University of Washington and Western Fish Disease Laboratory, Seattle, to discuss research.

8/7/63 - R. L. Uhler presented a seminar at the University of Utah, Salt Lake City, Utah.

8/10-30 - W. J. Clarke visited Germany and London presenting seminars at the University of Munich and at the World Association of Veterinary Pathologists at Hanover, Germany.

8/12-16 - V. G. Horstman attended the American Society of Animal Science Meeting and visited labs at Oregon State University, Corvallis, Oregon.

8/14-16 - H. A. Ragan attended the American Society of Animal Science Meeting and visited labs at Oregon State University, Corvallis, Oregon.

8/19-21 - R. T. O'Brien visited Advanced Electronic Center, Ithaca, N. Y., for consultation and an interview.

8/24/63 - C. E. Cushing attended AIBS meetings at the University of Massachusetts, Amherst, Massachusetts.

8/26/63 - L. L. Eberhardt lectured at a training school, Montana State Biological Station, Flathead, Montana.

8/28/63 - L. K. Bustad worked on ionizing radiation project at the University of Washington, Seattle.

8/28-31 - R. C. Thompson attended the Second International Symposium on the Response of the Nervous System to Ionizing Radiation, UCLA, Los Angeles, California.

8/29/63 - D. D. Mahlum lectured in Yakima for the Yakima High School Science Teachers.

IV. Achievements

None

V. Honors and Recognitions

None

VI. Professional Group or Organization Assignments

None

DECLASSIFIED

APPLIED MATHEMATICS OPERATION

MONTHLY REPORT - AUGUST, 1963

ORGANIZATION AND PERSONNEL

Jon C. Stouffer was transferred from the Technical Graduate Program to a permanent assignment as a mathematician effective August 1, 1963.

OPERATIONS RESEARCH ACTIVITIES

Work on the economic model of the Tri-City Area is ended. The Model is now completed and ready for use in estimating likely impacts of HAPO and AEC changes on the Area's economy. The Model should also be useful in indicating directions for useful Area diversification and growth.

Outage work frequency analysis continues as a basis for refining the system of measurement by reactors. In addition to developing probability models for KW and H reactors, a similar model has been requested for B reactor. The necessary groundwork and analysis is now under way.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

N-Reactor Department

A study is being conducted to determine the feasibility of monthly "quick and dirty" monitoring of control information on fuel element cladding intended for more complete examination on a quarterly basis using the MERCY program.

An outline of a proposed elementary training course in statistics and probability for quality control personnel was reviewed and commented on.

Irradiation Processing Department

Preirradiation dimensional measurements have been taken on fuel elements canned by the Alsi and hot-die-sizing processes. Detailed statistical descriptions of these values in terms of the averages, standard deviations, comparisons of variances have been made for the outside diameters, inside diameters, warp, length, and weight measurements in order to compare the two groups with respect to process control.

R values, which are measures of the temperature imbalance in the water as it leaves a tube, have been systematically calculated as a function of time for tubes of different initial rib heights. Curves which relate the R value to time are being fitted to these data. The resulting relationships are to be used as aids in designing tubes with added protection against corrosion. Further work is being done on enlarging the machine data processing program to eliminate some of the hand calculations needed in determining R.

An analysis of the corrosion rates of welded and nonwelded plates indicates that the average corrosion rates vary significantly for different samples of material and for different corrosion times.

No statistically significant differences were found between UT-4 test response of machined OD cap wafers and as-impacted cap wafers. A similar analysis indicated that machined OD cap wafers give significantly better performance when they are spire etched.

A portion of a production test to compare incremental dimensional stability properties of salt and induction heat-treated ingots was patterned in the form of Graeco-Latin squares so that the tube, ingot and rod number components of variance can be estimated.

Data are being analyzed to determine the reproducibility between different observers of a metallographic technique which estimates the percentage of UAL_3 around the circumferential bond obtained by a transverse cut of a fuel element.

Curves were developed which give the probability that five or more out of eight turbines will respond to immediate and heavy demand situations when some of the turbines are idling and some are not, where the probability that an individual will respond successfully is different for idling and nonidling turbines.

Data based on stud pull results are being analyzed to compare different deoxidation methods in the preparation of nickel-plated cores to be used in the hot-die-sizing canning process.

The relationships between the times and temperatures of can components heated in a bath were studied for different types of can components. The average, standard deviation, and 95 percent confidence limits on the time to reach specified temperatures were given because no simple model serves to relate the variables.

A study indicates that stud pull strength values are larger on the average when the can components used in the hot-die-sizing canning process are cleaned by the standard method rather than with a modification of it.

DECLASSIFIED

The effects of day-to-day variation in pH upon bond strength and pulled nickel plate are being studied.

A study indicated that the pull strength of rails made with transverse grain is significantly greater than for those made with longitudinal grain. Further, though somewhat inconclusive, analyses were made of the effect on pull strength of the use of different welder heads and soap and oil as cleaners.

Chemical Processing Department

During the month the variability of the minor constituent (70-58) control measurements decreased sufficiently to permit changing from three to two sampling positions per part. When more data is available an over-all review of the analytical control required for 74C components will be made.

Assistance was given in an attempt to obtain limits on the shipper-receiver difference for plutonium buttons. The observed differences are considerably larger than any known measurement variance so these limits are based on observed differences.

Testing of the first magnetic tape prepared to numerically control the Sheffield rotary contour gauge was completed at the Schenectady laboratories of Equipment Development Services. Suggestions for improvement have been incorporated into the tape preparation program and tapes made for several additional tests.

Designs were completed for four high-shear-valued power roll forming blanks.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HL

2000 Program

Debugging continued of the FORTRAN language program for the solution of the boundary value formulation of the nonlinear system of second order differential equations expressing mass transfer in the column. Test data from a nine compartment linear approximation to the second order nonlinear system are being used to debug the FORTRAN routine. To date the solution through the top two compartments of the column is working correctly.

4000 Program

A mathematical solution was obtained to a pelleted fuel heat transfer problem which arose as a result of a hazards analysis investigation.

An experiment was designed to evaluate a blow back filter cleaning system as a function of venturi geometry, and spacial and ambient variables. The dependent variable over pressure is to be maximized as a function of these independent variables. A nested set of preliminary experiments are currently in progress to evaluate the apparatus setup error and the basic reproducibility of over pressure measurements.

Eight equations of an empirical nature were developed which predict the amount of Pu-239, Pu-240, Pu-241, and Pu-242 as functions of the percent burnup. Further assistance in related work is being given in obtaining the least squares fits of several sets of data by the NELLY routine for a nonlinear model.

Discussions were held with interested personnel in the interpretation of results obtained by the NELLY routine in the least squares fit of an incremental volume swelling model.

5000 Program

A program is being written to output plots of program data prior to statistical analysis.

Further work was done on the least squares program to calculate individual standard gamma spectra from a composite standard. A new version of the GEM program was put into use which provides for entering raw spectra data with the channel summing being done internally in GEM. This version also prints a listing of the input cards for each case. A number of modifications were made to the IRA system during the month.

Work continued on the crystal indexing program.

6000 Program

A second phase of the Monte Carlo atmospheric diffusion problem was started. A computer program has been written to customer specifications and some preliminary data obtained for study.

The statistical analyses of data from X-irradiation experimental studies on Tribolium were completed. The effects of X-irradiation, temperature, time, species, and grouping on the number of adult progeny by mating pairs of Tribolium were investigated.

An analysis was completed of data which investigates the onset of reproduction for flour beetles, Triboleum castaneum, cultured at different temperatures from the first day of metamorphosis to two weeks thereafter.

The swimming performance of young chinook salmon after the fish were reared in different types of reactor effluents was investigated. Comparisons of

DECLASSIFIED

DECLASSIFIED

swimming performance of young Chinook salmon reared in 0%, 3% PIE, 3% PIEX, 6% PIE and 6% PIEX reactor effluents and then tested in water velocities of 1.8, 1.6, and 1.4 ft./sec. were completed.

Carl A. Bennett

Manager
Applied Mathematics

CA Bennett:dgl

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Fuel Conservation Study

A general equation has been developed to express the efficiency of fissile fuel utilization in a thermal reactor. A variety of individual fuel conservation indices can be developed from this general equation with the "worth ratio" concept. The worth ratio, R, of a fissile material credits it with an equivalent amount of the uranium-235 contained in natural uranium.

As shown in Figure 1, two functions have been chosen to represent the worth ratios of uranium-235 and plutonium in thermal reactors. The first of these, Curve A, is the amount of uranium-235 contained in the feed to a diffusion plant such that a unit amount of uranium-235 is contained in the product, provided the product enrichment is greater than the tailings composition. This can be expressed as

$$R(X) = \frac{1 - X_W/X}{1 - X_W/X_F} \qquad X > X_W \qquad (1a)$$

$$R(X) = 0 \qquad X \leq X_W \qquad (1b)$$

where

R = worth ratio

X = enrichment

X_F = cascade feed composition = 0.007115 for natural uranium

X_W = cascade tails composition

Equation (1a) was derived from the mass balance equations for a diffusion cascade. The maximum value of the worth ratio is $X_F(1 - X_W)/(X_F - X_W)$ or, for example, 1.55 for a tails composition of 0.002531 as in the present AEC uranium price schedule. The minimum value is zero.

The second worth ratio function, curve B in Figure 1, is based on the assumption that the worth of a fissile atom is unaffected by its concentration as long as the concentration is sufficiently high. This can be expressed as

$$R(X) = 1 \qquad X > X_w \tag{2a}$$

$$R(X) = 0 \qquad X \leq X_w \tag{2b}$$

where R, X, and X_w are as defined for Equation (1).

Using the worth ratio functions, the general fuel conservation equation for uranium enrichment of a thermal reactor can be written as

$$\psi = \frac{kE}{X_i\ R(X_i) - FX_oR(X_o) - VCZR(Z)} \tag{3}$$

where

ψ = fuel conservation index, megawatt days per gram "lost"

E = reactor exposure, megawatt days per ton

k = 1.1023×10^{-6} ton per gram

X_i = initial uranium enrichment, fraction

X_o = final uranium enrichment, fraction

Z = fissile concentration of bred plutonium, fraction

F = ratio of uranium discharged to uranium charged

V = assigned plutonium value ratio

C = ratio of plutonium discharged to uranium charged

R = worth ratio given by either Equation (1) or Equation (2).

Of the parameters appearing in the right-hand side of Equation (3), all, except V and R, are determined completely by the burn up

calculation. R, the worth ratio, is determined by the burn up calculation and the specification of a diffusion plant tails composition. V, the plutonium value ratio, is an arbitrary number which, for example, can be used to account for the higher value of plutonium in a fast reactor. In most computations, the value of V has been unity, although some computations using 1.25 have been made.

A variety of particular fuel conservation indices can be generated with Equation (3) by selecting the appropriate worth ratio function. Some of the possibilities are shown in Table I. Note that there is a total of twenty-seven possible combinations, but that it is seldom meaningful to use a function other than Equation (1) with the uranium charged.

Example calculations of the various fuel conservation indices are shown in Table II. These data were computed for minimum fuel cost operation of a water moderated reactor simulation for three different uranium price schedules. For comparison, note that very nearly one megawatt day is produced by the complete fissioning of one gram of uranium-235. Also shown in Table II, are the results for the minimum fuel cost operation of a reactor designed for high neutron economy. (That is, high fast fission factor, low neutron temperature, D_2O moderation.)

It may be noticed that, in some cases, two or more of the conservation indices are identical. This occurs when the discharged uranium composition is less than the tails composition for then, the worth ratios are all identically zero. It is evident that a wide range of values is possible in any given situation.

It is planned to complete the fuel conservation study by computing indices for a range of reactor types and fueling modes. These will include plutonium recycle systems and various unconventional reactor types.

Reduced Spatial Concentration Fuels

A reduced spatial concentration fuel containing uranium-238 and thorium-232 enriched with plutonium was studied to determine fuel cycle costs and reactivity response characteristics. Fuel spatial concentration was 15 percent of normal UO_2 density. It was assumed that the remaining 85 percent volume was filled with an inert diluent of negligible cross section.

Figure 2 shows the relatively flat reactivity response curve for one case of the series.

TABLE I

TABULATION AND DESCRIPTION OF SOME POSSIBLE FUEL CONSERVATION INDICES

Indices derived from Equation (3) by the choice of a worth-ratio function

Index Neutrons Computed	Worth Ratio Function[1] Uranium Charged	Uranium Discharged	Plutonium	Comment
Yes	A	A	A	Equitable credit of all isotopes
Yes[2]	A	A	B	Very nearly the same as above
Yes	A	A	N	No credit for bred plutonium
No	A	B	A	Unfairly credit to discharged uranium
Yes	A	B	B	Same credit to discharged fuel
No	A	B	N	Not as useful as Type AA-
No	A	N	A	Very nearly same as below
Yes	A	N	B	Used by Dietrich (γ)
Yes	A	N	N	Minimum possible system value (ψ_{min})
Yes	B	B	B	Heat produced per fissile gram destroyed
Yes	B	N	N	Minimum possible reactor value

(1) Letters A and B refer to the curves in Figure 1. The letter N means that the quantity is neglected.

(2) Computed with V = 1.0 and with V = 1.25. See Equation (3)

TABLE II

SAMPLE FUEL CONSERVATION INDEX CALCULATIONS

Reactor Operated at Minimum Fuel Cost

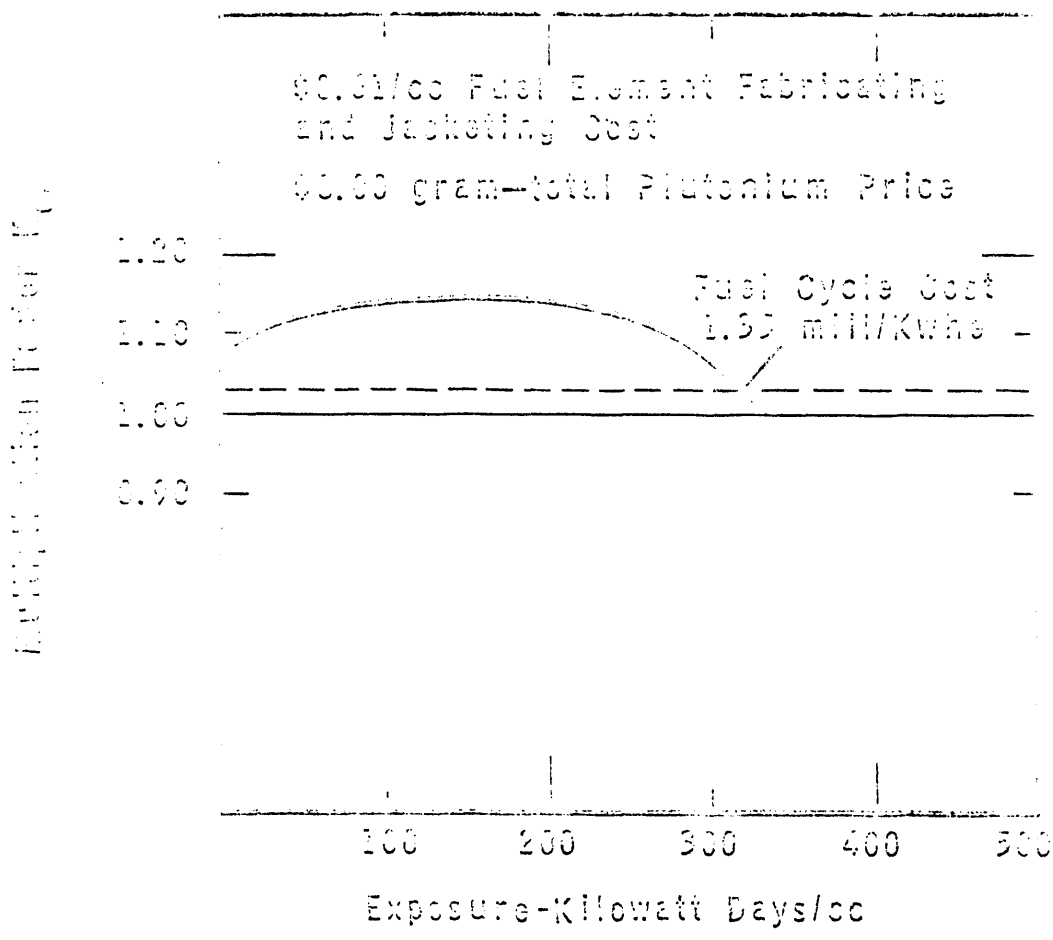
Cost of Natural Uranium $/kg Uranium	Optimum Tails Composition,[1] %	Uranium Enrichment Initial	Uranium Enrichment Final	Exposure Mwd/t	Fuel Cost, Mills/kwhe	Fuel Conservation Indices[2] AAA	AAB	AAN	ABB	ANB	ANN	BBB	BNN
Water Moderated Reactor Simulation													
13.00	0.319	1.64	0.23	23,700	1.08	1.71	1.36	1.09	1.36	1.36	1.09	2.79	1.59
23.50	0.253	1.59	0.24	22,300	1.29	1.84	1.54	1.18	1.54	1.54	1.18	2.81	1.55
100.00	0.116	1.48	0.28	19,000	2.44	2.38	2.16	1.46	2.36	1.80	1.29	2.84	1.42
Reactor With High Neutron Economy													
23.50	0.253	0.99	0.07	21,800	0.90	4.99	3.36	2.10	3.36	3.36	2.10	4.84	2.43

(1) For a separative duty cost of $30 per kilogram uranium

(2) As defined in Table I

FIGURE 1

INFINITE MULTIPLICATION FACTOR AS A FUNCTION OF EXPOSURE FOR A REDUCED DENSITY FUEL*



Composition	Percent of Total	Plutonium Composition
Pu 239	20.23	76% Pu 239,
Pu 240	4.50	10% Pu 240,
Pu 241	1.33	5% Pu 241,
Pu 242	.27	1% Pu 242
Th 232	48.03	
U 233	24.44	

*Fuel Density was 15% of Normal UO_2 Density

Computer code parameters were adjusted to simulate burn up of this fuel in a hard spectrum, light water moderated reactor with a moderator-to-fuel volume ratio of 1.8. The fuel use charge was 4-3/4 percent, economic interest 12 percent, fuel element fabrication and jacketing cost $0.61/cc and plutonium price $8/gram total. The fuel cycle costs for this fuel would be slightly changed if a plutonium price of $10/gram fissile were used.

The fuel cycle cost shown in the figure was calculated at an exposure of 320 kw-days/cc. This corresponds to an exposure of about four years at a specific power level of 220 w/cc (equivalent to 20 mw/t at normal oxide density). This was not the lowest fuel cycle cost observed; however, this case was reported because of its flat reactivity characteristics and relatively long fuel life. The optimum moderator-to-fuel ratio has not been determined.

It should be noted that the addition of relatively small amounts of thorium-232 and uranium-238 to the low-exposure plutonium resulted in a fuel exhibiting Phoenix characteristics. This experiment indicates that the Phoenix effect may be achieved with plutonium fuels having insufficient plutonium-240 fertility by adding small amounts of thorium-232 and uranium-238, and operating the fuel in a hard spectrum so a large proportion of the neutron captures in the fertile material are in the resonance region.

Plutonium-238 Production

Plutonium-238 is often considered, as is curium-244, as a heat source for various space missions. The calculation of curium-244 production has been completed, and now plutonium-238 production is being analyzed.

When uranium-235 is used as a fuel, a portion of the neutron captures results in uranium-236 formation; which, upon another capture, forms neptunium-237, and with still another capture, plutonium-238 is formed. Because in power reactors fueled with uranium, uranium-238 is present from which large quantities of the other plutonium isotopes are formed in comparison with the plutonium-238 -- the plutonium-238 is usually so dilute in this mixture that it is of no practical use.

The best way to achieve high purity is to separate neptunium-237 from the discharged fuel and then irradiate it in a separate capsule. This means that the important isotope in the plutonium-238 production is the neptunium-237. However, this difficulty can be overcome at some increase in expense by fueling the reactor with nearly pure uranium-235 (93%); this reduces the formation of other plutonium isotopes and, thus, direct production is achieved. The increase in

expense of burning fully enriched uranium can be offset, if the price paid for plutonium-238 is high enough.

Besides recycling the neptunium in capsules, the production of neptunium-237 can be increased by also recycling some of the uranium with high uranium-236 concentrations.

Calculations have been made for incidental production of neptunium-237 in various power reactor types and also in some special types with higher average neutron energies (harder spectra). The ratio of uranium-236 formation to fission in uranium-235 is increased with harder spectra, also the uranium-236 cross section is very responsive to spectral hardness, increasing rapidly with hardness. Figure 3 shows the relative production ratio of neptunium-237 for various spectral indices "r", (Higher r's indicate harder spectra.) for reactors using slightly enriched uranium fuel. Also included is a curve showing production from a pressurized water simulation, where the spectral index (r) is allowed to vary with enrichment and burn up. "r" varied from 0.12 to 0.155 during these calculations.

Code Development

The PLOTTER code, which summarizes the burn up information from MELEAGER, has been modified in that a choice of the data to be tabled can be made and also summarized in a single table containing data from all enrichments.

A new code called CURVES was written to utilize a curve plotting machine installed by EDPO. The code curver fits point data and then plots a continuous line using the curve plotting machine. Many curves may appear on one sheet of standard graph paper.

Seed-Blanket Systems Study

Preliminary computations using the ALTHAEA code were begun to determine the behavior and economics of seed-blanket systems. Geometry similar to APWR was assumed and equal seed and blanket volumes were used. Natural and 1.5% enriched uranium at 96% of theoretical density were considered for the blanket, while three concentrations of fully enriched uranium and plutonium were considered for the seed. Exposures computed for the blanket are very high but seem to be consistent with values considered by others studying seed blanket systems.

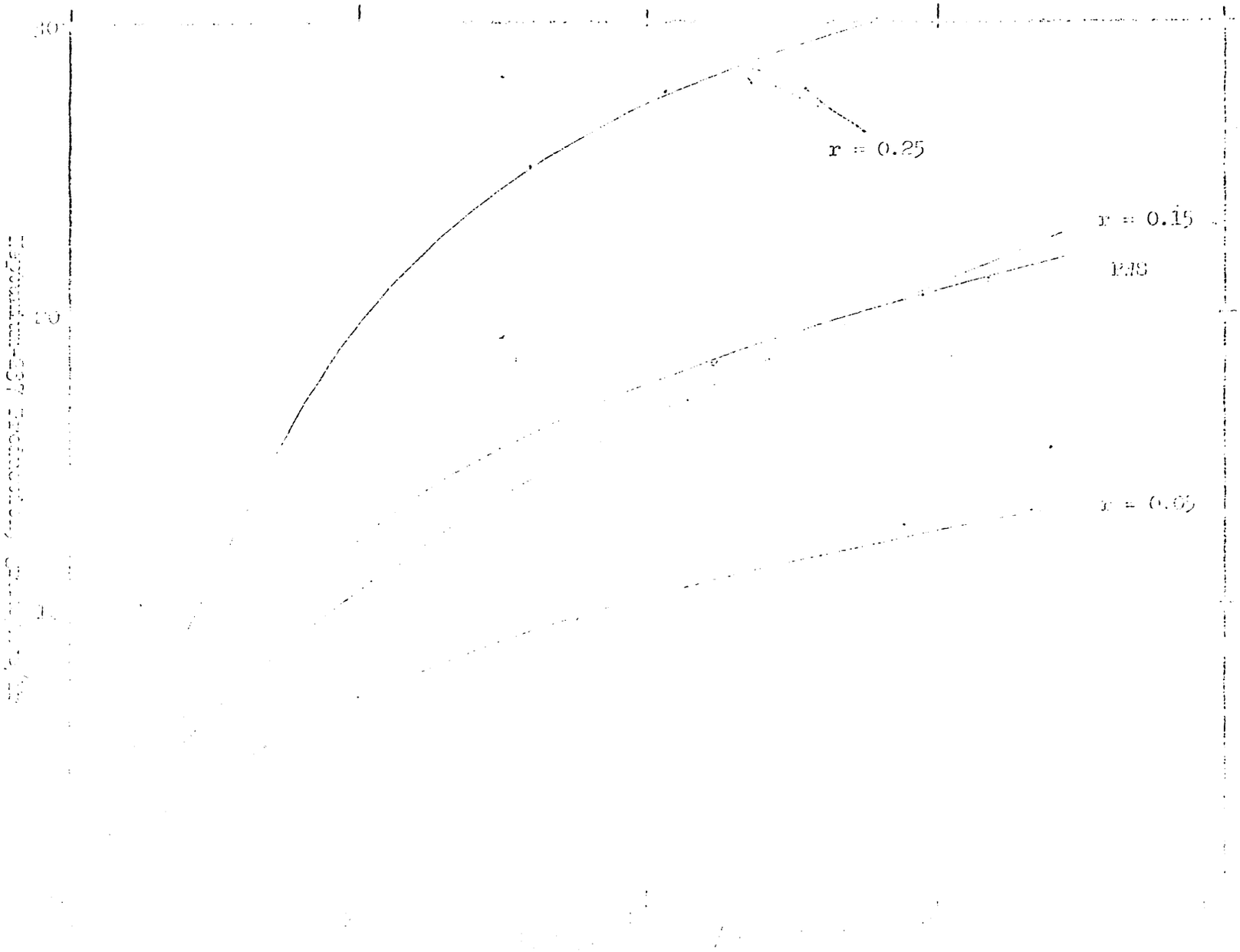

r = 0.25
r = 0.15
r = 0.05

TABLE III

SEED BLANKET CALCULATIONS

Blanket	Seed	Exposure Mwd (1)	Max/Min Power Ratio	Average Power Mw (1)
Natural U		63400	1.9	21
	U-235 (2)	63000	2.7	21
Natural U		56000	2.1	19
	U-235 (3)	68000	2.2	23
Natural U		49000	2.2	18
	U-235 (4)	69000	2.1	25
1.5% Enr.		63000	1.5	21
	U-235 (3)	67000	1.8	22
1.5% Enr.		67000	1.5	19
	Pu (5)	84000	2.3	24
1.5% Enr.		63000	1.5	19
	Pu (6)	78000	2.3	23

(1) Exposure and power level for one ton of blanket. The seed power level and exposure are given for an equal volume.

(2) The seed was fully enriched uranium-235 at a concentration of 0.2 g/cc present as finely divided UO_2 in zirconium matrix. Six seeds were used during the life of the blanket.

(3) Same as (2) except the concentration was 0.4 g/cc and three seeds were used.

(4) Same as (2) except the concentration was 0.6 g/cc and two seeds were used.

(5) The seed contained 93% plutonium-239, 7% plutonium-240 at a concentration of 0.4 g/cc present as finely divided PuO_2 in a zirconium matrix. Three seeds were used.

(6) Same as (5) except the plutonium was 76% plutonium-239, 18% plutonium-240, 5% plutonium-241, and 1% plutonium-242.

Nuclear Safety Activities

Approval of AEC was obtained to operate the Fuel Element Rupture Test Facility in PRTR using a PRTR pressure tube with PRTR fuel elements. Approval of the Technological Hazards Council was requested and the matter will be discussed at the September meeting of THC. A report on the reactor process tube surveillance program for PRTR was transmitted to the THC. A revision to the schedule for destructive testing of tubes is proposed from one tube per 30 operating days to one tube per 90 operating days.

Supplement 3 of the Safeguards Analysis for the Plutonium Recycle Critical Facility was transmitted to AEC and the THC for approval. Approval for operation of PRCF at an increased power level, 10 kw rather 100 watts was also requested of AEC and THC. The increased power level is needed to obtain meaningful data on irradiated fuel elements because of the estimated neutron flux from γ-n reactions.

OTHER ACTIVITIES

Hanford Isotopes Production Plant Study

The report on "The Hanford Isotopes Production Plant" (HW-77770) was revised in minor details in accordance with an AEC-DID request and released. No changes in data or conclusions were necessary. With the release of this document the assignment has been completed. Summarized data from the report appeared in "Chemical Week" and "Nucleonics Week" concurrently with its release; this illustrates the timeliness and general interest in the subject matter.

Heat Source Radioisotopes Studies

All candidate heat source radioisotopes with half lives in the range of 100 days to 2000 years, and which require an isotopic separation process for either or both target and product, were reviewed to identify those which may be sufficiently attractive in half life, type of radiation, production feasibility, and potential cost to

justify intensified efforts on the study of unique isotope separations processes.

Of the twenty-five candidate isotopes, only four appear to be suitable. Two of these (cesium-134 and holmium-166) are gamma emitters and thus would be competitive with the cheap fission products, strontium-90 and cesium-137, and therefore need not be considered seriously. The remaining two are thulium-170 and thallium-204. Thulium-170 is predominately a beta emitter but with weak gamma. Shielding requirements would thus very likely be controlled by bremsstrahlung. Its short half life of 127 days limits its widespread consideration. Without isotopic separation it would be produced cheaply and probably with power densities in the oxide form comparable with plutonium-238 (metal). Its short half life and minor shielding expectation make it appear competitive with costly polonium-210 and curium-242, Thallium-204 is a pure beta emitter. Its half life is nearly four years. Its moderate beta energy (0.77 Mev) assures minimum shielding problems with the bremsstrahlung. Although it also may be suitable without isotopic separation, such processing would assure concentrations of unusually high power density. This and the fairly long half life make it suitable for many uses where costly plutonium-238 (or promethium-147) is required.

This study (reported in HW-78658) supports intensified scouting efforts on investigation of special isotope separation techniques such as the Szilard-Chalmers process.

Milton Lewis

Acting Manager
Programming

M.Lewis:bc

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF AUGUST 1963

A. ORGANIZATION AND PERSONNEL

Dr. B. B. Ewing, Professor of Sanitary Engineering, University of Illinois, and H. N. Hauser, Instructor, Columbia Basin College, terminated their summer employment with Environmental Studies and Evaluation August 23, 1963. Janice B. Persons, on temporary assignment with Environmental Studies and Evaluation, transferred to a permanent assignment as secretary to the Manager, Radiation Protection Operation, replacing Lillian W. Omberg who terminated August 23, 1963. I. C. Roberts joined the Environmental Studies and Evaluation staff on August 1, 1963, following assignment with the IAEA in Vienna. R. A. Walters, Engineer, Internal Dosimetry, terminated employment August 16, to study for an advanced degree in Radiobiology at Rochester University.

B ACTIVITIES

Occupational Exposure Experience

Four new plutonium deposition cases were confirmed by special bioassay analysis during the month. In each case, the body burden was estimated to be less than 1 percent of the maximum permissible body burden (MPBB plutonium with bone as reference = 0.04 μc). Three employees were involved in inhalation incidents at the Weapons Manufacturing Building (234-5) on May 6, 1963, May 23, 1963, and April 25, 1963. A fourth employee was involved in an inhalation incident at Plutonium Fabrication Pilot Plant Building (308) on May 8, 1963. The total number of individuals who have received internal plutonium deposition at Hanford is 325. With the death of one employee, the number of deposition cases currently employed is 236.

There were six incidents involving 6 employees that required special bioassay sampling for plutonium analysis this month. The following is a brief description of the more significant incidents.

A process operator at the Weapons Manufacturing Building (234-5) received a plutonium contaminated injury at 1050 on August 2, 1963. The employee was performing a gauging operation in Hood 46-C when the plutonium metal piece slipped, cutting through the hood and surgeons glove. A survey of the wound area with a Juno revealed approximately 600,000 d/m of plutonium present. After successive flushings, the contamination was reduced to 5000 d/m. Examination at the plutonium wound counter showed 0.017 μc plutonium initially present at the wound site. After tissue excision, examination with the plutonium wound counter revealed 7×10^{-4} μc remained. The employee is a previous plutonium deposition case with a body burden estimated to be less than 10 percent of the maximum permissible body burden.

An operator at the Weapons Manufacturing Building (234-5) received a plutonium nitrate contaminated injury at 2015 on August 5, 1963. The injury occurred when a piece of glass tubing broke as the employee was attempting to connect it to tygon tubing. A survey of the wound area revealed approximately 20,000 d/m of plutonium present. After several attempts at decontamination, contamination around the wound was reduced to less than 500 d/m but a direct survey of the wound continued to show approximately 20,000 d/m present. Examination at the plutonium wound counter at 2145 showed 0.042 μc initially present at the wound site. After tissue excision the wound count was less than 1×10^{-4} μc. (The detection limit for the plutonium wound counter is approximately 1×10^{-4} μc.) The employee is a previous plutonium deposition case with a body burden estimated to be less than 1 percent of the maximum permissible body burden.

An operator at the Weapons Manufacturing Building (234-5) received a plutonium contaminated injury at 2315 on August 7, 1963. The injury occurred when the employee was performing housekeeping in maintenance hood 18-C. A survey of his surgical glove showed 10,000 d/m contamination and blood smears counted 300 d/m. Examination at the plutonium wound counter at 0100 on August 8, showed 2.8×10^{-3} μc initially present at the wound site. After excision of tissue the wound count was approximately 6×10^{-4} μc. The employee is a previous deposition case with a body burden estimated to be less than 1 percent of the MPBB.

The above cases bring the total number of plutonium contaminated injuries for the year to 16 with 11 requiring excision. In 1962, there was a total of 15 contaminated injuries with 8 requiring excision.

Also during the month, there were three incidents involving 5 employees that required evaluation for possible intake of radioactive materials. These incidents are summarized below.

Two CPD employees were exposed to airborne fission product contamination at the Purex Building (202-A) at 1745 on August 2, 1963. In addition, one of these employees received an external exposure as a result of high level clothing and skin contamination. The incident occurred when the employees were attempting to repair a leaky flange. Prior to replacing the gasket at the flange it was necessary to bleed the line. When the valve was opened, nitrate solution sprayed on the two employees and in the area where they were working. Because of the acidic solution, the employees immediately showered. After showering, contamination on the chin and chest of one employee was 40,000 c/m. Contamination on the other employee ranged from 6000 to 8000 c/m. Clothing from the first employee measured approximately 12 rads/hr. Evaluation of the employee's film badge dosimeter showed approximately 0.3 rads for the period the badge was worn. Autoradiographic examination of the contaminated coveralls indicated the contamination was fairly uniform in the area where the film badge dosimeter was worn.

Therefore, the dose recorded by the dosimeter was considered to be a reliable estimate of the skin dose received by the employee. Examination at the whole body counter on August 5, 1963, revealed a trace quantity of ruthenium-103 in both employees.

A Hanford Laboratories millwright received nasal contamination of 70 c/m and skin contamination ranging from 2000 to 4000 c/m while repairing a master slave manipulator in the 327 Building on August 8, 1963. The equipment was decontaminated on the outside, however, upon disassembly contamination was found on the inside surfaces. A monitor noticed an increase in reading on the constant air monitor and notified the employee to put on respiratory protection. Examination at the whole body counter revealed the presence of 2.7×10^{-2} μc of zirconium-niobium-95. (MPBB zirconium-niobium-95 (total body) is 20 μc.)

A fire occurred in a steel drum containing approximately 200 pounds of uranium fines at the NRD Fuels Building (333) at 0025 on August 6, 1963. The firemen placed a hose in the drum to cool the uranium fines causing the material to flow out in the vicinity of the oxide burner that is located to the east of the 333 Building. The fire truck and firemen were released from the area by a PRTR monitor. Surveys in the morning indicated that approximately 14,000 square feet of road and ground were contaminated from 100 c/m to 100,000 c/m with uranium.

Several other incidents of significance which occurred during the month are summarized below:

A critical radiation alarm signal was received at the 300 Area patrol headquarters from the NRD Fuels Building (333) at 1109 on August 7, 1963. The crash alarm system was immediately initiated. The alarm did not sound in the 333 Building nor did a panel light indicate any abnormality. Therefore, the personnel did not evacuate the building. Investigation revealed that the signal light in the patrol headquarters was caused by a malfunction resulting from work on the telephone system.

During the shipment of two casks from Hanford to the National Reactor Test Station in Idaho by commercial carrier, one of the casks was accidentally tipped over. Because of the ICC Class D label attached to the shipment, the carrier immediately notified RLOO, AEC. Investigation quickly revealed that both casks were empty and that the Class D label was attached to the shipment in error. Contamination and radiation surveys performed by Radiation Monitoring had shown that the casks could be shipped without such restrictions. Accordingly, the carrier was informed that the shipment of the casks should continue. Administrative steps were taken to assure that Class D restrictions be imposed on such shipments only when required.

Forty-one employees in PRTR received a radiation dose from internally deposited Tritium equal to or greater than 25 mrems during the period. The maximum dose received by one employee was 290 mrems and the average for the 41 employees was 76 mrems.

Environmental Experience

Concentrations of fallout materials in the air of the Pacific Northwest averaged 5 pc/m^3 of Beta activity during the four week period ending August 23, 1963, the same value that was observed during the previous month.

The following 262 biological, produce, and food samples were obtained for radiochemical analysis:

Milk	181 gallons	57 samples
Fish		82 samples
Pasture grass	96 pounds	48 samples
Oysters	4 pounds	2 samples
Ground round	6 pounds	3 samples
Dried Oysters		10 samples
Beef thyroids		19 samples
Fresh fruit, vegetables and berries		41 samples

One aerial monitoring flight was made during the month following standard pattern 2-E (S.E. portion of Washington State) to provide background measurements and training.

Stack sampling equipment was placed in operation at B-Plant August 1. The sample of stack gas, which is collected by a five fingered probe at the 50 foot level, is drawn through a manifold. The particulates in the air sample are collected on a 2-inch Gelman type AM-5 membrane filter. The build-up of activity is measured by a 2-inch by 1/16-inch NE 102 scintillating crystal and recorded on a 5 cycle logarithmic counting rate recorder. Performance of this equipment to date is very good.

Studies and Improvements

Changes were made in the formats of the Employee Pencil Exposure Report and the Pencil Exposure Errors Report to accommodate the new mechanized pulse pencil reader. The changes were necessary because the input of the new reader was not compatible with the existing report format.

The report "Hanford Personnel Exposure - Employees Whose Dose for the Period Substantially Exceeds the Average Dose Received per Period", was sent to the field for the first time this month. This exception report was primarily designed to assist in the control of the radiation exposure received by individuals whose work assignments require only occasional surveillance by radiation monitoring personnel. The report lists those individuals whose dose for the current period is substantially higher than their average dose for the last twelve months, based on the four week film badge dosimeter period.

The Mobile Whole Body Counter laboratory was displayed in several outer area locations during the month. Four-hundred and eighty employees visited the new facility during August.

At the present time neutron film badge dosimeters are left in service for only two weeks because of the track fading problem. The original track fading study that was completed in 1957 indicated a 25% fading loss in proton recoil tracks in 21 days. However, this fading rate loss was based on the fading which occurred if the entire dose was received on the first day. A recalculation of the fading loss for a two week badge period and a four week badge period, using the same fading rate but assuming that the dose occurs in equal increments each work day during the badge-use period (which is much more representative of actual field conditions), gives a net fading loss of 10% for a two week and 18% for a four week badge period. The higher fading loss for a four week badge can be compensated for by proper calibration techniques, and in addition the four week badge would theoretically have a fast neutron threshold dose detection capability 45% lower than the two week badge (on a dose per unit time basis). This becomes quite significant when it is considered that the dose detection threshold for the two week badge is about 110 mrems (95% confidence limits). It should be emphasized that these are theoretical values derived mathematically from a track fading study conducted in 1957. A study was planned and is currently under way which hopefully will experimentally verify these calculations.

An evaluation was made of the film ring dosimeter results from IPD's Operating and Radiation Monitoring personnel. The results showed that the continued routine issuance of film ring dosimeters to these two groups is not warranted, based on the criterion for ring use as stated in RPS 5.1. A report summarizing the results of the study and announcing the termination of the program was issued. There will be temporary rings available for those who need them in all of the 105 Buildings.

A special study was initiated to determine if the present radiometallurgy Building stack sample is representative of the stack emission. The sampling equipment consists of five perforated pipes which traverse the exhaust plenum at the stack base connected to individual filter holders. Normally a sample is obtained from one of these probes. At the present time, samples are collected from all five of the probes. Concurrently, samples are collected from the cell exhaust duct. From the comparison of these samples, it should be possible to determine if any one of the sample probes gives a representative sample.

Work was completed on an analog computer program of a mathematical model for calculations of the distribution of radionuclides in man. The model is intended to handle parent-daughter pairs and curves are now available for Te^{131}, I^{131} and Te^{132}, I^{132}.

The design of the mechanical system for the new machanized densitometer was given to the Drafting group for the preparation of final Hanford drawings. The design includes the assembly, part layout, and detailing of all components. Flexibility in the film handling components was obtained by providing alignment adjustments throughout the mechanical system. The capability to adjust the film position will reduce fabrication costs by eliminating many close

tolerances. Throughout the design, a special effort was made to provide for normal adjustments to compensate for any part wear that may take place and to provide ease in mechanical maintenance.

The parity punch provided for coding the identification tape on the personnel film badge dosimeters and providing a method of auditing the performance of the mechanized densitometer was placed in routine operation by the External Dosimetry Operation.

The periodic dose rate standardization measurements for all of the K-fluorescent X-ray radiators, including a new 43 Kev radiator, was completed with the free air ion chamber. All calculated doses were in good agreement with previous results.

A factory representative of the Field Emission Corporation was at Hanford for three days to set up and test the performance of the new flash X-ray unit. All three X-ray tubes, the transmission lines and the remote X-ray head were thoroughly checked and approved as satisfactory. A CPD criticality alarm was tested with this equipment at an X-ray energy of about 600 kilovolts. Although calibration of the actual dose rate from this equipment was not completed the alarm responded satisfactorily for each test conducted. These tests were conducted with radiation pulse durations of 0.12 microseconds. This pulse duration is significantly shorter than that anticipated for any possible criticality event at Hanford. To further demonstrate that the alarm was triggered by the effect of the radiation and not electromagnetic fields generated by the X-ray equipment, several tests were conducted with a dummy load in the equipment in place of the X-ray tube. In each case without the radiation, the criticality alarm failed to alarm indicating that the criticality alarm was indeed responding to the short radiation burst and not some associated effect.

The He^3 neutron detector was successfully used with the electronic system in the coincident mode. The manufacturer recommends a helium pressure in the detector of five atmospheres for best operation and economy. Linearity and sensitivity measurements for the He^3 neutron detector were conducted at pressures of about five and eight atmospheres. The detection sensitivity increased with increasing He^3 pressures. In order to calculate the incident neutron spectrum, the raw data from the detector requires correction for the variations in the He^3 neutron-proton cross section. No He^3 neutron-proton cross-section data is available in the literature above 4.2 Mev. Measurements were conducted to extend this cross-section data to 5 Mev. The range to 5 Mev should be sufficient for most field measurements except for high energy neutron sources such as plutonium-beryllium. Preliminary measurements indicate that the neutron-proton cross-section for He^3 decreases rapidly for neutron energies above 5 Mev.

The greatest problem encountered with the He^3 neutron detector is its sensitivity to gamma radiation. When making measurements in a mixed gamma neutron field, gamma radiation interferes with the neutron spectrum analysis for neutron energies below about 1 Mev by producing superfluous counts. This interference can be partially eliminated by conducting the measurements with and without He^3

in the detector and subtracting the resulting spectrum from each other. Shielding the detector from the gamma radiation also helps if the gamma energy is not too high. A better system for providing automatic gamma compensation while making neutron spectrum measurements with the He^3 detector is being developed and is the subject of an invention report.

The Goetz aerosol spectrometer was operated in the Plutonium Fabrication Pilot Plant in an attempt to obtain calibration data using a plutonium aerosol. Some difficulty was encountered indicating that plutonium particles apparently were shifting their location on the collection filter during the processing. This was reduced to a minimum by careful handling of the filter papers.

Research Studies

Effect of Reactor Effluent on Quality of Columbia River Water (02)

A study of the effects of reactor effluent on river water quality, with emphasis on temperature, continued. Dye studies of travel time and dispersion were begun. Temperature traverses and velocity measurements were interrupted by equipment failure. Testing of meteorological instruments received to date was completed.

Instantaneous dye released in midstream were followed from Vernita Ferry to Hanford, and from Hanford to the Richland Marina. Travel times observed below Hanford indicated that current estimates may be too high. Dispersion rates between Vernita and Hanford were even less uniform than expected, emphasizing the importance of shoreline geometry and main current patterns.

Mechanisms of Environmental Exposure (02)

The I^{131} burdens in the thyroids of two men positioned downwind from the point of an experimental release of I^{131} were followed with the whole body counter. Preliminary analysis indicated an effective half-life of about 5.5 days for both persons. The peak uptake was about 15% of the total inhaled I^{131} as estimated from air sample results and pre-test breathing calibration. Special thyroid counting for this test was terminated on August 12, because the thyroid burdens had declined to below pre-test levels.

A part of the study conducted last month which involved the experimental release of I^{131} on a field grid included the testing of carbon tetrachloride and carbon impregnated filter paper as iodine collectors. Laboratory results indicate that both materials collect I^{131} in proportion to the amount available in the cloud, but that the paper was nearly twice as efficient as the carbon tetrachloride. The orientation of the paper did not affect its efficiency.

C. RELATIONS

Two suggestions were submitted by Radiation Protection Operation personnel during the month. One $10.00 award was made, none were rejected and six are pending evaluation.

Safety meetings were held throughout the Section during the month. Revisions and additions to the Manual on Health and Safety were among those topics discussed.

One Technical graduate assigned to Coolants Development and one Captain, USAF, assigned to the Experimental Animal Farm were given a 60-minute orientation talk. One new Biology employee received the same type of information. A two-hour radiation orientation session was conducted for a new laundry supervisor. Radiation orientation lectures were given in the 300 Area to 10 new hires and transferred personnel.

A quiz on Disaster Monitoring techniques has been prepared for radiation monitors to stimulate interest and serve as a refresher on the training material presented over the past two years.

The training program for RMO Radiation Monitors has been organized, and a schedule established beginning September 10. Meetings will be held in the mornings in the Outer Conference Room of 325 Building. Each session will be given five times consecutively, before moving on to the next subject. The first session of each will be given on a Tuesday, and will be presented for Radiation Monitoring Operation Specialists and Supervisors. The following four sessions of each will be presented for Radiation Monitors on the next four Fridays.

D. SIGNIFICANT REPORTS

HW-76525-7 - "Radiological Status of the Hanford Environs for August, 1963", by R. F. Foster.

HW-78825 - - "Monthly Report - August, Radiation Monitoring Operation", by A. J. Stevens.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

External Exposure Above Permissible Limits	August	1963
Whole Body Penetrating	0	1
Whole Body Skin	0	0
Extremity	0	0
Hanford Pocket Dosimeters		
Dosimeters Processed	6,148	51,363
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	9,744	76,208
Results - 100-300 mrads	199	1,353
- 300-500 mrads	8	121
- Over 500 mrads	5	29
Lost Results	19	175
Average Dose Per Film Packet - mrad (ow)	9.0	7.1
- mr (s)	19.5	30.2
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	1,708	13,454
Results - 50-100 mrem	1	14
Results - 100-300 mrem	0	2
Results - Over 300 mrem	0	0
Lost Results	8	90
Fast Neutron		
Film Processed	507	3,665
Results - 50-100 mrem	9	241
Results - 100-300 mrem	19	407
Results - Over 300 mrem	0	4
Lost Results	7	69
Hand Checks		
Checks Taken - Alpha	43,662	304,135
- Beta-Gamma	54,893	466,215
Skin Contamination		
Plutonium	34	202
Fission Products	40	315
Uranium	1	5
Tritium	0	0
Thorium	1	1

Whole Body Counter

Subject	Number of Examinations 747-A WBC	1963	Mobile WBC	1963
GE Employees				
Regular	45	488	30	30
Incident Cases	11	121		
Terminations	17	88		
New Hires	46	403		
Special Studies	29	421		
Non-Employees				
Children	4	16		
Visitors	12	41		
Environmental Studies	2	15		
	166	1593		

Bioassay

	Current Reporting Limit	Results Above Reporting Limit August	1963	Samples Assayed August	1963
Plutonium	2.2×10^{-8} μc/sample	57	725	531	4,783
Fission Products	3.1×10^{-5} μc/sample	0	68	470	4,310
Strontium	3.1×10^{-5} μc/sample	0	45	0	45
Tritium	5.0 μc/l	84	1,044	106	1,745
Uranium	0.14 μgm/l	0	0	83	1,211
Special Studies		0	0	3	314

Calibrations

	Number of Units Calibrated August	1963
Portable Instruments		
CP Meter	1,127	8,537
Juno	269	2,117
GM	608	4,549
Other	183	1,487
Audits	113	854
	2,300	17,544
Personnel Meters		
Beta Gamma Film	648	6,548
Rings	75	885
Other Film	264	2,462
	987	9,895

	Number of Units Calibrated	
	August	1963
Miscellaneous Special Services	2,506	14,003
Total Number of Calibrations	5,793	41,442

Carl M. Unruh
for the
Manager
RADIATION PROTECTION

CM Unruh:jbp

FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

Authority to obligate 04 Program capital equipment funds was increased by RLOO-AEC to $278,600 - a $68,600 increase in this FY 1964 control. The additional funds are specifically earmarked for a Dynapak machine for Reactor and Fuels Laboratory.

A Research and Development proposal to study economic-biologic interactions was submitted to RLOO-AEC. The study would clarify and refine selected concepts, test assumptions, and suggest likely alternatives. One man-year of effort at $30,000 is proposed.

Special accounting codes were established for the activities described below:

.3V Welding assistance to the Boeing Company - $6,100 was authorized to cover the costs of approximately 80 hours welding time, salaries and indirect costs of associated personnel and auxiliary services such as cutting, polishing and microscopy.

.8F Consultation with the Advance Electronic Center, Ithaca, New York - consultation by R. T. O'Brien pertaining to the electronic biological detection system at the Center.

.8G Consultation with Walter Reed Army Medical Center - consultation by C. E. Newton, Jr. on health physics, clinical X-ray and other related programs.

General Accounting

Following is the status of letters requesting AEC concurrence in proposed actions:

AT-304	Participation in Standardizing Activities - ASTM Committee C-5 - Manufactured Carbon and Graphite Products - E. M. Woodruff	Approved 8-9-63
AT-310	Washington State Air Pollution Control Board - J. J. Fuquay	In hands of AEC
AT-311	Teaching Assistance to Washington State University - Course CE-547	In process

The following revised OPGs were issued:

OPG No.	Title
4.1	Hanford Engineering Standards
7.12	Privileged Non-Defense Information

Hanford Laboratories' material investment at August 1, 1963 was:

	(In thousands)
SS Material	$22 792
Reactor and Other Special Materials	1 295
Spare Parts	328-1)
	$24 415

(1- Includes a reserve of $81 thousand.

The cumulative value of nuclear material consumed in research by Hanford Laboratories during FY 1964 is $89,659 comprised as follows:

02 Program	$ 1 487
04 Program	88 172
	$89 659

The results of the physical inventory of movable catalogued equipment assigned to the Chemical Laboratory follow:

	Quantity	Value
Balance Prior to Inventory	2 797	$2 738 456
Reconciled Adjustment	7	6 237
Adjusted Book Balance	2 804	2 744 693
Physical Inventory	2 797	2 742 978
Missing Equipment	7	$ 1 715
% of Missing Equipment to Adjusted Book Balance	0.25%	0.06%

Of 126 items originally not located all except seven were found through able effort by custodial and financial personnel. The number of initially unlocated items indicates a need for improved control.

A summary report of the FY 1963 physical inventory of movable catalogued equipment in the custody of Hanford Laboratories was prepared and distributed. Comparison of this report representing one complete cycle of inventories within Hanford Laboratories disclosed 42 missing items valued at $14,917 compared to 16 missing items valued at $5,963 in the FY 1961-62 inventory. In addition FY 1963 inventory results show 82 items valued at $36,772 added to record compared to 75 items valued at $32,842 in the FY 1961-62 inventory. The percentage (0.26%) of missing equipment, though not considered excessive, indicates a need for improved control. Therefore, the FY 1964 inventory will mark a return to a certification inventory by custodial personnel to be followed by a witnessed inventory the ensuing year.

The annual physical verification of zirconium holdings was conducted on August 28, 1963. The inventory count by C&AO and Hanford Laboratories' personnel is complete, and the reconciliation is in progress.

The heavy water inventory at the end of August 1963 showed a loss of 763 pounds valued at $10,478 for the PRTR and a gain of 4 pounds valued at $56 for the PRCF. Heavy water scrap generated during the month amounted to 2,375 pounds resulting in a $1,682 charge to operating costs. Heavy water accumulated at August 31 for return to SROO amounted to 13,221 pounds valued at $165,811.

A listing of reactor and other special materials excess to the needs of Hanford Laboratories and valued at $66,444 was submitted to RLOO-AEC. Included was $54,319 of contaminated platinum scrap. The New York Operations Office-AEC hopes to select a contractor for the recovery of contaminated platinum scrap such as this during the Fall. Inquiries received promise earlier disposal of the clean materials.

Laboratory Storage Pool activity is summarized below:

	Current Month		FY to Date	
	Quantity	Value	Quantity	Value
Beginning Balance	1 683	$816 815	1 480	$811 520
Items Received	150	49 535	436	139 556
Items Reclaimed by Custodians	(15)	(16 105)	(32)	(29 092)
Equipment Transfers	(67)	(24 586)	(85)	(33 133)
Items Disposed of by PDR	(7)	(190)	(41)	(1 058)
Items Disposed of by Excess	(70)	(3 159)	(84)	(65 483)
	1 674	$822 310	1 674	$822 310 -1)

(1- Includes 136 items valued at $113,416 on loan at 8-31-63.

During the month, 98 items valued at $39,293 were loaned and/or transferred in lieu of purchases. A total of 121 items valued at $113,416 has been redirected to useful purposes this fiscal year. Operating cost for the month of July was $1,198.

Total investment of equipment and material in custody of the Laboratory Storage Pool at August 31, 1963 was $1.6 million including Reactor and Other Special Materials valued at $301,252, SS Materials valued at $163,800 and other materials valued at $353,909.

The following action was taken on projects during the month:

Physical Completion Notices Issued

*CAH-822 Pressurized Gas-Cooled Loop Facility
CAH-867 Fuel Element Rupture Test Facility
CAH-963 Geological and Hydrological Wells - FY 1962

*AEM Services only.

Construction Completion and Cost Closing Statements Issued

CGH-951 A-C Column Facility, 321 Building
CGH-857 Physical and Mechanical Properties Testing Cell, 327 Building
CGH-858 High Level Utility Cell, 327 Building

The following contracts were processed during the month:

CA-407 George W. Watt
DDR-166 Crane Company - Deming Division
SA-289 Crane Hoist Engineering Corp.
MRO- 62 Ling-Temco-Vought, Inc.
MRO- 63 Textron Electronics, Inc.
MRO- 64 X-ray Products Corporation

Personnel Accounting

D. W. McLenegan, Consulting Engineer, Programming Operation, died on August 18, 1963.

W. W. Roberts retired effective September 1, 1963.

A Patent Award (HWIR 874) was made to R. F. Warnick on a method of displacing uranium and poison slug columns in a process tube during operation.

TECHNICAL ADMINISTRATION

Employee Relations

Nineteen nonexempt employment requisitions were filled; 19 remain to be filled.

Suggestion plan activity included 44 submissions, 25 adoptions and 20 rejections.

Information and Presentations

A technimated flow diagram depicting a CPD separations process was completed and installed in the Visitors Center through the cooperative efforts of CPD, R&OHO, HU&PO and Hanford Laboratories' personnel.

A news release on the conversion of PRCF to heavy and light water capability was prepared and distributed. Nuclear publications, science editors of selected newspapers and news magazines and syndicates were covered.

Visitors Center activity:

August attendance	3 293
Average attendance per day open	106
Cumulative attendance since 6-13-62	54 383
Conducted groups	3 (totaling 127 people)

Plant Tour activity:

	Number	Total People
General public relations tours	1	40
Special tours	4	116

Documented information flow during the month was comprised of 1,110 titles (12,665 copies) received at Hanford and 70 titles (5,416 copies) sent off-site.

Professional Placement

Advanced Degree - Seven Ph.D. applicants visited HAPO for employment interviews. Four offers were extended; five acceptances and four rejections were received. Four offers are currently open.

BS/MS (Direct Placement) - Two offers were extended. There were no acceptances and one rejection was received. Two offers are currently open.

BS/MS (Program) - There was no employment activity in this category this month.

Technical Graduate Program - Three Technical Graduates were placed on permanent assignment. Eight new members were added to the roll and one member terminated. Current program members total 79.

FACILITIES ENGINEERING

Projects

At month end Facilities Engineering Operation was responsible for 10 active projects having total authorized funds in the amount of $6,735,000. The total estimated cost of these projects is $10,640,000. Expenditures on them through June 30, 1963 were $836,000. (Costs for July are traditionally not reported.)

The following summarizes project activity in August:

Number of authorized projects at month end ----------	10
Number of new projects authorized ----------	0
Number of projects completed ----------	0
New projects submitted to the AEC ----------	0
Projects awaiting AEC approval ----------	1
CAH-106 - PRTR Storage Basin and Experimental Facilities Modifications	
Project proposals being prepared ----------	7
PRTR Decontamination and D_2O Cleanup	
Heat Transfer Apparatus for Model Studies	
300 Area Crib Waste Transport System	
Addition to the Critical Mass Laboratory	
141-M Building Addition	
Addition to the Fire Alarm System - 100-F Area	
Fire Sprinkler Systems - 300 Area Laboratories	

The current status of projects authorized or awaiting approval is:

CAH-916 - Fuels Recycle Pilot Plant - Construction is 8 percent complete and on schedule. A major design change was negotiated to provide a partial basement under the chemical research wing. Installation of underground utilities is progressing. Substructure foundations are being formed and poured.

CAH-922 - Burst Test Facility for Irradiated Zirconium Tubes - Construction is 55 percent complete compared to a scheduled 58 percent. The lag was caused primarily by late delivery and poor prefabrication of structural steel. The steel has now been erected and interior paneling is being installed.

CAH-958 - Plutonium Fuels Testing and Evaluation Laboratories, 308 Building - Construction is 63 percent complete compared to a scheduled 53 percent. Plastering is complete. Fabrication of the metallograph hood was completed, and the hood has been delivered to the construction site. Painting and electrical work are in progress.

CAH-962 - Low Level Radiochemistry Laboratory - Detailed design by the fixed price A-E is 90 percent complete and on schedule. The architect-engineer's work was reviewed in Seattle by Commission and Company representatives on August 22, 1963.

CAH-977 - Facilities for Radioactive Inhalation Studies - Detailed design by the fixed price A-E is 97 percent complete compared to a scheduled 99 percent. Company and Commission representatives went to Yakima to review the design with the Architect-Engineer on August 21, 1963.

CAH-982 - Addition to Radionuclide Facilities - 100-F Area - Carson and Kesterson, Architect-Engineers, are performing design on this project. Title I design is almost complete. A Spokane firm is performing the electrical and mechanical design.

CGH-992 - Additional Fuel Loading Equipment, 308 Building - Construction is approximately 98 percent complete and on schedule. Installation of floor and ceiling tile and a fire detection system remain to be completed.

CAH-995 - 309 Building Air Conditioning Modifications - Construction is 95 percent complete. The completion date was extended to September 30, 1963. The water chiller, damaged in transit, was repaired and is installed. By mid-August all materials and equipment were on hand.

CGH-999 - Plutonium Recycle Critical Facility Conversion for Light Water - Design is complete. Fabrication of components in the J. A. Jones Company shops began on July 29, 1963, and is progressing satisfactorily. Construction is about 30 percent complete. Work at the job site has consisted principally of installation of the new moderator pump and controls.

CAH-100 - High Temperature Lattice Test Reactor - All design funds authorized Vitro Engineers ($26,000) have been expended. The Commission requested all design activity stopped effective August 23, 1963. A request to Washington-AEC for additional advance design funds was disapproved. It currently appears this project will remain inactive until authorizing legislation for FY 1964 is passed by Congress.

CAH-106 - Plutonium Recycle Test Reactor Storage Basin and Experimental Facilities Modifications - The Commission deferred action on the preliminary project proposal until the Company submits a completed design criteria document and indicates an intent to assign all design to Vitro Engineering. The Company had planned to award design-and-fabricate type contracts directly to equipment vendors for the underwater fuel element handling equipment. After reviewing the matter with the Commission, a letter was transmitted indicating the Company's intent to revise the project proposal to request design of the wet storage basin, fuel storage racks, and building addition. The request for underwater handling equipment design will be deleted and submitted later as a supplemental project proposal accompanied by a separate design criteria document.

Pressure Systems

Engineering review and surveillance of pressure systems continued. Leak detection testing was conducted on 300 Area piping systems and maintenance is being performed on the systems as required.

Elevators at the Meteorology tower and in the 308, 325, 326, 327 and 3760 Buildings were inspected with the third party inspector. Minor corrective maintenance was performed. All safety devices tested satisfactorily.

The in-reactor test section of the C-1 loop was installed and welding radiographed. Reactor down-time limitations prevented final pressure tests. Loop is operating on single-pass cooling.

Engineering was accomplished for on-site fabrication of a Hastelloy "C" chlorine condensing and storage tank at 250 psig for salt cycle process work and heat transfer cells for loops TF-2 and TF-3 at 1706 KE at 600° F and 2000 psig.

Engineering Services

Engineering work provided in support of design and construction on active projects and project proposal and design criteria for new projects included: (1) field liaison on CAH-916, FRPP; (2) review of A-E design on CAH-962, Low Level Radiochemistry Building; (3) study of digital computer for control and

problem solving on CAH-100, High Temperature Lattice Testing Reactor; and (4) project engineering for CGH-999, Plutonium Recycle Critical Facility Conversion for Light Water and CGH-992, Additional Fuel Loading Equipment, 308 Building.

Other engineering work and consulting services were provided to customers as requested. Major work items included: (1) study for alleviation of beta-gamma radiation in 108-F building emanating from the reactor effluent system; (2) study of 144-F building ventilation system malfunction; (3) preliminary work on engineering study of a fast reactor critical facility; (4) mechanical engineering for test thimbles for N Reactor start-up tests; (5) operation and engineering of modifications and improvements to NPR charging machine; (6) electrical engineering for 308 building vacuum welding hood; (7) development of equipment for automatically metering fuel compositions to obtain required fuel mix; (8) engineering assistance on development of a rotating crystal spectrometer; (9) development of a tilting rack for cask decontamination in 325 decontamination tank; (10) preliminary scope design for a containment test facility; (11) review and modification of temperature control system for 232-Z incinerator; (12) reviewed scope design and prepared specification for ventilation control valves for briquetting and degreasing hoods - 234-5 building; (13) preparation of purchase drawings for liquid chlorine storage vessel; and (14) preparation of purchase specification for EDEL II loop control systems.

Plant Engineering

Plant engineering work included: (1) development of color standards for use by Architect-Engineers and maintenance forces; (2) preliminary scope design of a PRTR viewing center; (3) preliminary planning for engineering study of a consolidated laboratory services shop; (4) engineering of a concept of storing plutonium alloy fuel rods; (5) evaluation of sampling pumps purchased for the retention waste system; (6) assistance with specifications for switches and transformers for 314 gas loop power supplies; (7) purchase drawing for replacement hydrogen gas cylinder leads; (8) engineering for installation of 325 building generator; (9) scope layout of 325 building second floor modifications; (10) investigation of methods of sealing cover block joints at 325 building; (11) electrical layout for additional laboratory service; (12) design criteria for 326 basement modifications; (13) preliminary scope layout of a 329 addition; (14) analysis of 329 supply fan motor size; (15) engineering of power supply circuitry for electrolysis hoods to "fail-safe" operation; (16) design criteria of service requirements to 747 building without the powerhouse; (17) review and modification of ventilation for 747 building; (18) engineering of modifications to 1706-KER balance room; and (19) vendor information drawings for purchase of storage racks for 3718 building.

Facilities Operation

Landlord costs for the month of July were $124,752. Improvement maintenance activity indicated by the work pending will exceed the budget. More funds will be requested at the mid-year budget review.

The following table summarizes waste disposal operations:

Item	June	July
Concrete waste barrels disposed to 300-N burial ground	8	12
Loadluggers of dry waste disposed to 300-N burial ground from the 325 Building	3	4
Loadluggers of dry waste disposed to 300-N burial ground from other 300 Area sites	19	21
Crib waste volume (gallons)	270 000	415 000

The new trench and vertical burial pipes are almost completed at the "Wye" burial ground. The "Wye" burial ground was closed during construction, and all burials were made at 300-N during August.

No basins of retention waste exceeded the activity level of 5×10^{-6} μc-β/ml or 5.0×10^{-5} μc-a/ml during August.

The crib waste influent rate continues at about 600 gph. This high rate resulted in such high inventories on the weekends that overtime transport to 200 Areas was required.

The crib at Redox has been caving in. Consequently, the Hanford Laboratories unloading station is being moved to T-plant tank farm and should be relocated there within 30 days.

Sample disposal and decontamination work in the 325 building decontamination facility is being largely performed remotely. System improvements and equipment modifications are continuing.

A new manifold was installed for the control air lines for the heating and ventilating system in 326 building. This corrected certain weaknesses in the original piping and makes it possible to sectionalize the system for repairs.

The new demineralizers at 327 building have been test operated and are ready for service.

The 329 ventilation system was tested this month during a scheduled outage. It was determined that adequate flows could be maintained under shutdown conditions with only one exhaust and no supply fans operating. Consequently, it was decided to replace the faltering two-speed motors on the supply units with single-speed higher capacity units.

Drafting

The equivalent of 184 drawings were produced during the month for an average of 19 man-hours per drawing. Major jobs in progress are: (1) PRTR as-builts; (2) inhalation studies hood as-builts; (3) salt cycle experimental laboratory for "C" cell, 325 building; (4) preliminary conceptual drawings for fast super-critical reactor; (5) high temperature gas loop; (6) HTLTR mock-up core and safety and control rods; (7) refraction compounds research glove box; (8) glove boxes for 231-Z powder processing line; (9) drawings associated with engineering work reported above; and (10) miscellaneous drawings for Technical Shops purchases.

Drafting work estimated at 790 man-hours was assigned to Vitro Engineering Corporation during the month. Work performed by Vitro during the month amounted to 250 man-hours.

Construction Supervision

Activity during the month on construction work (J. A. Jones Company) being performed for Hanford Laboratories components is given below:

	Unexpended Balance	
	Regular	Waste Calcination
Orders outstanding beginning of month	$185 502	$72 015
Issued during the month (incl. suppl. and adj.)	165 912	
J. A. Jones expenditures during month (includes C.O. costs)	157 164	31 878
Balance at month's end	194 250	40 137
Orders closed during month	101 587	

In addition, work on three maintenance work orders issued to plant forces and having a face value of $1,732 was supervised.

A contract is being executed with Crane Hoist Engineering Company for safety and maintenance inspection of Laboratories' cranes and hoists. Work is expected to start in September.

Major nonproject jobs in process, by building, are: (1) pre-cooling system, 108-F; (2) feeder stalls, 141-C; (3) back-up compressor 144-F; (4) modifications to source handling cask, 108-F; (5) modifications to breathing air system, 144-F; (6) tensile test glove box installation, 231-Z; (7) installation of steel framing and drain lines, 241-CR; (8) installation of service to arc melt furnace, 306; (9) replacement of steam and condensate lines, 308; (10) installation of flooring, 308; (11) construction of shielding walls, 309; (12) gas loop modification, 309; (13) rupture loop instrumentation modification, 309; (14) "C" cell door modification, 325; (15) construction of heat and vent system, 325 basement; (16) laboratory modifications, rooms 400 and 406, 325; (17) roof shelter, 325; (18) shower room repair, 325; (19) complete test cell, 327; (20) exterior stair construction, 327; (21) renovate lunchroom, 328; (22) modify laboratory 11-B, 329; (23) renovate exterior, 3702; (24) modify electrical service, 3717-B; (25) modify 3718; (26) construct burial ground at 300 Wye; and (27) fabricate waste calcination equipment.

Ten requisitions totaling $1,500 were issued during the month. Total value of equipment being processed is $96,000.

for W W Parsley
Manager
Finance and Administration

W Sale:whm

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output for August was 1545 MWD, for an experimental time efficiency of 83.3% and a plant efficiency of 71.2%. August output is the highest monthly output since startup of the PRTR. There were nine operating periods during the month, of which seven were terminated by scrams, one was terminated manually and one continued through month-end. A summary of the fuel irradiation program as of September 2, 1963, follows:

	Al-Pu		UO_2		PuO_2-UO_2		Other		Program Totals	
	No.	MWD	No.	MWD	No.	MWD	No.	MWD	No.	MWD
In Core	17	1445.4	7	1082.1	61	5185.6			85	7713.1
Maximum		132.0		213.0		128.7				
Average		85.0		154.6		85.0				
In Basin	26	1707.9	59	4556.2	12	232.0	1	7.3	98	6503.4
Chem. Process.	32	2309.3	1	78.0	—	—	—	—	33	2387.3
Program Totals	75	5462.6	67	5716.3	73	5417.6	1	7.3	216	16603.8

Note: (MWD/Element) X 20 = MWD/TU for UO_2 and PuO_2-UO_2.

D_2O and indicated helium losses for August were 763 pounds and 101,582 scf., respectively.

Equipment Experience

A total of 73 reactor outage hours were charged to repair work. Main items were:

Instrument repairs	25 hours
Valve repairs	18 hours
Modifications to process tube outlet temperature detectors	10 hours
Replacement piping on primary heat exchanger	9 hours
Shim rod repairs (3 rods repaired, one installed)	8 hours

Preventive maintenance required 243 manhours or 4.5% of the total maintenance effort.

Improvement Work Status (significant items)

Work Completed

Deionized Water System
Provide Separate Emergency Backup Water for FEEF

Work Partially Completed

Inline Gas Sampling
Process Tube Level Indicator
Independent Criticality Alarm System of the PRTR Control Room
Inlet Gas Seal Replacement
Backup Emergency Power to Primary Pumps
Helium Compressor Unloading Orifice Modification
125-Volt Battery Disconnect Contactor
Control Wiring for 13.8 KV Motor Operated Switches
Filtered Water Pump #3
Shim Rod Shroud to Top Cap Modification
Improved RTD Connector Sealant and Bracing
Gamma Scanner Installation
#2 Exide 125 V Battery Charger
Flash Tank Modification
Instrument Power Transfer System
Stiffener on FEEF Air Baffle
Control Room and Instrument Calibration Shop Air Conditioning (CAH-995)
H-80 Relocation and Trim Change

Design Work Completed

Flow Monitor Tubing Snubber Installation
Autoclave Installation for ZR-2 Fretting Corrosion Studies
Helium Compressor Piping Modification for Snubber Installation
Compressed Air Supply Revision
Pressurizer to Stack Valve Relocation
Actuator with Hydraulic Snubber for Bottom Blow-Down Line
DT-1 Storage Tank Sight Gauge
Installation of New Alarm Annunciator
Galvanometer Shunt Replacement
Light Water Injection System Modifications
S-60 Valve Trim Change

Design Work Partially Completed

Additional Fuel Storage and Examination Facility (CAH-106)
Vibration Snubbers for Earthquake Protection
Decontamination Building and D_2O Cleanup Facility

Design Work Partially Completed (Continued)

Flux Wire Scanning System
Supplemental Emergency Water Addition
Permanent Installation of Closed Circuit TV
Indication of DC Solenoid Failure
Rupture Monitoring System Modifications

Process Engineering and Reactor Physics

PRTR Test #57 (Increased Phosphate Concentration in HX-1) was completed. The results of the test plus laboratory studies showed that the desired elimination of free caustic in the secondary side of the steam generator could not be obtained using the maximum allowable phosphate concentration when injecting disodium phosphate.

PRTR Test #67 (Deaerator Efficiency) was completed.

Final report of PRTR Test #69 was issued after performance. In this test, the linearity of the uncompensated ion chambers was checked and found satisfactory.

Procedures

Operating Procedures issued	1
Revised Operating Procedures issued	2
Revised Operating Standards issued	5
Temporary Deviations to Operating Standards issued	2
Revised Process Specifications accepted for use	1
Equipment Standards issued	4
Equipment Manuals issued	3

Drawing As-Built Status:	August	Total
Approved for As-Built	25	1 014
In Drafting		23
In Approval		14
Deleted or voided		81
		1 132
Scheduled for review		338
		1 470

Personnel Training:	Manhours
Qualification subjects	392
Specifications, Standards, Procedures	56
Emergency Procedures	20
Maintenance Procedures	37
	505

Status of Qualified Personnel at Month-End:

Qualified Reactor Engineers	9
Qualified Lead Technicians	6
Qualified Technicians	17
Provisionally Qualified Technicians	2

A new set of qualification questions in reactor physics was prepared for Reactor Engineers.

Experimental Reactor Services

The status of the various test elements as of the shutdown on September 2, 1963, is shown below. Those test elements which had reached their assigned goal exposure or had been permanently discharged for other reasons prior to August 1, 1963, have been deleted from this table.

Test No.	Channel Location	F.E. Number	Description	Date Initial Charge	Date Discharged	Approx. Accumulated MWD
14	1956	5097	Moxtyl-Swaged	4/2/62	--	80.3 repad
14	1352	5098	Moxtyl-Vipac	5/8/62	--	128.7 repad
14	1758	5099	Moxtyl-Vipac	5/8/62	--	100.5 repad
37	1649	1097	UO_2 Physics	5/12/62	--	122.1
37	1552	1098	UO_2 Physics	5/12/62	--	117.6
37	1548	1099	UO_2 Physics	5/12/62	--	120.3
37	1651	1100	UO_2 Physics	5/12/62	--	108.9
48	1253	5150	Moxtyl ($\frac{1}{2}$"x$\frac{1}{2}$" pads)	8/1/62	--	78.3
47	1647	5121	Unautoclaved LX PuAl	6/13/63	--	45.0
47	1653	5194	Unautoclaved LX PuAl	7/6/63	--	35.2
47	1453	5193	Unautoclaved LX PuAl	7/6/63	--	34.6
54	1948	5116	Moxtyl (clip-on pads)	5/8/62	--	102.5(87.5 w/clip)
54	1554	5118	Moxtyl (clip-on pads)	5/8/62	--	122.9(107.9 w/clip)
61	1443	5185	Moxtyl-Physics	5/28/63	--	49.7
61	1756	5186	Moxtyl-Physics	5/28/63	--	53.0
61	1847	5187	Moxtyl-Physics	5/28/63	--	59.9
61	1455	5192	Moxtyl-Physics	6/13/63	--	50.7

Two fuel elements were examined; one was rejected.

Visual examination of 14 process tubes indicated that although more shallow marks ($\sim$.001) have appeared on the tubes, since the last examination, there was no change in the deeper ($>$ .010") fret marks.

Plutonium Recycle Critical Facility

Operation

The entire month of August was devoted to the reactivity measurement series for Pu fuel samples. Testing of cold samples was essentially completed at mid-month, and the first irradiated sample test was complete at month-end. Final preparations for irradiated fuel handling, related mechanical and critical testing, and the first two complete cycles of the irradiated fuel handling sequence consumed the second half of the month. Lost time from unscheduled events was negligible; beneficial use was realized for each workday of the month.

Improvement Work Status

Design of the bypass switch for pump interlock was completed during the month.

Process Engineering

No off-standard conditions were observed from routine audits. A review of the PRCF Operating Procedures was completed and a report submitted.

Procedures

Revised Operating Procedure issued	1
Process Specification accepted for use	1
Revised Process Specifications accepted for use	2

Fuel Element Rupture Test Facility

Operation

Four hot operating runs were conducted during the month, two of which were terminated for planned maintenance, one was terminated as the result of a scheduled diesel generator load test, and one was terminated because of a plugged control air line to the temperature control valve.

Improvement Work Status

Work Completed

Instrument Air Backup
Flow Monitor and Alarm-Discharge Operation

Work Partially Completed

Test Section Emergency Water Injection
Safety Circuit Bypass Switch

Design Work Completed

Emergency Power for RLP-2

Procedures

One Process Specification was accepted for use.

Processing of Spent Fuels

Approval was obtained from RLOO-AEC to dispose of 39 irradiated UO_2 fuel elements through production channels.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 26,271 hours. This includes 17,008 hours performed in Technical Shops, 5,623 hours assigned to J. A. Jones Company, 3,573 hours assigned to offsite vendors, and 67 hours to other project shops. Total shop backlog is 21,899 hours, of which 90% is required in the current month with the remainder distributed over a three-month period. Overtime worked during the month totaled 517 hours.

Distribution of time was as follows:

	Manhours	% of Total
N Reactor Department	3 763	14.33
Irradiation Processing Department	4 352	16.57
Chemical Processing Department	343	1.30
Hanford Laboratories	17 813	67.80
Hanford Utilities and Purchasing	0	-

LABORATORY MAINTENANCE OPERATION

Total productive time was 21,050 hours of 24,400 hours potentially available. Of the total productive time, 86% was expended in support of Hanford Laboratories components with the remaining 14% directed toward providing service for other HAPO organizations.

Manpower utilization for August was as follows:

A.	Shop Work		2 980 hours
B.	Maintenance		8 470 hours
C	1. Preventive Maintenance	1 850 hours	
	2. Emergency or Unscheduled Maint.	2 000 hours	
	3. Normal Scheduled Maintenance	3 850 hours	
	4. Overtime	770 hours	
C.	R&D Assistance		9 600 hours

The standards calibration and certification workload continues at a high level. Eighty-nine instruments were certified during this report period.

WD Richmond

Manager
Test Reactor and Auxiliaries

WD Richmond:bk

INVENTIONS OR DISCOVERIES

All persons engaged in work that might reasonably be expected to result in inventions or discoveries advise that, to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during the period covered by this report except as listed below. Such persons further advise that, for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

INVENTOR	TITLE OF INVENTION OR DISCOVERY
G. A. Last and R. S. Kemper, Jr.	Formation of Submicron Uranium Carbide Particles in Metallic Uranium, HW-78625, August 12, 1963. HWIR-1643
G. S. Allison	The Field of High Surface Area to Fuel Volume Ratio Nuclear Fuel Elements of a Wide Range of Materials and Operating Temperatures, HWIR-1645, August 21, 1963
R. J. Hoch	Particle Velocity Selectors
R. J. Hoch	Neutron Spectroscopy or Mechanical Velocity Selectors
L. A. Bray and F. P. Roberts	Separation of Rare Earths from Cerium (HW-78690)
G. W. R. Endres and T. L. Ryan	A Gamma Radiation Discriminating He^3 Neutron Spectrometer
W. S. Kelly	"Zipper Duct" - G.E. Case 1077. Invention reported March 3, 1959. Patent Application signed July 30, 1963

Manager, Hanford Laboratories

END

DATE FILMED

3/29/93