HW-79726

HANFORD LABORATORIES MONTHLY ACTIVITIES REPORT

NOVEMBER, 1963

DECEMBER 16, 1963

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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Compiled by
Section Managers

December 16, 1963

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PRELIMINARY REPORT

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TABLE OF CONTENTS

Table I - Hanford Laboratories Force Report

Date: November 30, 1963

	At Beginning of Month		At Close of Month		
	Exempt	Salaried	Exempt	Salaried	Total
Chemical Laboratory	145	131	145	132	277
Reactor & Fuels Laboratory	202	192	205	195	400
Physics & Instruments Laboratory	100	76	100	78	178
Biology Laboratory	43	64	43	64	107
Applied Mathematics Operation	18	5	18	5	23
Radiation Protection Operation	43	94	42	93	135
Finance & Administration Operation	154	116	149	118	267
Programming Operation	17	2	18	2	20
Test Reactor & Auxiliaries Operation	62	314	63	317	380
General	4	4	4	5	9
TOTAL	788	998	787	1009	1796

BUDGET AND COST SUMMARY

November operating costs totaled $2,879,000, an increase of $105,000 over the previous month; fiscal year-to-date costs are $13,568,000 or 42% of the $32,447,000 control budget. Hanford Laboratories' research and development costs for November 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	$ 76	$ 79	$ 396	$ 1 180	34
03	53	51	211	250	84
04	1 318	1 224	5 954	13 726	43
05	123	127	619	1 456	43
06	297	269	1 364	3 604	38
08	12	9	60	100	60
	1 879	1 759	8 604	20 316	42
Sponsored by					
NRD	171	173	806	1 798	45
IPD	49	58	294	490	60
CPD	128	137	591	1 668	35
Total	$2 227	$2 127	$10 295	$24 272	42%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

Radiometallurgical examination of an outer N-fuel component irradiated to 3100 Mwd/ton was started. The component incurred 2.5% swelling; visual examination has revealed no deficiencies in the performance of the component.

Fluted, N single tube N-fuel elements being irradiated in the ETR have achieved an exposure of 1400 Mwd/ton at a maximum fuel temperature of 520 C.

Ninety-four Zircaloy-2 clad uranium rods have been irradiated, examined visually, and diameter measurements completed for the study of clad thinning. With uniform cladding, below a clad temperature of

of 325-350 C, the cladding strain limit is approximately 1.5%. Striations, which reduce clad thickness by 15%, reduce the strain limit to approximately 0.8%.

A device has been completed for measuring the geometry of the end cap region of N-fuel elements.

A Zircaloy-2 clad, Th-U fueled crud probe is successfully operating in a pressurized loop under prototypic N-Reactor conditions.

Two tubes of poison column elements and four tubes of target elements, complete with steel surfaced aluminum supports, were completed well ahead of a December 1 deadline. These elements will be used for physics measurements in N-Reactor startup.

Sixteen outer N-fuel components have been closed by projection welding on a 1200 kva welder at Rocky Mountain Arsenal. This initial test demonstrates that the welder has adequate capacity, but some redesign of tooling will be required.

A lubrication process which permits the extrusion of 2-S aluminum leaving extremely good finish has been developed.

Three tubular Zircaloy-2 clad Th-U fuel elements have been irradiated to 3600 Mwd/ton and have shown a fuel volume increase of only 0.5% for the highest exposure element.

Capsules containing Zircaloy-2 clad uranium rods have been assembled to test the swelling resistance imparted by submicron platelets of uranium carbide within the uranium metal. These capsules are scheduled for reactor charging in December.

Gas phase hydriding at 400 C has been performed on two N-Reactor burst specimens. Both specimens were subjected at room temperature to brittle fracture test; both failed in a brittle manner at pressures one-sixth to one-fourth that required for a nonhydrided specimen.

At 190 C (374 F), the creep rate of the beta phase plutonium was 80 times the creep rate of gamma phase plutonium.

Evidence indicates that self-damage exerts a significant effect upon the physical properties of alpha plutonium. Material which was rolled with very little difficulty in its freshly cast condition has been found to be unworkable after about 3 mo.

Crystallographic anisotropy, which has a marked effect on the tensile properties of unirradiated Zircaloy-2, continues to influence tensile properties after irradiations to 2×10^{21} nvt. The yield strength in both longitudinal and transverse directions continues to increase with irradiation.

Additional Zircaloy samples confirm the 10-fold increase in corrosion associated with irradiation. Samples exposed to the irradiated coolant in the loop, but out of the flux zone, did not experience accelerated corrosion.

The 3000 psi, 300 C stress rupture tests on steam generator tubing containing flaws through greater than 79% of the tube wall are continuing. Of the original 25 tubes, 13 failed in less than 500 hr and the remaining 12 tubes have undergone 1200 hr of testing. A test of material of the same flaw depth was started at conditions of 4500 psi and 300 C. To date, no failures have occurred after 325 hr of test.

Nickel and iron base alloy corrosion rates in 550 C steam from atmospheric pressure to 5000 psi are virtually pressure independent. Elimination of the pressure variable should simplify further corrosion testing in high temperature steam.

Heat transfer experiments were continued with an electrically heated test section representing the downstream half of an N-Reactor fuel column with prototypic coolant piping. At a heat generation rate in the test section of 2500 kw (equivalent to a 5000 kw maximum tube power on the reactor), flow reductions were induced in each of the fuel element flow channels to determine resulting changes in coolant flow and heat distribution. In another experiment, the test section was run at a heat generation rate of 3100 kw to determine heat transfer conditions at

power level conditions equivalent to 124% of the maximum design tube power for the reactor. No sign of fuel temperatures higher than those predicted from earlier tests with smaller test sections was observed.

Laboratory heat transfer experiments were started to determine the thermal hydraulic characteristics of the K-V I&E fuel elements in zirconium process tubes in the K-Reactors. The results of the preliminary data indicate that more mixing of the coolant occurs in the last few feet of the fuel charge than had been anticipated.

The corrosion of aluminum coupons exposed to effluent water in a half-plant test at D-Reactor show corrosion rates of aluminum alloys 1.2 to 1.7 times greater at pH 7.0 than at pH 6.6. In another half-plant test at F-Reactor, corrosion of 1100 and 6061 alloy aluminum coupons was higher in water treated with 18 ppm alum than with water treated with 6 to 15 ppm alum.

Sodium nitrite (100 ppm) and sodium dichromate (100 ppm), but not sodium silicate (10-200 ppm SiO_2), were found to be effective in inhibiting the corrosion of carbon steel in stagnant process water.

Operation of KER-1 using NH_4OH for pH control continues to be satisfactory. The two instrumented crud detectors continue to function well and give no indications of crud deposition. Radioactivity of the piping in KER-2 (stainless steel, LiOH) was measured to be 2-5 times that in KER-1.

Corrosion of Zircaloy-2 in crevices under N-Reactor fuel element supports on an electrically heated sample was found to be less (1 mil vs. 7 mils) in a test using pH 10 NH_4OH than in a previous test with LiOH. The test results were compromised by the inadvertent introduction of LiOH during the latter part of the test.

Samples of N-Reactor silver-plated retaining rings and seal rings were tested at 550 F with pH 10 NH_4OH adjusted water. The plating was completely corroded away in 2 wk. Silver-bronze and copper samples were also corroded.

A section of pitted tubing from N-Reactor steam generator 4A was tested in TF-2 at 550 F, 1700 psi pH 10 NH_4OH to determine the rate of propagation of a pinhole leak. After 3 days, a crack 1 in. long suddenly developed 3-1/2 in. from the pinhole. The pinhole leak was produced by over-pressurizing the tubing before testing, hence is not typical of N-Reactor steam generator tubing.

An outer N-Reactor fuel element which had been irradiated to about 150% of goal exposure in 262 C water was rupture tested at 300 C with a normal N-Reactor shutdown cooling rate. Rupture started after 95 min incubation. The ruptured area was 2 in. in diameter. Weight loss was 25 g.

A modified Winkler procedure was developed which permits analysis for O_2 in the presence of high concentrations of N_2H_4 and N_2SO_3.

Improved ultrasonic nondestructive testing techniques are being applied to the Zircaloy tubing now being used for fabrication of swaged fuel rods. A few defects have been found in tubing originally passed on the basis of tests conducted with less sensitive equipment at the time the tubing was procured.

Extensive postirradiation examination of recently failed, swage compacted PRTR fuel elements and new analyses of the irradiated fuel and of nonirradiated archive samples support the hypothesis that the fuel was contaminated with an organic compound, probably from malfunctioning of equipment used in the fuel processing. A number of prototypic PRTR fuel capsules containing typical fuel contaminated with known quantities of water and oil were shipped to the MTR for irradiation to provide the needed information on the effects of water and hydrocarbon impurities on Zircaloy caldding failure rate.

Work continued on the determination and control of moisture in ceramic fuel. One sample of fines (-325 mesh) contained 160 ppm water detected by heating the sample to 180 C and 90 ppm of additional water

when heated to 400 C. Desorption of water from PuO_2 and UO_2–1% PuO_2 was found to occur at a higher temperature than for pure UO_2. A program was started to provide extremely dry atmospheres in selected glove boxes used for ceramic fuel fabrication.

An eddy current sensing coil was successfully incorporated into a vibrationally compacted UO_2 fuel rod. Twenty similar fuel rods are being fabricated for assembly into a Mark-I PRTR fuel element to be used for vibration tests in the EDEL-I Loop and in the PRTR.

Test capsules containing an early fuel formulation for EBWR fuel rods were recharged into the ETR for irradiation to approximately 1.4×10^{20} fissions/cm^3 (500 Mwd/ton).

Additional fission product studies concerning irradiated U-PuO pellets revealed a depressed Ru^{106} concentration often associated with the outer limit of the subgrain structure of the large columnar grain region.

Five swaged fuel elements containing impacted, 1 wt% PuO_2-UO_2 were fabricated for PRTR tests. Sufficient fuel material is in process for four clusters.

Assembly of UO_2-PuO_2 fuel rods for PCTR tests is in process.

Air leakage into hoods of the vibrational compaction line was essentially eliminated by application of a silicone rubber sealant to all gasketed joints.

A glove box was designed, fabricated and installed to house a large billet heating furnace. Seven impaction cans, each holding about 15 lb of oxide, can be heated to increase the impaction production rate, using a Model 1220-B Dynapak machine.

The second cycle rejuvenation fuel element was re-enriched and returned to the MTR for a third cycle of irradiation.

[redacted]

PRTR fuel elements (5117 and 5119), which had broken rod wire wraps repaired in October, were examined in the PRTR basin. Both elements appeared to be in a satisfactory condition after irradiation during October and November.

Mark III tube monitoring equipment has been completed and is ready for use upon installation.

A specimen from the eleventh Zircaloy-2 PRTR pressure tube to be destructively examined burst in a ductile fashion at room temperature and 9200 psig. The burst was a through crack, initiated at 16 mil deep ID pit, which allowed almost instantaneous depressurization from the burst pressure. With the use of an elastic patch over the ID side of this crack, the tube specimen was again burst tested at room temperature. It took 8800 psig to cause brittle propagation of this starting crack.

Calculations were made for boiling burnout conditions in the PRTR Rupture Loop when fueled with a fuel element designed for irradiation in the EBWR. It was found that a maximum tube power of 1800 kw would be possible at a flow rate of 123 gpm and an outlet temperature of 500 F.

Calculations were summarized which indicated (1) the possibility of PRTR operation at 125 Mw without modifications to the pressure relief valve and (2) that convective cooling of the fuel elements would be adequate in the event of a complete loss of pumping power.

Fretting corrosion investigations of the EDEL-I pressure tube having "sharpened" end brackets showed that fretting could be started in less than 123 hr at PRTR primary system conditions. Vibration measurements of eight selected tubes in the PRTR were completed and the data are being analyzed.

Zirconium concentrations in the PRTR primary water were normal, indicating no significant fretting in PRTR during the month.

Fretting corrosion of 1 mil in 1 mo of both the supports and the pressure tube was found through a test in TF-7 of a PRTR fuel element

with the original thin supports. (In a previous test under the same conditions, fretting did not occur with 360° supports.) Hydriding of 200-300 ppm was found near the fretted region of a PRTR wire wrap compared to 50 ppm in unfretted regions.

A modified oxalic-peroxide-peracetic solution was successfully used in IRP to decontaminate mixed uranium oxides.

UN–20 wt% PuN solid solution pellets are being prepared for irradiation testing in the MTR. Samples of PuN have been sintered to 97% TD.

The high temperature brew furnace is now installed in the 308 Building.

A new apparatus to measure electrical conductivity has operated successfully to 1600 C.

The reflection electron microscope has been altered to allow observations at temperatures to 3100 C. Studies were made on a 20 vol% UO_2 tungsten cermet using the new equipment.

An improved electron microscope replication method for irradiated specimens has been developed and appears to give highly satisfactory results.

Resolution of autoradiographs obtained from irradiated fuel elements has been further improved by using a "stripping" film technique.

Recent studies reveal that many small pores observed in irradiated fuel elements may actually be metallic inclusions. This finding is significant to interpretations of microstructures of irradiated fuel elements.

Final resolution of uncertainties of fuel structure associated with central melting during irradiation of UO_2 fuel elements was achieved by a series of short- and long-term irradiations of fuel pellets in which tungsten shot was relocated from its originally random distribution throughout the fuel.

New factors have been derived for conversion of Mwd/adjacent ton to ne..ron flux having energy greater than 0.18 Mev.

Continued study of the effective size of graphite on oxidation rate by water vapor has shown that 1.5 and 2 in. diam samples oxidize at about the same rate, whereas 0.5 in. diam samples oxidize much more rapidly. Diffusion within the sample is important to the oxidation rate.

In the gas system (carbon monoxide-water-argon) it has been found that gamma energy absorbed by the argon is efficiently transferred to the other components resulting in an enhanced radiation induced reaction.

In the irradiation testing (at about 1000 C) of the graphites impregnated with furfuryl alcohol, expansion of some samples was observed for the first time in the last irradiation period. This effect is of considerable theoretical and practical importance. Contraction saturation of test graphites having different heat treatments was also observed.

Twelve thoria fuel elements were canned and charged into a production reactor. Thoria fabrication development equipment will be installed. Preliminary experiments in pneumatic impaction of thorium oxide were performed. Spray calcined ThO_2, impacted at 1200 C and 357,000 psi impact pressure, achieved a density of 9.60 g/cc (95.7% TD).

Concluding a 1 mo study, a document was issued and transmitted to AEC-DRD describing several possible plutonium fueled core designs for the Military Compact Reactor. In the document the potential benefits in size and weight of the systems were estimated, and associated technological problems were discussed.

Calculations of the time delay in the Doppler coefficient in mixed UO_2-PuO_2 fuels have been completed. For 1.5 wt% PuO_2-UO_2 EBWR fuel, the time constant for conduction heat transfer from the PuO_2 to the UO_2 is 1 msec. This is well under the shortest attainable periods in EBWR of 3 to 5 msec. Investigation of radiant heat transfer indicates a

substantial contribution to the UO_2 temperature is probable from this mechanism. The conduction model is therefore believed to be pessimistic in the safe direction for purposes of excursion analysis.

Theoretical calculations indicated an excessive pressure drop (~130 psi) for the proposed tensile specimen assembly for the ATR gas loop. These calculations were substantiated by experimental tests run in air on an assembly with one-fourth the proposed length. Engineering studies of an alternate holder indicate an acceptably low pressure drop (~15 psi).

Inconel 600 samples exposed to oxygen and air at 1200 C are more resistant to oxidation than Haynes 25 or Hastelloy X.

Stress-rupture tests at 2100 F (1149 C) on Haynes 25 have been completed. The extrapolated data predicts a 1000 hr rupture life at 1100 psi for this temperature.

Tensile tests on both cold worked and annealed Zircaloy-2 irradiated at 280 C (540 F) show that crystallographic anisotropy continues to exhibit marked influence for integrated neutron exposures up to 2.0×10^{21} nvt (fast). The difference in strength with direction in the cold worked material is attributed to differences in deformation mechanisms (principally slip in the rolling direction and twinning in the transverse direction). Deformation by twinning appears to be less affected by thermal recovery at this irradiation temperature than is deformation by slip.

Interpretation of strain-rate change experiments performed at room temperature and 200 K (-100 F) indicates that work hardening in iron at these temperatures was a result of long range stresses buildup rather than from short range stresses (≤10 Å) that would be affected by thermal activation.

Tensile tests performed on the Inconel 625, one of the materials being examined for application in the BONUS superheat reactor, indicated

only small changes in tensile properties that resulted from irradiation at 280 C (540 F) to 5 x 10^{20} nvt. However, this alloy exhibited a 65% increase in yield strength after being irradiated in 80 C (176 F) water to an exposure of 1 x 10^{20} nvt.

The in-reactor creep test started last month on 20% cold worked Zircaloy-2 at 20,000 psi and 375 C (707 F) has accumulated over 1000 hr. Two outages have occurred during this time; creep rate increased during both outages.

A creep capsule containing annealed AISI 304 SS has been completed and is awaiting charging into the reactor. The initial test condition will be 30,000 psi stress and 500 C (932 F).

A metallographic technique for plutonium which involves coating a specimen with a thin film of plastic is under investigation. It has been found that the plastic coating not only retards oxidation but improves image resolution and contrast as well.

X-ray measurements of lattice parameter and line width on a second series of irradiated molybdenum foils did not confirm the results of earlier experiments. Reasons for the discrepancy are being sought.

Studies of nickel quench-hardening are in progress. It has been found that increases in yield stress after quenching, without postquench aging treatments, are achieved only in nickel of the highest purity (99.997%). This suggests that impurity atoms have a very pronounced effect on the distribution of quenched-in vacancies. Further verification of these results is continuing.

Strain rate cycling experiments conducted at room temperature with unirradiated molybdenum have revealed that the variation of Δt with t is linear at low stress levels and becomes exponential at stresses greater than twice the yield stress.

Electron microscope examination of molybdenum foils containing < 10 ppm carbon irradiated to 1 and 4 x 10^{18} nvt show a striking increase

in the number of spot defects at the higher exposure. Molybdenum foils containing 100-200 ppm carbon, irradiated to the same exposure, did not show such an increase in defect number. It is believed that interstitial atoms act as nucleating centers for defects created during irradiation.

Two controlled temperature general swelling capsules reached their goal exposure and were discharged. One operated at 575 C, the other at 500 C. Two other capsules were charged into a single reactor test hole. Seven irradiated capsules, now being stored, are available for recovery and examination.

A replica from a polished and etched cross section of a tubular fuel element irradiated in a hot water, high pressure loop has been examined in the electron microscope. Profuse fission gas porosity was observed in the hot central regions of the section and a few tiny pores were seen in the cold edge regions. A few areas existed in the hotter regions which contained crystallographically aligned microtears. It appears that large external restraints are effective in suppressing most of the grain boundary and crystallographic tearing observed in small, high purity uranium specimens irradiated unrestrained, but have little or no effect on the formation of fission gas porosity.

Prototypical fuel handling tests with 12- and 18-in. N-Reactor fuel elements were completed on the magazine loader and charge machine mockup. No significant self-support damage (>5 mils) was detected on 24 in. fuel element self-supports even though back pressures up to 1900 lb (forces which previously caused damage) were exerted on the fuel element column.

In the study of tungsten extrusion and the bonding of tungsten to tungsten, three tungsten-molybdenum billets have been extruded at preheat temperatures varying from 2200 to 1800 C. Tungsten components were nested tubes. Surface quality improved with each extrusion. Bonding also improved and was found to be complete in the rear of the later extrusions.

[redacted]

In the evaluation of cladding and structural materials for use in the HTLTR, exposure to nitrogen and graphite at 1000-1200 C caused chromium bearing alloys to embrittle and undergo pronounced decrease in room temperature tensile properties. As a result, Hastelloy X and Inconel 625 have been dropped. Testing is continuing on Hastelloy B, Nickel A, TD Nickel, and Inconel 600.

Samples of depleted UO_2 to be used for EBWR fuel elements were pneumatically impacted for use in preparing thermal diffusivity specimens.

An additional shipment of 561 EBWR cladding tubes was received during the month. A centrifugal hone was designed and fabricated to rework at HAPO the 789 tubes earlier received from the vendor with burred and scratched internal surfaces. An additional 600 tubes were ordered to insure enough acceptable tubes for EBWR requirements. In addition, 500 tubes were ordered for use in fabrication of PRTR fuel rods by swaging. Other Zircaloy cladding tubes were ordered for fabrication of 12 nested, tubular PRTR fuel elements of the Mark III type.

Preparation of EBWR fuel material (impacted, 1 wt% PuO_2, UO_2) is in process.

Two irradiation tests of UO_2-W cermet fuel plates were conducted for an offsite customer. Plate temperatures of approximately 3000 C were achieved during the short irradiations.

2. Physics and Instruments

Procurement and fabrication of materials and instrumentation for N-Reactor physics startup tests are 75% complete. Fuel elements required for cold tests are on hand. The relative activation rates to be expected in the N-Reactor spectrum are being determined.

Simulation of N-Reactor activities by analog is progressing rapidly. The steam generator portion of the primary coolant system simulation is functioning properly, although the complete primary system with closed loop controllers has not yet been successfully simulated. The secondary loop, excluding the steam generators, has been formulated.

Installation of fuel failure detection and flow control instrument systems for the PRTR Rupture Loop was essentially completed. Loop activity is being monitored to obtain background data.

Fourteen criticality experiments were completed this month using the new Remote Split-Table Machine. Reflector savings were measured for various combinations of materials (water, Lucite, stainless steel containing gadolinium, camdium, and enriched uranyl nitrate solution) placed at the ends of a rectangular prism core. Specific data pertinent to design of an enriched uranium dissolver were obtained also.

The range of application of critical mass experiments has been extended through completion of a computer code for accurately calculating the geometrical buckling of a sphere partially filled with fissile solution.

Another series of total cross section measurements for neutrons of 3 to 15 Mev was completed. First Hanford measurements were made on six samples for which no data previously existed including Cr^{53}, the first Hanford-measured electromagnetically separated isotope. Measurements were repeated on four other elements.

Studies of slow neutron scattering from 95 C water continued. Measurements of the detection efficiency of the analyzing spectrometer were completed.

Work on the Phoenix fuel program continued with studies of time-step-size effects on calculated multiplication vs. exposure curves. The effect of differing treatments of the Pu^{240} resonance was also explored. A Phoenix Fuel Program Proposal is being prepared.

A critical appraisal of the Pu-Al fuel burnup studies is under way in preparation for publishing the low exposure data and assuring good accuracy for the remaining experiments. A paper describing the work was given at the New York City ANS Meeting in November.

Considerable work continued on the development of nuclear physics methods and codes. Additions to SMGAS code allow solution of the modified heavy gas equation with a source. Work has continued on RBU and

the RBU Basic Library. Revisions were made to the codes: COMBO, HFN, INTERSET, PHYSICS CHAIN, and HRG. The use of working tapes has been incorporated into the transport theory code S-XII; the size of problems which can be handled is now approximately doubled with no loss of calculation time or reliability from tape handling.

The necessary measurements of perturbations encountered in the approach-to-critical experiments with the 1.8% Pu-Al were performed.

Extensive calculations were performed to guide planning of the experimental program with the HTLTR. Strength and metallographic tests were made on three possible structural materials for the HTLTR: Nickel A, TD Nickel, and Hastelloy B.

Operational tests conducted on the PRTR fuel underwater gamma scanner have disclosed that it may be necessary to reduce the time required to scan fuel rods and activated wires by modifying the system so that an increased counting rate is obtained.

Details of installing the new process control computer and providing the necessary operator training courses are being negotiated with the computer vendor.

Four plutonium samples irradiated to different exposures in connection with the neutron flux monitor program have been prepared for analysis by mass spectrograph. Specifications have been prepared for the offsite construction of experimental regenerating units, and procedures have been initiated to obtain 100 mg U^{234} for this purpose. Fabrication of new cables for the B^{11} detector is proceeding, and the performance of Al^{27}, B^{11}, Rh^{103}, and Tc^{99} in this type of detector is being investigated theoretically.

The multiparameter eddy current nondestructive testing technique is being applied to the practical inspection of tubing. Difficulties are being experienced, but these were not unanticipated. The eddy current hydride detector successfully revealed 250 ppm hydrogen in two samples of N-Reactor process tubes.

Studies were begun of the performance characteristics of the new secondary-electron scintillation ion-detector on the mass spectrometer. Preliminary results show good promise that the detector will operate without systematic errors owing to the mass number of the ion or spatial position of the analyzed ion beam.

A coincidence beta ray counter offers promise as a means of detecting P^{32} that can be used with the truck whole body counter. Work on plutonium counters proceeded slowly because of multichannel analyzer malfunction. The neutron generator for the University of Washington spermatogenesis project was received, assembled, and put into operation. Difficulty was encountered and overcome in calibrating the calorimeter being used for Pm^{147} half-life measurements.

Significant progress was made in atmospheric diffusion research through demonstration that the lateral growth of a plume out to 16 miles can be expressed solely in terms of parameters readily calculable from wind measurements at the source. The theoretical equation was proposed in 1921, but, as far as is known, the Hanford results are the first determination of the functional forms of the coefficients needed to make application possible.

An ultrasonic test was developed to evaluate the quality of the spot welds on the supports of N-Reactor fuel elements. The nondestructive test was demonstrated to measure uranium penetration in the weld, lack of penetration, voids and unbonds. Sectioning of test specimens and metallurgical evaluation verified the ultrasonic test results.

A complete nondestructive inspection was applied to analyze the condition of a 40 ton crane hook assembly that was dropped in the Purex Canyon area. Radiography was used to confirm the condition of the critical parts. All major parts were not damaged; however, some bolts and retaining pins required replacement.

[redacted]

3. Chemistry

Comparative tests of aluminum coatings in the flux zone of the C-Reactor showed rather definitely that organic surface coatings are not able to survive the high-flux, high-surface-temperature conditions in the reactors, hence cannot greatly inhibit the radionuclide production process.

A one-slug dissolver, for use in dissolving irradiated thorium metal slugs, was fabricated, tested, and installed in B-Cell of the High Level Radiochemistry Facility (325-A). In testing the dissolver, a 6 in. thorium fuel element (unirradiated) was dissolved in 68 hr. Other laboratory work was directed at flowsheet definition for in-cell recovery of U^{233} from the dissolved thorium fuel elements.

The 8 in. x 10 ft "cold" spray calciner was experimented with to evaluate the possibility of applying the spray calcination technique to the conversion of thorium nitrate solution to fuel-grade thoria. The product was a free-flowing powder with an estimated particle size of ca. 50 μ, a tap density of ca. 23% of theoretical, and excellent behavior upon cold pressing and pneumatic impacting.

Routine recovery of Np^{237} and waste Pu^{239} from Purex first-cycle acidic waste by amine extraction is under consideration. Laboratory studies of the fate of Tc^{99} in this operation showed that direct recovery of about 80% of the Tc^{99} which enters the Purex Plant would also be achieved.

Preliminary experiments were performed to test the separation of Po^{210} from bismuth by carrying it on cerium bismuthide precipitated from molten bismuth metal; 81% of the Po^{210} concentrated in 5% of the bismuth.

Hydroxylamine sulfate was demonstrated to be a superior stripping agent (relative to oxalic acid) for stripping plutonium and neptunium from trilauryl amine solutions. The strip solution may be routed directly to purification steps without prior treatment; oxalic acid must be destroyed.

Additional calculations on isotopic content of recycled thorium showed that at 1 ppm U^{232} in U^{233}, the Th^{228} in recycled thorium will decrease each cycle instead of increasing as previously thought. In other studies on thorium, approximately 22 lb of ThO_2 were prepared for target element fabrication by agitated-bed, direct denitration of thorium nitrate.

Based on the persistence of low but detectable concentrations of gross beta-emitters in several wells within 0.5 mile of the Columbia River, a nominal travel time of 7 yr for radioruthenium to move from the Purex plant to these wells is established.

The transient centerline temperature behavior in waste containers (Sr^{90} and Cs^{137} sorbed on zeolites) is being studied by an analytical solution programmed for the IBM-7090. The specific concern is with changes encountered when the through-gas drying step is terminated and heat removal shifts to radial transfer to the surroundings.

An alternate, less expensive (than hot air evaporation) approach to in-tank solidification is under development. In this concept, electrical heaters are immersed directly into the tank contents.

About 1 kg of Tc^{99}, previously isolated by adsorption from Purex alkaline tank supernate on an ion exchange bed, was eluted and the eluate successfully evaporated (at Hot Semiworks) to a volume of 100 liters. The solution, which also contains about 1M iron, presumably from concentrator corrosion, is now awaiting purification.

Laboratory tests demonstrated a satisfactory modified flowsheet for removal of cesium from Redox alkaline high-level waste supernate. The modified flowsheet, using water instead of ammonium carbonate for a scrub solution, results in considerable savings in essential material costs as well as simplified disposal of the waste from the scrub step.

Preliminary laboratory studies have been made of a simplified method of recovering Pu^{238} from solid Np^{237} target elements. Nearly complete (>99%) separation of the two actinides can be achieved by a simple, batch, amine extraction-scrub-strip cycle.

Of possible interest for the Hanford Waste Management Program is the removal of trivalent rare earths by inorganic ion exchange processing. Thermodynamic equilibrium constants for the exchange reaction in a sodium-cerium system were determined for two exchangers, Linde 4A and 13X.

A barium titanate transducer was evaluated as a means of measuring small rates of temperature change which may be encountered in process streams containing high concentrations of fission products. Good sensitivity was observed at temperature changes in the range of 0.004 to 0.1°C per min.

Difficulties in dissolving PuO_2 on an engineering scale in the Salt Cycle solvent are attributed to traces of moisture in cell atmosphere contacting the molten salt medium.

In recent studies of the factors important in the electrocodeposition of uranium and plutonium dioxides from chloride melts, it was found possible to achieve plutonium enrichment factors as high as 50 and rare earth decontamination factors of about 30. Thus far, however, electrolysis under the conditions needed for these results recovered only about half the plutonium from the melt.

Two full-level runs were made during the month in the A-Cell spray calciner. A run with Redox waste to which glass making additives had been added was successful, with low ruthenium evolution (< 2%). A good, glassy-appearing melt was produced. The other run, with current sugar-denitrated Purex waste, was terminated prematurely because of electrical power difficulties. Early samples indicated very low ruthenium evolution in this run, also.

Glass making studies were continued. In one investigation, it was found that the formation of a sulfate-rich second phase in ORNL lithium glass is critically related to the sulfate/phosphate ratio. In other laboratory experiments, incorporation of mixed fission products equivalent to 0.5 to 4 x 10^4 Mwd/ton in a Purex phosphate glass resulted in a polyphase

melt at temperatures as high as 1275 C. The resultant solids were hard and strong, however, and may represent satisfactory storage media, even though inhomogeneous.

Decontamination factors in the ion exchange process for treating Purex tank farm condensate in the pilot plant were much lower than those obtained previously with solids-free condensate. The cesium DF across the thin bed of clinoptilolite was only 1.7, and the ruthenium DF across the strong base anion exchanger was about 4.

Implementing of the new Containment Systems Experiment is under way. The location of the experiment will be in T-Plant canyon head-end section rather than U-Plant or 200-N as originally proposed.

The test model, 662 node electrical analog network was constructed, and a program was prepared to get the analytical solution for potentials at each node point of the analog.

Initial application of multidimensional gamma spectrometry to the determination of radionuclides in urine showed that Na^{24}, K^{40}, Cu^{64}, and Cs^{137} could be identified in an overnight count on a 24 hr sample. After decay of the Na^{24}, the radionuclides Cr^{51}, Zn^{65}, Sc^{46}, Co^{60}, and perhaps Np^{239} can be measured.

The protection index of phenol was measured as a function of temperature. The data show the same pattern of increase of protection index with decreasing temperature as was found with all other solutes examined.

Several techniques are being investigated for producing thin-walled compactions by the pneumatic impacting process. Two schemes with good preliminary results are: (1) the use of a thick-walled can (designed to give uniform wall thickness) and subsequent removal of excess metal by machining, and (2) the use of a thin-walled stainless steel can surrounded by a tight-fitting mild steel jacket, which can be subsequently dissolved in acid.

4. Biology

Dr. Jacob Eapen of the Biology Division, Atomic Energy Establishment, Trombay, India, began a 1 yr assignment with this laboratory on November 25, 1963. He is sponsored by the International Atomic Energy Agency.

N-Reactor personnel are considering using two new materials for cleaning pipe systems. These are hydrazine hydrate and Vertan 675. Both of these are being tested for toxicity to fish. The hydrazine hydrate appears to be highly toxic.

Trout which had been exposed in the laboratory for 22 wk to temperatures 4 F above the normal river temperature showed a higher mortality rate than controls. Most of the mortality is attributed to columnaris.

The neutrophil reserve capacity of swine receiving 25 μc Sr^{90}/day (bone marrow, 3-4 rads/day) was tested by injecting a bacterial endotoxin. These animals were shown to have the same neutrophil reserve as controls.

Np^{237} administered to rats was found to increase liver fats as indicated by the incorporation of C^{14}-labeled acetate.

Plutonium injected intravenously into rats was found to be secreted into parts of the intestine in the following decreasing order: duodenum, jejunum, ileum, large intestine, and cecum.

Although DTPA and TTHA cause the removal of 25% of plutonium nitrate inhaled by rats, another chelating agent, diethyldithiocarbamate was found ineffective.

Calcined cerium oxide particles in lungs showed a half-life of 550 days. This is to be compared with a biological half-life of less than 400 days for the material prepared by milder chemical oxidation.

An unusually high incidence of retinal degeneration is present among dogs in our colony. This is being attributed to heredity and early history of diseases.

Zn^{65} and Cr^{51} levels in plankton taken from the Columbia River were markedly higher during November than for October.

5. Programming

In a rapidly expanding nuclear economy the doubling-time is more nearly expressed by a compound interest (i.e., geometric progression) relationship than by the simple interest approach which has been common practice. Doubling-time is defined as the number of years required for a breeder system to double the size of the initial fissile inventory. The initial inventory usually includes a minimum out-of-reactor stockpile.

Changes in wind speed and direction during a several-hour (as opposed to instantaneous) release of fission products tend to reduce the distance at which a given dose would be expected. This finding could have a significant impact on reactor siting considerations.

TECHNICAL AND OTHER SERVICES

A fire occurred in the Redox Final Concentration Building (233-S) on November 6, 1963. Ten firemen involved in fighting the fire received skin contamination ranging from a few thousand dis/min to greater than 40,000 dis/min (readings taken after personnel showered). Positive nasal contamination was found only on two individuals. Three operators who were working in the 233-S Building when the fire occurred left immediately. Surveys showed that plutonium contamination had spread throughout a major portion of the building and ranged from a few hundred dis/min to several hundred thousand dis/min. Some contamination was found on the roof of the building around three ventilation ducts and the flashing around the edge of the roof where smoke had been vented. The contamination levels have been reduced in most of the building with the exception of the stairwell and process area #2. Ground surveys in the vicinity of 200-W Area following the fire indicated no significant release of plutonium from the building. Preliminary analysis of bioassay samples are inconclusive. However, it appears that any internal deposition of plutonium from this incident will be minor.

There were no new plutonium deposition cases confirmed during the month. A re-evaluation of a deposition case previously estimated to be less than 10% of the MPBB resulted in its removal from the list of confirmed deposition cases. The total number of individuals who have internal plutonium deposition at Hanford is 324 of which 234 are currently employed.

About 0.2 curie of I^{131} was emitted from the Purex stack on November 6, 1963, during dissolution of the remainder of the metal in C-cell dissolver. This metal had been held for decay of the I^{131} content since September 2, 1963, when it was inadvertently charged after only 18 days cooling.

Concentrations of fallout materials in the air of the Pacific Northwest have slowly decreased since August 1963. The average value for November, 1 pc β/m^3, was the lowest monthly average observed since the USSR resumed testing in the fall of 1961.

A study was initiated to assess the feasibility of predicting variation in the thickness of N-Reactor fuel cladding employing pre-extrusion measurements of the uranium billet and the zirconium shell.

Discussions were held, with all sites involved, concerning a suggestion to increase the minimum plutonium content specification for alpha weapon components. A counter proposal to retain the use of a statistical tolerance statement for demonstration of specification compliance is being prepared by CPD.

A number of linear models have been fitted in an attempt to calibrate probolog measurements with wall thickness.

The EDPM Program which prepares the magnetic tape input to the prototype Sheffield Rotary Contour Gauge has been completed and debugged.

Closed form solutions were obtained for two instrument development problems.

[redacted]

Calculation of the initial set of channel group probabilities for inclusion in the IRA system has been completed and a comparison is being made with the empirical values previously used. In progress is a study of the variability of these probabilities and the effect on fit criteria and the standard error estimates in the GEM Program.

Closed form solutions were obtained to a set of four simultaneous nonhomogeneous, linear, ordinary differential equations. This represents a first attempt at a model of biological uptake and retention processes.

Empirical equations predicting the concentration of the 239, 240, 241, and 242 isotopes of plutonium as a function of percent burnup were fitted.

SUPPORTING FUNCTIONS

PRTR output for November was 357 Mwd, for an experimental time efficiency of 43.5% and a plant efficiency of 17%. There were nine operating periods during the month: one of which was terminated for scheduled refueling and planned maintenance; one was terminated manually owing to indications of a leak inside of HX-5; three were terminated by scrams while switching to automatic control; two were terminated by spurious flow monitor trips; one was terminated by a manual shutdown when the charge on the 125 v dc batteries was found to be below specifications; and one was terminated by a high ΔP trip on a Rupture Loop. A summary of the fuel irradiation program as of November 30, 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	11	982.3	1	211.2	73	7571.1			85	8764.6
Maximum		99.6		211.2		178.5				
Average		89.3		211.2		103.7				
In Basin	32	2600.1	31	3603.8	14	237.5			77	6441.4
Buried							1	7.3	1	7.3
Chemical Processing	32	2309.3	35	1965.8					67	4275.1
Program Totals	75	5891.7	67	5780.8	87	7808.6	1	7.3	230	19488.4

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

A total of 159 PRTR outage hours was charged to repair work. Main items were:

HX-5 leak location and repair	114 hr
Valves	11 hr
Rupture Loop equipment	8 hr

A total of 194 PRTR outage hours was charged to experimental time. Main items were:

Location and removal of three leaker fuel elements	54 hr
Rupture Loop tests	50 hr
Charge-discharge	42 hr
Core Level Instrumentation test	17 hr
Process tube inspection	15 hr

The November heavy water inventory indicates a loss of 1034 lb for the month.

The Plutonium Recycle Critical Facility core was dismantled. The D_2O moderator and all fuel elements were removed to storage. Safety rods, control rods, and miscellaneous hardware items for use with a D_2O moderator were mothballed. Deactivation work was completed on November 8. Project work (CGH-999, conversion of facility to H_2O moderator) began on November 11 and lasted through month-end.

Inspection of the PRCF core vessel interior revealed no active corrosion over the past 10 mo of moderator system operation.

Operation of the Fuel Element Rupture Test Facility's rupture loop was started during the month. An advanced rupture detection system, installed by Physics and Instruments, was also activated and base measurements established during initial operation before installing the first fuel element near the end of the month.

Total productive time in Technical Shops Operation for the period was 21,564 hr. Distribution of time was as follows:

[redacted]

	Man Hours	% of Total
N-Reactor Department	3 903	18.09
Irradiation Processing Department	4 002	18.56
Chemical Processing Department	555	2.58
Hanford Laboratories	13 104	60.77
Hanford Utilities and Purchasing Department	0	---

Laboratory Maintenance Operation's total productive time was 19,300 hr of 23,000 hr potentially available. Of the total productive time, 92.5% was expended in support of Hanford Laboratories components, with the remaining 7.5% directed toward providing service for other HAPO organizations. Manpower utilization for November was as follows:

A. Shop Work		2700 hr
B. Maintenance		7200 hr
1. Preventive Maintenance	1900 hr	
2. Emergency or Unscheduled Maintenance	1700 hr	
3. Normal Scheduled Maintenance	3600 hr	
4. Overtime (Included in above figures)	1300 hr	
C. R&D Assistance		9400 hr

The heavy water inventory at the end of November 1963 showed a loss of 1,036 lb ($14,276) for the PRTR and a loss of 37 lb ($510) for the PRCF. Heavy water scrap generated during the month amounted to 6,015 lb, resulting in a $10,045 charge to operating cost. A shipment of heavy water (19,091 lb) was made to SROO on November 4. This shipment has a value of $240,946. An order has been placed with SROO for 27,500 lb ($374,000) of new heavy water to be delivered January 15, 1964.

Cumulative data on Hanford visits:

	Number of Visitors	
	November	Since June 13, 1962
Visitors Center	1 071	59 184
Plant Tours	116	n.a.

[redacted]

Summarization of HAPO professional recruiting activity this month:

	Plant Visits	Offers Extended	Acceptances Received	Rejections Received	Open Offers at Month-End
Ph. D.	2	3	0	0	2
BS/MS (Direct Placement)		2	1	0	3
BS/MS (Program)		0	1	1	2

Six technical graduates were placed on permanent assignments and one terminated, reducing the current program strength to 69.

Authorized funds for nine active projects total $6,505,500. The total estimated cost of these projects is $10,799,000. Expenditures through October were $1,578,000.

H M Parker

Manager, Hanford Laboratories

HM Parker:JEB:esl

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REACTOR AND FUELS LABORATORY MONTHLY REPORT

NOVEMBER 1963

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 02 PROGRAM

1. Metallic Fuel Development

N-Reactor Fuel Evaluation. Radiometallurgical examination of an N-outer fuel component irradiated to 3100 MWD/T near the center of a NAE charge in KER Loop 4 was started. The element operated during the first three-fourths of its exposure with a volume mean fuel temperature of 400 C. The last one-fourth of its total exposure was accumulated at reduced coolant temperatures. A 2.5% increase in the volume of the fuel was experienced during the irradiation. Superficial examination has revealed no deficiencies in the performance of this outer component; i.e., no obvious warp, no bumping or rippling of the clad surfaces, no crud deposits or corrosion effects, and the inner bore appears undistorted. Metallographic examination of the fuel at magnifications up to 1000X reveals no observable swelling porosity or no grain boundary tearing or cracking. The uranium shows the highly distorted structure typical of low temperature irradiation. Examination will be extended to higher magnifications by electron microscopy to resolve the porosity which should exist as the source of the 2.5% volume increase.

Fluted Fuel Element Irradiation. The fluted N-single tube size fuel elements being irradiated in the ETR are currently undergoing their eighth cycle of irradiation in the ETR M-3 pressurized loop. At the end of the seventh cycle these elements had achieved an exposure of 1400 MWD/T at a maximum fuel temperature of 520 C and had shown 0.8% volume increase in the fuel. This compares to 0.5% volume increase at the end of the sixth cycle (1200 MWD/T).

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 strain capabilities of Zircaloy-2 cladding as a function of cladding thickness uniformity, temperature, and exposure. All the samples have been examined visually and diameter measurements completed. The data from all these fuel rods confirms that irradiation damage and cladding thickness uniformity are the major factors effecting the stability of Zircaloy-2 cladding. It was shown on samples with uniform cladding thickness that below an average cladding temperature

of 325-350 C the total cladding strain limit is approximately 1.5%. The introduction of a striation in the cladding at the fuel cladding interface of 12 to 15% of the cladding thickness reduces the cladding strain limit to approximately 0.8%. The striation effect is independent of temperature, that is, it overrides the increase in the total cladding strain limit above 350 C observed on the samples with uniform cladding. Transverse cross section of samples are being prepared to investigate more thoroughly the cladding conditions under which instability is observed.

Lithium-Aluminum Target Irradiation. Irradiation of eight experimental lithium-aluminum target elements in KER Loop 2 has continued at a coolant pressure of 1600 psi and a calculated target temperature of 300 C.

Further examination of the target elements which were discharged from Loop 1 after 61 days at full reactor power continues to show no evidence of swelling of the target core.

Empirical Fuel Swelling Expression. Experimental swelling data obtained from two KER charges of NAE fuel elements and four KER charges of KSE-3 and KSE-5 fuel elements have been used to further evaluate an empirical fuel swelling expression. Exposures from 360 to 3600 MWD/T at volume mean fuel temperatures from 250 to 535 C produced fuel volume increases ranging from 0 to 5.4%. The developed expression is,

$$\%\Delta V = B \left\{ 3 + \left[\frac{550\ T_r}{P_r + P_c} \right] \cdot \left[\frac{1}{1 + \exp\left(\frac{450 - T - 100B}{20} \right)} \right] \right\}$$

B = burnup (at%)

$T_r = \frac{\text{vol. mean fuel temp. } (^{\circ}C)}{273}$

P_r = restraining pressure of clad (atmosphere)

P_c = coolant pressure (atmosphere)

T = vol. mean fuel temp. ($^{\circ}C$).

Calculated fuel swelling using the above expression for each element of the six KER charges compares well with the experimentally observed fuel swelling.

Fuel Element Measurement Equipment. A device for measuring the geometry of the end cap region of N-fuel elements has been fabricated

and assembled. The device is capable of making continuous measurements of the outer radius and wall thickness in longitudinal traverses of up to 1.5 inches from the ends of the fuel element. Preliminary data show that these measurements are reproducible to ± 0.0005 inch.

Fuel Fouling Detector. A Zircaloy-2 clad thorium-uranium fueled, fuel fouling detector is successfully operating in the ETR P-7 loop under prototypic N-Reactor conditions. The original assembly, which was charged at the beginning of the cycle, was damaged and time did not permit its replacement with another assembly from which temperature measurements could be taken. The present probe, which has a short, sealed thermocouple, was charged November 18, at the mid-cycle shutdown.

Li-Al Physics Elements. A total of 18 poison column elements and 80 target elements were fabricated, complete with steel shod aluminum supports, to meet the November 15 deadline for N-Department. An additional two tubes of target elements were completed well ahead of the December 1 deadline. These elements have a 1% LiAl core and were clad with a coextruded aluminum jacket. The fabrication of these elements required considerable development work including a method of coextruding an aluminum jacket around the core; a procedure for protecting the crucible during melting; a method of cleaning and special procedures for welding. A majority of the elements were extruded on the 333 press from components which were cast and extruded at Hanford. The remaining elements to complete the charge were extruded in the 306 Building facility from welded billet shells and cast cores.

Metal Cleaning - Bright Etching of Aluminum. Alloy 1100 aluminum clad, aluminum - 1 wt% lithium core, poison and target elements for N-Reactor startup physics tests were bright etched using a phosphoric-nitric acid bright dipping bath. Bath composition and conditions were:

94% ortho phosphoric acid (85% H_3PO_4)

6% nitric acid (70% HNO_3)

Trace metallic copper

Temperature: 90-95 C

Time: 5-10 minutes.

Prior to brightening the elements were cleaned by dry abrasive blasting using 220 mesh aluminum oxide grit. The matte finish produced by the

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abrasive pretreatment was still evident in the lustrous matte finish of the etched element. Sixteen elements 2-5/16" diameter x 26" long, 76 elements 1¼" diameter x 26" long, and 1050 supports (X-8001 Al alloy) were bright etched prior to the attachment of the supports. No cleaning step was performed after attachment of the supports.

Weld Developments. In an effort to determine the conditions for projection welding of hot headed N-Reactor outer fuel tubes, 16 pieces were welded on a 1200 KVA 600,000 ampere resistance welding machine at U.S. Army Rocky Mountain Arsenal, Denver, Colorado.

Of the 16 completed welds, four had satisfactory external appearance, seven were used in arriving at approximate conditions, and five showed various defects obvious to casual inspection. No metallographic facilities were available on site so the welds had to be judged by external appearance and basics such as chisel tests. The four "good" welds, representing three sets of weld conditions, were examined destructively at Hanford. Insufficient heat shown by incomplete welding was evident in the lower power welds. Porosity, apparently caused by insufficient pressure during the weld cycle, showed in several of the samples.

Another attempt will be made using this information after a revision of the tooling. The wafer collet current distributor section of the tooling contained two nickel wafers. These proved to be unsatisfactory because of sticking to the fuel element and will have to be replaced with two wafers of other material.

Second Generation N-Outer Support. An "on hand" alloy of 2Nb, 2Sn, 96 zirconium was rolled to 0.040-inch thick sheet for an alternate test material for the second generation outer support. The sheet was not slit into support feed strips for die forming because it was very severely edge cracked and extremely hard and brittle. Several alloy compositions of Nb-Zr will be melted into buttons for fabrication feasibility tests. Design is completed and fabrication has started on a die to hot forge a Zircaloy-2 shape of considerable increased strength for the second generation support. This hot forging will be done on one of the resistance welders in 306 Building. The die should be ready for proofing about December 20.

Study of U-Zr, U-Nb and U-Nb-Zr Alloys. A program has been started to study the properties of uranium alloys with additions of Ni and Zr up to 5%. These alloys have been arc melted and are currently being rolled with various degrees of cold work. The corrosion evaluation will take place in warm water at three different pH levels and with various heat treatments on the alloys themselves. A complete metallurgical evaluation is also being made.

Uranium Sulfur Alloys. Powdered uranium monosulfide has been sintered and broken up to form suitable size pieces for melting with uranium wafers to form uranium-sulfur alloys up to 1 wt% sulfur. The uranium-uranium sulfide mixture when arc melted to form a homogeneous button indicated a loss in surface tension due to the additions of the sulfide. Subsequent rolling of the buttons showed edge cracking and general brittleness above 750 ppm sulfur.

Coextrusion of Aluminum Clad Li-Al. Several 2S aluminum clad Li-Al alloy billets were coextruded at reductions of 7 to 1 and 21 to 1 to develop an extrusion technique which would produce satisfactory extrusions for the N-Reactor physics tests. A lubrication process which appeared to produce the most satisfactory results consisted of applying a flash coating of copper to the surface of the aluminum billet, heating the billet to 400 C, and applying Fiske 604 commercial lubricant to the billet just prior to extruding using unheated tooling.

Further work is planned to evaluate the effect of die condition, billet and tooling temperature and extrusion speed on this aluminum extrusion technique. Also, a series of coupons will be autoclaved to determine effect on the corrosion properties of the aluminum after extruding with the copper flash coating. These tests will determine if the usual cleaning methods will effectively remove the copper on the surface.

Heat Treatment of Uranium-Carbon Alloys. Decarbonization of uranium in the solid state by heating in the gamma phase is under investigation. Ingot uranium specimens containing approximately 650 ppm carbon were heated to temperatures in the gamma range (800-1100 C) for periods of one to nine hours. Carbon losses of 100-200 ppm were observed as a result of this treatment. Metallography of the specimens is now under study.

Studies on the Release of Tritium from Aluminum-Lithium Alloys. The second capsule to study tritium outgassing rates from aluminum-lithium alloys is presently being irradiated in K-East Reactor. The capsule is operating at 275 C (527 F) on reactor heat only. The remaining in-reactor capsule heater has been operated at 1200 watts for approximately one hour without burnout. It will be used to raise the temperature to 300 C (572 F), and possibly higher to determine if significant tritium is released.

2. Corrosion and Water Quality Studies

Comparison of High and Normal Alum in Process Water. Initial corrosion data were obtained from a half-plant test at F-Reactor which showed

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higher corrosion of aluminum in process water treated with high alum (18 ppm) than with normal dosage (6-15 ppm). Penetrations of coupons exposed at pH 6.6 and 1.8 ppm dichromate for 22 days at the downstream end of production channels were:

Alloy	High Alum	Normal Alum
1100 Al	1.25 mil	1.0 mil
6061 Al	1.6	1.3
X-8001 Al	0.95	0.90
Carbon steel	0.23	0.23

Comparison of pH 6.6 and 7.0 in Process Water. Data from a half-plant test at D-Reactor show lower corrosion rates of aluminum at pH 6.6 than 7.0, but higher rates for carbon steel. Penetrations of coupons exposed for 100 days at the downstream end of production channels were:

Alloy	pH 6.6	pH 7.0
1100 Al	1.49 mil	2.49 mil
6061 Al	2.00	2.42
X-8001 Al	1.24	1.95
Carbon steel	0.16	0.10

Laboratory Evaluation of Inhibitors. Sodium silicate in concentrations of 10 to 200 ppm SiO_2 did not inhibit corrosion of carbon steel in 34-day static tests at room temperature. Sodium nitrite and sodium dichromate at concentrations of 100 ppm reduced corrosion of carbon steel by factors of 5 to 10. These tests are continuing to longer exposure.

Evaluation of NH_4OH for pH Control. In-reactor operation of KER-1 was resumed at pH 10 with ammoniated coolant following an extended reactor shutdown. Coolant recirculation was continued during the shutdown period to maintain the adherent magnetite film. The process tube was flushed with ammoniated coolant to remove the process water before in-reactor operation was resumed. These operations prevented recurrence of the crud release problems previously encountered during startup after extended shutdown periods.

The coolant quality was satisfactory during this entire operating period. In general, the ammonia concentration was maintained at 12-15 ppm. The loop operated at a feed-and-bleed rate of 1 gpm, and it was necessary to add approximately 0.2 pound of ammonia per day to maintain the desired concentration. The radiolysis of ammonia was

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about 0.03 pound per day based on ammonia concentration measurements. This rate is in excellent agreement with that based on measurement of dissolved H_2 and N_2. Total dissolved gas in the coolant was approximately 8 cm^3/liter water (STP), and the H_2/N_2 ratio was approximately 2.

None of the coolant samples obtained were visibly turbid, and total solids concentrations were low (0.5-3.0 ppm). pH control was within 10.1 ± 0.2 except on two occasions, when temporary mechanical problems were encountered with the chemical injection system.

The two instrumented crud detectors continue to function well. No crud buildup has been detected after 840 effective full power hours.

Crud was stripped from fuel elements previously exposed in KER-1 to pH 10 NH_4OH. Two different decrudding solutions were used, a hydrazine-ammonia-versene solution and a sulfuric oxalic acid solution; the latter was more effective. Both the solutions and solid samples of the crud will be analyzed to determine total crud and its chemical and radiochemical content.

Modification was completed of the chemical-injection, bleed-control, and sampling systems of KER-4 to permit conversion from lithiated to ammoniated coolant. Testing of ammoniated coolant will begin in December.

Activity Monitoring in KER Loops. During the past six months activity levels have been routinely monitored in KER-1, a carbon-steel loop operated with ammoniated coolant, and KER-2, a stainless-steel loop operated with lithiated coolant. The activity in KER-2 is two to five times that in KER-1. The main sources of activity are the pump strainer and blowdown, the main flow control valve, the mockup tube inlet and outlet, and the by-pass control valve on the primary heat exchanger. The activity generally builds up to a peak 24 to 30 hours after reactor startup, and then drops to an equilibrium value reached in three to four days.

Crevice Corrosion of Zirconium Under Heat Transfer. A second test of corrosion under N-Reactor fuel supports was completed in TF-3 using an electrically-heated sample. Test conditions were 314 C bulk temperature, 336 surface temperature, 200,000 Btu/hr-ft^2 heat flux, pH 10. Boiling was present in the crevice under the supports. The pH was maintained with NH_4OH for 75 days, and with LiOH for 28 days as the result of inadvertent use of Li-form ion-exchange resin. Corrosion of the Zircaloy-2 up to 1 mil was found in all crevices under six support welds. Much more severe corrosion (up to 7 mils) had been

encountered in a previous test under the same conditions with LiOH. This may indicate that corrosion in crevices with boiling can be eliminated or greatly reduced by use of NH_4OH rather than LiOH. The test is being repeated with NH_4OH.

Corrosion of N-Reactor Components. Samples of silver-plated N-Reactor retaining rings and seal rings, Type 304-L brazed to carbon steel with Easyflow 45 silver alloy, pure copper, and pure silver were tested in TF-2 for two weeks at 550 F with pH 10 NH_4OH. The silver plating was completely corroded away when exposed to high velocity water and was substantially corroded when exposed at semi-stagnant conditions. The copper and brazed samples were also attacked, with more corrosion at the high velocity.

A section of intergranularly attacked tubing from N-Reactor steam generator 4A was tested in TF-2 at 550 F, 1700 psi, pH 10 NH_4OH to determine the rate of propagation of a pinhole leak. After three days a crack one-inch long suddenly developed $3\frac{1}{2}$ inches from the pinhole. Two other samples being exposed at the same conditions developed cracks $\frac{1}{4}$ inch long at the pinhole immediately upon exposure. All the samples had previously been exposed at 300 C, 3000 psi, until the pinholes had developed.

An N-Reactor rolled-joint test section was installed in TF-1 for testing of the corrosive effects of decontaminants in the crevice between the carbon steel and zirconium.

A corrosion test of two N-Reactor tube rupture monitors was completed in which they were subjected to ten decontamination cycles of alkaline-permanganate and sulfamic acid. Corrosion results are not available. Three other monitors have received 14 of 20 decontamination cycles.

Rupture Studies of N-Fuel Elements. One N-Reactor outer fuel element, irradiated to about 150% goal exposure at 262 C water temperature, was rupture tested in the IRP. The fuel element was predefected with a 0.025-inch hole in the side at the center of the element. After a 95-minute incubation period at 300 C, the fuel element began "rupturing". The latest N-Reactor normal shutdown schedule for cooling water temperature was followed. Examination of the fuel element showed a ruptured area about two inches in diameter with several small raised and torn areas. While attempting to weigh the element, it broke into two pieces, with a resulting clean break through the uranium and Zircaloy-2 cladding. Weight loss of the element was 25.4 grams.

Analytical Chemistry - Oxygen Analysis. A modified Winkler procedure was developed to eliminate the interference from reducing agents in

dissolved O_2 analysis. The modification consists of adding excess I_2 to react with the reducing agents, determination of residual I_2, and then the standard Winkler procedure. The residual I_2 is subtracted from the total I_2 obtained after the Winkler analysis to determine the O_2 concentration. Preliminary results indicate that the procedure is satisfactory, even in the presence of relatively high concentrations of N_2H_4 or Na_2SO_3 (50-200 ppm).

Corrosion Capsule Studies. In-reactor test facility 2A at 105 KE was changed from a gas-filled to a water-filled test hole, necessitating capsule design modifications. To study the heat transfer characteristics of the proposed zirconium corrosion capsules, a simulated in-reactor test hole was set up. Relationships between heater power and specimen temperature were determined for a wide variety of heat transfer conditions.

3. Gas-Atmosphere Studies

Graphite Burnout Monitoring. Small burnout monitors exposed from 5/1/63 to 11/5/63 in channel 3478 of D-Reactor showed a burnout profile having a sharp peak of about 4% per 1000 operating days (%/KOD) at 80 inches into the stack.

This peak is greatly reduced from the maximum of 22%/KOD measured during the preceding shorter test period from 1/3/63 to 5/1/63.

Graphite-Zirconium Compatibility in N-Reactor. The reaction:

$$CO + H_2O \longrightarrow CO_2 + H_2 \qquad (1)$$

was studied in a gamma radiation field to determine the g value for hydrogen production (molecules produced per 100 ev energy absorbed). Application of this information to the N-Reactor atmosphere computer program requires:

1. Gamma dose rate and distribution in N-Reactor.

2. Efficiency of energy transfer from activated helium to carbon monoxide and water in promoting the reaction.

Based on the specific tube power in N-Reactor, the gamma dose rate has been estimated to have a maximum of about 1.8×10^8 R/hr and is expected to follow a cosine distribution front to rear.

The efficiency with which the energy absorbed in the helium gas is transferred to the reacting species has not been studied. However,

when argon is present in concentrations comparable to those of CO and H_2O the major portion of the energy absorbed in the argon leads to reaction. The energy transfer has been studied in the range from 36 to 87 vol% argon. Linear extrapolation of these data to near 100% argon yields a maximum value for the efficiency of transfer of about 0.5. This value is used in calculations showing effect of this radiation-induced reaction on the operating range for input water level shown in the accompanying table. It should be emphasized that these calculations do not include any effect of radiation on the reverse reaction:

$$CO_2 + H_2 \longrightarrow CO + H_2O \qquad (2)$$

The table lists for some particular input compositions of hydrogen, carbon monoxide, and carbon dioxide, the maximum permissible input water (WHI), the minimum permissible input water (WLO), and the range (WHI-WLO). The high and low limits are based respectively on maximum graphite burnout of 2% per thousand operating days and sufficient water present to prevent zirconium hydriding.

N-Reactor Atmosphere Computer Results

Inlet Composition, mm Hg				Water Vapor, mm Hg		
pH_2	pCO_2	pCO	Dose Rate, R/hr	WHI	WLO	Range
0.076	0.076	0.076	0.0	.469	.317	.152
			1.8×10^8	.521	.371	.150
			1.0×10^9	.559	.402	.157
0.76	0.76	0.76	0.0	.483	.313	.170
			1.8×10^8	.532	.367	.165
			1.0×10^9	.768	.615	.153
7.6	7.6	7.6	0.0	.666	.298	.368
			1.8×10^8	.718	.352	.366
			1.0×10^9	.955	.599	.356

The results given above show that the effect of irradiation on the range of safe water vapor operation is negligible; however, the location of this safe operating range is markedly influenced. The effect of reactor radiation on gas reactions cannot, therefore, be considered inconsequential.

4. Process Tube Development

Hydriding of Process Tubes. Gas phase hydriding at 400 C (750 F) has been performed on two N-Reactor process tube burst specimens. The hydrogen was confined to a strip $1\frac{1}{2}$ inches wide, extending the full length of each piece. One specimen absorbed about 500 ppm hydrogen and the other about 1150 ppm. Metallography showed a uniform concentration of about 250 ppm throughout the wall thickness of each. Both had a case of massive hydride at the outer surface where the hydrogen had entered the specimen. The case was thicker and harder in the specimen with the higher total hydrogen content.

A brittle fracture test was conducted at room temperature on the specimen with the nominal 500 ppm by firing a projectile into its wall while it was subjected to an internal pressure of 5000 psig. The specimen failed full length with a brittle fracture. The effect of the hydrogen can be estimated by comparing this pressure with the 19,000 psig necessary to cause a similar failure in a nonhydrided piece.

A similar test was performed on the specimen containing the nominal 1150 ppm hydrogen. It was fired at a pressure of 500 psig and again at 1650 psig. In neither case did the specimen fail, but large areas of the extremely brittle case spalled off around the projectile impact areas. An alternative test method was then applied in which a one-inch slot was milled through the tube wall. This slot was then patched and the specimen pressurized to failure. It fractured full length at a pressure of 2230 psig. This compares with 13,000 psig for a similar nonhydrided specimen.

These tests demonstrate that hydrogen in these amounts reduces the critical pressure for brittle crack propagation to one-fourth to one-sixth that for nonhydrided Zircaloy-2. It is not yet clear how much of the effect is attributable to the massive hydride case and how much to the underlying uniform concentration.

N-Reactor Steam Generator Tubing. The 3000 psi 300 C (572 F) stress rupture tests on steam generator tubing containing flaws through greater than 80% of the tube wall are continuing. Of the original 25 tubes, 13 failed in less than 500 hours and the remaining 12 tubes have undergone 1331 hours of testing. A test of material of the same flaw depth was started at conditions of 4500 psi, 300 C. These pieces all failed between 330 and 340 hours of test with typical pinhole failures.

5. Thermal Hydraulic Studies

N-Reactor Studies. Laboratory experiments were continued with a full scale electrically-heated model of the downstream half of an N-Reactor fuel column and typical outlet piping and fittings. Heat transfer conditions were determined at power levels equivalent to 124% of the maximum design tube power for the reactor. These experiments were conducted to extend the range of investigation to high powers of possible future interest, and to check the validity of extrapolating pressure-drop-flow relationships developed from the large quantity of data taken previously at lower powers.

In these experiments the power of the electrically-heated model was 3100 kw. Coolant conditions in the outlet riser ranged from 5°F subcooled to 7% quality, at a pressure of 1200 psig.

Plugging of Single Channels in N-Fuel Elements. Forty-four sets of experimental measurements were made in a series of single channel plugging experiments. These experiments were also conducted with the full scale electrically-heated model and piping of the downstream half of an N-Reactor fuel column.

The purpose of these plugging experiments was to determine redistributions of heat and flow among the three parallel coolant channels in an N-Reactor fuel column and the changes in total tube flow which would result from various degrees of plugging in any channel.

Measurements were made with the fuel column model at powers of 0 and 2500 kw. The 2500-kw data will provide information applicable to a reactor tube operating at 5000 kw. The zero power data will provide reference points for analysis of the "at-power" data to determine the contribution of single channel boiling to the flow redistributions.

Flow to each of the channels was reduced by closing valves in individual inlet lines. In the zero-power experiments, flow to each channel was reduced in steps until the channel was valved off completely. In the 2500-kw experiments, flows were reduced in a similar manner with the flow reduction continued to a value calculated to be very close to burnout conditions at some point in the heated section. At each step, measurements were made of pressure drops, flow rates, and inlet and outlet coolant temperatures for each channel.

Data from these experiments have not yet been fully analyzed. Writing of a digital computer program to handle data from these and the two-phase pressure drop experiments is in progress.

[redacted]

Experience with the electrically-heated fuel model tends to confirm the applicability of the burnout experiments with short test sections to larger, more complicated sections. In both the two-phase pressure drop experiments and the plugging experiments, conditions very close to those which produced film boiling in short "burnout" sections were obtained. In one case a temperature rise indicative of film boiling occurred almost exactly at the predicted burnout conditions. In no cases were indications of film boiling noted with heat fluxes below those predicted to produce burnout.

Present Reactor Studies. Laboratory heat transfer experiments were started to determine the detailed thermal hydraulic characteristics of K-V I&E fuel elements in zirconium process tubes in the K-Reactors.

The experimental apparatus consists of a 1.520-inch OD x 0.420-inch ID electrically-heated rod placed in a 1.724-inch ID smooth-bore tube approximately 28 feet long. This gives a nominal annulus of 0.102-inch. Ceramic spacers 0.086-inch high were used to represent the self-supports on the K-V fuel elements. When positioned in the tube in the "worst" position (spacers straddle a vertical plane), the minimum and maximum annuli widths between the heated "fuel" and the process tube are 0.081-inch and 0.123-inch, respectively. This is being run as the normal case, although it corresponds to an eccentricity of 20.6%.

Twenty-eight steady state runs determining pressure-drop-flow relations have been completed at constant tube powers of 700 kw and 1000 kw (nominal). During this "first stage" of the steady state experiments, only those flow rates which are sufficiently low to check the validity of the present method of establishing outlet water temperature limits are being investigated. After "first stage" experiments have been completed for higher tube powers, the experiments will be extended to lower flow rates to investigate steady state film boiling conditions.

Although the data to date are very preliminary, two items of interest might be mentioned. First, the data for pressure drop from tube inlet to end-of-active-charge show a "W" shape near the minimum point in the pressure drop-flow curve as was reported for K-Reactor experiments with normal I&E fuel in a ribbed tube (HW-74669). The preliminary data indicate that this condition results from increased pressure drop due to "local boiling" in the tube before two-phase pressure drop in the outlet fittings becomes dominant.

Second, circumferential water temperature measurements, taken at the end-of-active-charge (EOAC) and five feet upstream from the EOAC,

indicate that appreciable fluid mixing occurs in the annulus along the last few feet of fuel charge. This indicates that circumferential water temperature measurements made only at the end of the fuel charge could be misleading when used to infer water temperature differences between minimum and maximum width annuli at other axial locations.

As an example, in one run without bulk boiling, the R value, defined as the ratio of maximum temperature rise through the tube at some point in the annulus to the average temperature rise of the coolant flowing through the tube

$$\frac{\Delta T \text{ max}}{\Delta T \text{ bulk}}$$

was 1.15 at the end of the charge but was 1.28 at five feet upstream. Boiling predictions or corrosion rate predictions based on end-of-active-charge R values could be quite misleading. These results are preliminary and further experimentation might cause modification of these findings.

6. Shielding Studies

N-Reactor Shield Evaluation. Preparations for the N-Reactor shield tests have consisted of (1) calculations and experiments to check out the feasibility of the proposed N-Reactor experiments, (2) design of hardware to allow placement of detectors in the shield and reflector and to preserve the gas seal of the reactor, and (3) procurement and check out of instrumentation.

Previous calculations of Peterson and Simpson were normalized to the N-Reactor power levels expected for the zero power physics and power ascension tests, providing calculated neutron and gamma ray fluxes through the reflector and shield as a function of reactor power.

To supplement the calculated information, an experiment was run in the PCTR in which gold, sulphur, iron, titanium, nickel and cobalt foils were placed along with some neutron spectrometer equipment. Count rates and activation levels indicated that the gamma ray ion chambers, the neutron spectrometer, and the gold and sulphur foils can be used in the reflector during zero power tests. During power ascension tests the ion chambers and all foils may be used in the reflector; the spectrometer, gold and sulphur foils may be used in the shield. At full power, ion chambers and foil measurements will be used in the shields.

Design of the reflector thimble is complete and has been approved by N-Reactor staff. Fabrication of the thimble, the installation tools, the graphite foil holders and the graphite thimble components are being initiated. The thermocouple system is complete and ready for installation.

The ion chamber system from the 105-DR shield wells was not removed at the last reactor shutdown, hence procurement of another system has been initiated. Several suitable chambers were located and borrowed from IPD, together with an electrometer. Additional chamber cases and cables will be fabricated. Work on the neutron spectrometer systems is continuing; however, satisfactory performance has not yet been obtained.

MAC 31 and 15 Group Cross Section and Group Constant Libraries. The document which presents the two cross section libraries for MAC Code has been completed in draft form to be submitted for publication. NASA, Lewis Research Center, requested MAC 31 group cross section data for zirconium and beryllium. These data were furnished, in ready input form, from the recently completed 31 group compilation.

7. Graphite Studies

N-Reactor Graphite Irradiations. Irradiation of the two third-generation capsules, H-5-3 and H-6-3, in the series of long term irradiations of N-Reactor graphite is progressing satisfactorily. All thermocouples in both capsules are functioning properly.

Oxygen-Bomb Calorimetry. The oxygen-bomb calorimeter, for total stored energy measurement, has arrived and is being calibrated.

A stirrer assembly, to replace that supplied by the manufacturer, was designed and is being built. The bucket and lid were modified for ease of handling. For convenience in use, tare weights were made for the bucket and its contents. The Mueller Bridge and the platinum-resistance thermometer were calibrated. The jacket-temperature regulator met manufacturer's specifications.

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 Fabrication. Five swaged 19-rod elements were completed and charged into the PRTR. These elements contained impacted UO_2- 1 wt% PuO_2 fuel cores. UO_2-1 wt% PuO_2 powder was impacted and processed to provide fuel for an additional four clusters. Existing equipment is being modified to permit vacuum outgassing of this fuel at 625 C and 10^{-3} torr for one hour.

Five reclaimed UO_2-0.5 wt% PuO_2 elements were also completed for the PRTR.

PRTR Fuel Element Cladding Evaluation. To assure that the tubing now being used for fabrication of swaged fuel rods is of the best possible integrity, it is being retested using techniques not available when it was originally purchased. The tubing is being tested with ultrasonic equipment to a defect level of 5% of the wall thickness, i.e., 1.5 mils. After solving instrumentation problems inspection is now proceeding at the rate of 75-100 tubes/shift. A few defects have been found in tubing approved on the basis of earlier testing.

Post-Irradiation Examination of Failed PRTR Fuel Elements. A defective swage-compacted Mark I element (FE-5307) containing impacted UO_2- 1 wt% PuO_2 fuel was discharged from the PRTR after an exposure of 280 MWD/T. Post-irradiation examination revealed that two of six rods in the element failed. These, and four other nonfailed rods, contained fuel from the same blend (batches 64-67). Circumferential cracks developed in the cladding of the two rods permitted the escape of fission gases. Insignificant swelling of the hydrided cladding occurred in the failure regions. No fuel was washed out of the rods. During disassembly of the fuel elements in the PRTR storage basin and in the hot cell, both failed rods broke in areas removed from the points of original cladding penetration. The cladding was locally embrittled and massively hydrided.

Four nonfailed fuel rods (two containing the suspect fuel blend, batches 64-67), and two containing other fuel blends were split lengthwise. No evidence of internal cladding corrosion or abnormal concentrations of zirconium hydride was found. The tightly compacted, swaged fuel was removed with difficulty from the cladding halves. However, circumferential rings of fuel adhered to the inner cladding surface of some of the rods. There was no evidence of localized fuel-clad reaction or of increased zirconium hydride concentration adjacent to the adhered

fuel. Carbon content of the adhered fuel was approximately 500 ppm, compared with approximately 100 ppm in the bulk fuel. The high carbon content, plus the appearance of cementing, suggests radiation polymerization of an organic contaminant. Infrared absorption analyses of carbon tetrachloride extractions of archive fuel specimens revealed traces of the same pattern observed with oil from the gear box of a pulverizer used in the fuel processing system.

Post-irradiation examinations of both failed and nonfailed vibrationally compacted PRTR fuel rods support the hypothesis that failure is caused by rapid, local gas-phase hydriding of the inner surface of the Zircaloy cladding.

Zirconium hydride in the absence of zirconium oxide in the failure regions has been identified by x-ray diffraction. Also, hydrogen has been found in the gas collected from nonfailed rods that suffered internal hydriding in the susceptible end-cap crevice regions. Gas-phase hydriding will occur rapidly at PRTR cladding temperatures in the absence of fluoride; however, fluoride contamination would accelerate the attack. If it is assumed that water contamination in the fuel is the only source of hydrogen, acceptable limits are difficult to establish because of the variable area of attack.

Fuel Element Refurbishing. PRTR fuel elements (5117 and 5119), which had broken rod wire wraps repaired in October, were examined in the PRTR basin. Both elements appeared to be in a satisfactory condition after irradiation during October and November. It was recommended that they be recharged.

Fuel Element Rejuvenation. The second cycle rejuvenation fuel element was re-enriched by the addition of 12 g of UO_2 (0.9% U-235) and returned to the MTR for a third cycle of irradiation. This enrichment is expected to enable approximately the same heat flux as was generated during the second irradiation cycle.

Re-enrichment for the second time was accomplished by machining off one end cap and boring a 1/8-inch diameter axial hole through the fuel rod. Only minor drilling, at each end, was required because of radial relocations of the enriched UO_2 during the previous irradiation. This left an integral center void which extended throughout at least 90% of the fuel length.

After re-enrichment by compacting UO_2 in the center void, a new end cap was welded on the fuel rod, the fuel rod was decontaminated and returned to the MTR for re-irradiation.

Nupac Process (High Energy, Pneumatic Mechanical Impact Densification of Fuel). A glove box was designed, fabricated and installed to house a large billet heating furnace. Seven impaction cans, each holding about 15 pounds of oxide, can be heated in a progressive manner to increase the fuel impaction rate. A mechanical manipulation device to handle the cans remotely has been designed and fabricated.

Fuel Element Development. An eddy current sensing coil was incorporated into a vibrationally compacted UO_2 fuel rod for fretting corrosion studies. The coil was encapsulated in a ceramic and connected to lead wires swaged into a 1/16-inch stainless steel sheath. The coil was then surrounded with ZrO_2 to act as a thermal barrier during irradiations. Radiographs showed no mixing of the ZrO_2 and UO_2 at the bottom of the coil. Although some mixing occurred above the coil, it was not particularly serious. This will be minimized on the final model by using a Zr-2 plate to separate the layers. Electrical measurements showed no appreciable change during assembly and loading. Twenty similar coils are being fabricated for assembly into a Mark I PRTR fuel element in mid-December.

Desorption of Water from Fuel Materials. The characteristics of the desorption of water from UO_2, PuO_2, and UO_2-PuO_2 were investigated by using a high temperature (0-950 C), coulometric electrolysis apparatus developed in cooperation with Analytical Laboratories personnel. Desorption of water from these materials occurs by at least two different temperature dependent mechanisms.

From UO_2, part of the adsorbed water is released at 30 C and the remainder is released at approximately 200 C. From PuO_2, adsorbed water is also released at 30 C, but the remainder comes off partly at about 300 C and again at approximately 500 C. About 25% of the water content is released above 200 C. Analysis of the water content of impacted UO_2-1 wt% PuO_2 indicated that the desorption curve follows the PuO_2 pattern rather than the UO_2 pattern.

In the conventional water analysis method for UO_2 and UO_2-PuO_2 solids, the sample is heated to about 200 C. Thus, the water content of PuO_2 and UO_2-PuO_2, as previously measured by the conventional method, has been lower than the real water content. A high temperature (20-850 C) water analyzer apparatus is being built for routine measurements of total water content of PuO_2 and UO_2-PuO_2.

A program was started to provide extremely dry atmospheres in selected glove boxes used for ceramic fuel fabrication. A goal to attain atmospheres with a water content of less than 5 ppm has been established.

Carbonaceous Impurities in PRTR Fuel. Organic compounds must be excluded from nuclear fuel since they are a source of hydrogen that may lead to hydriding of the fuel element cladding during irradiation. There is also a possibility that excessive carbon in the fuel may enter into an alternate reaction with moisture impurities to release hydrogen. The fuel preparation equipment was carefully examined for possible oil or carbon sources.

Preliminary results from carbon analyses and infrared absorption measurements of carbon tetrachloride extractions of UO_2 and UO_2-PuO_2 powders show pickup of carbon and organic material during processing. Infrared absorption data indicate that an organic (C-H) is present at levels of 5 to 300 ppm in crushed UO_2 and UO_2-PuO_2 powders.

A specimen of UO_2-PuO_2 powder, removed from a jaw crusher that developed a minor oil leak, was analyzed. Carbon content was found to be about 3000 ppm versus $\sim$ 200 ppm carbon content of the input UO_2 and PuO_2 materials. Carbon content of other recently crushed UO_2-PuO_2 powders (-325 mesh) showed about 600 ppm carbon versus 200 ppm carbon content of input UO_2 and PuO_2.

Evaluation of Fuel Impurity Effects. Eighteen prototypic PRTR fuel capsules containing UO_2-1 wt% PuO_2 contaminated with known quantities of water and oil were fabricated and shipped to the MTR for irradiation. Nine of the capsules contain fuel contaminated with hydrocarbons (> 500 ppm C) while the remainder contain standard process material from the PRTR fuel fabrication line. Fuel in three capsules of each set of nine was treated to reduce the water content to less than 10 ppm (gear oil was added to one group of three to replace hydrocarbons destroyed or removed during drying). Fuel in three other capsules from each group of nine contains 60-90 ppm water, which is typical of PRTR fuel processed in the recent past. The water content of the remaining six capsules was adjusted to approximately 300 ppm. These irradiation tests will provide needed information on the effects of water and hydrocarbon impurities on Zircaloy cladding failure rate.

Special Test Fuel Elements. Fabrication of UO_2-PuO_2 pellets for PCTR test elements is in progress. Three thousand pellets were circumferentially ground and 1000 were end ground.

Welding fixtures were designed and are being fabricated for completing end closures on four Zr clad and six Al clad, plutonium-bearing, N-Reactor physics test elements.

Irradiation of Uranium-Plutonium Oxide. Evaluation of previously reported, irradiated UO_2-PuO_2 pellets led to the following conclusions for 90% and 65% TD pellets containing to 7.45 mole % PuO_2. Exposures were 0.14 to 4.15 x 10^{20} fissions/cm^3, at 214-1160 w/cm rod power:

(1) A ring of voids (in some cases associated with white, metal-like inclusions) coincides with the intensely radioactive portion of the columnar grain region characterized by high Ru-106 and low Cs-137 concentrations.

(2) The lowest Ru-106 concentration is often associated with the outer limit of the subgrain structure of the large columnar grain region.

(3) Significant concentration gradients of Zr-Nb-95, Cs-137, Sr-90 and Ru-106 are sometimes found within 0.03 cm (0.01 inch) of the fuel surface. Otherwise, permanent migration of Pu, Zr-Nb-95, Ce-Pr-144, and Sr-90 is insignificant.

Short Duration Irradiation of UO_2-PuO_2. Successful functional tests were conducted in the MTR, VH-4 facility with the improved version of the "ICARUS" capsule transport vehicle. The modified connector pin successfully prevented recoupling of the vehicle and fuel specimen after release.

Irradiation Testing Program for EBWR-Type Fuel. Capsules GEH 14-421 and -424 were recharged into the ETR for cycle 59 for irradiation to 1.43 x 10^{20} fissions/cm^3 (5000 MWD/T fuel). The capsules contain UDO_2-2.5 wt% PuO_2.

Plutonium Handbook. The declassified, revised version of Chapter 2 of Section VI of the Plutonium Handbook, to be submitted for presentation at the 1964 Geneva Conference was reviewed by W. D. Stump, Dow Chemical Company, Denver, Colorado.

Fuel Re-use Study. A liquid metal loop facility to enable engineering evaluations of a fuel re-use concept is being designed. The concept consists of irradiating depleted fuel elements in the breeding blanket of a sodium-cooled fast reactor, and using the (plutonium) enriched fuel elements as fuel in a thermal reactor.

The liquid sodium facility in combination with an existing water loop will be used to investigate the effect of transferring fuel

elements from liquid sodium environment to water environment. Possible sensitization of the cladding to aqueous corrosion or problems arising from incomplete removal of sodium on the fuel elements are typical considerations.

A schematic flow and instrumentation diagram and a tentative component and piping layout of the liquid sodium loop have been prepared. Thermal stresses of the system which may operate up to 650 C are being computed on the IBM 7090.

Corrosion and Water Quality Studies

Iron and Nickel Base Alloys in 550 C Steam. Uniform corrosion results are being obtained for various stainless steel and nickel base alloys in 550 C (1022 F) deoxygenated steam at 5000 psi and at atmospheric pressure. There appears to be no pressure effect on the uniform corrosion rate of the alloys for either test condition, nor at previous test conditions of 550 C - 3000 psi steam. Following 72 days of exposure in 550 C - 5000 psi deoxygenated steam, the various alloys may be grouped in the following order of increasing corrosion:

Group I	(8.0 to 9.4 mg/dm^2)	- Hastelloy X, Incoloy and Inconel X.
Group II	(12.0 to 47 mg/dm^2)	- Hastelloy N, 446 SS, and 17-4 PH SS.
Group III	(57 to 68 mg/dm^2)	- An experimental 25% Cr, 3% Al, 0.5% Y - Fe alloy, Nichrome V, R-27.
Group IV	(144 to 248 mg/dm^2)	- 304L SS, two heats of 406 SS, 316L SS, and 430 SS.

The corrosion results are virtually the same as those previously obtained for these alloys in 550 C - 3000 psi steam for the same length of time. The weight gains at 3000 psi ranged from 7 mg/dm^2 to 269 mg/dm^2 compared to a range of 8 mg/dm^2 to 248 mg/dm^2 for the 5000 psi test. For individual alloys, there was some shifting between groups. Three alloys -- Hastelloy N, R-27, and PDRL-102 -- have shifted one group lower, while 446 SS and Nichrome V have been shifted up one group.

Following 35 days in 550 C (1022 F) atmospheric pressure steam, the weight gains range from 6 mg/dm^2 to 201 mg/dm^2 with virtually the

same alloy grouping as in 3000 psi steam. If longer exposures confirm the pressure dependence of the corrosion rates, considerable simplification in further testing becomes possible.

Zirconium Concentration in PRTR. Routine analysis of the PRTR primary coolant was continued. Zirconium concentration was 0-1 ppb, which is considered normal. It is therefore unlikely that significant fretting of zirconium occurred during the month.

Fretting Studies. A PRTR UO_2 fuel element with 1/16-inch wide supports was tested in TF-7 at 530 F, 125 gpm, and an induced vibration of 25 cps. Fretting of one mil was found on both the Zr-2 supports and the Zr-2 process tube after one month. Previous tests had shown substantially less fretting under the same conditions when supports with larger contact area were used.

A traverse of a PRTR Zr-2 wire wrap showed that hydrogen content increased from 50 ppm in nonfretted regions to 200-300 ppm where fretting had occurred. On a 360° ring support, 2-3 times as much hydride was found in a region where the oxide had been burnished off by fretting (but where no measurable penetration occurred) as in nonburnished areas.

A Magne-Dash autoclave was installed in the PRTR primary system. The first of a series of tests was started to obtain data on fretting of Zr-2 exposed to PRTR coolant.

Decontamination. A modified oxalic-peroxide-peracetic solution was successfully used to remove gross contamination and mixed uranium oxides following a fuel element rupture in the IRP Loop. The activity of the filter was reduced from 260 to 80 mr/hr; that of the test section from 19 to 6 mr/hr.

Reactor Components Development

PRTR Pressure Tubes. A paper entitled "In-Reactor Monitoring of the Zircaloy-2 PRTR Pressure Tubes" (HW-SA-3008) was prepared for presentation at the American Nuclear Society semi-annual meeting. This paper describes the equipment developed for in-reactor monitoring and gives the results found from inspections made to date.

A specimen from the eleventh Zircaloy-2 PRTR pressure tube to be destructively examined burst in a ductile fashion at room temperature and 9200 psig. The burst was a through crack, initiated at a 16-mil deep ID pit, which allowed almost instantaneous

depressurization from the burst pressure. With the use of an "elastic" patch over the ID side of this crack, the tube specimen was again burst tested at room temperature. It took a pressure of 8800 psig to cause brittle propagation of this starting crack.

Pressure Tube Monitoring. Eight process tubes were inspected this month. Included in this group was the new pressure tube installed in the rupture loop (P.C. 1946) after the initial 24-hour test run. This initial test run was made with a UO_2 type fuel element. During this test run, six very shallow fretting corrosion marks were formed. The upper and lower fuel element end bracket spacers each produced one mark. Three marks were associated with a single fuel rod spiral wire wrap. All appeared to be superficial. New fretted areas were found in the other seven tubes inspected. In nearly all cases the incidence of new fretted areas appeared on the average to be less than for the reactor as a whole. At this time there is no known reason for these differences.

Mark III monitoring equipment is now essentially complete. Part of the equipment has been moved to PRTR for installation by PRTRO.

The M-2 borescope equipment has been overhauled and new air and electrical lines were installed. Inspection of the equipment indicated that a slight amount of radiation darkening of the lens elements has occurred. These lens elements have now sustained an estimated radiation dosage of about 10^{10} Roentgen.

Shim Rod Development. The electrical drive motors and transmissions for both developmental rods have been assembled and operated satisfactorily. The selsyn readout (to 0.01 inch) position indicators also operated satisfactorily. The aluminum heat sink castings are unsatisfactory because of a mis-drilled penetration into a cooling passage, and these are being remade. The core of the first iron-shot filled shield plug was assembled and has passed the leak test. The zirconium lead screws have been received from the vendor. One rod assembly will have aluminum lead screws, while the other will be assembled with the zirconium lead screws.

Fretting Corrosion Investigation. The fretting corrosion investigation has been continued in two individual areas during November. The first has been concerned with the operation of EDEL-I; the second with monitoring of tube vibrations in the PRTR reactor.

The PRTR pressure tube full-scale model, EDEL-I, is being used to study the fretting corrosion problem. This system essentially

duplicates the operating pressures and temperatures in the PRTR. To maximize testing efficiency in the loop, the time required to produce fretting corrosion to an advanced degree is desired to be a minimum. To this end, four of the six gusset pads on a 19-rod cluster fuel element were sharpened to increase the rate of penetration. The four gusset pads which were sharpened had an edge approximately 5 mils wide. The remaining two gusset pads were left at the original 1/16-inch width.

The EDEL-I was then operated at 530 F and 118 gpm for 123 hours. Following this operating period, the fuel cluster was removed and inspected. The pressure tube was cleaned, dried, and visually inspected with a borescope. The inspection showed two marks, both of which being in the lower end cap area, created by the gusset pads vibrating upon the tube wall. One of these marks was caused by the unsharpened pad, while the other was caused by one of the sharpened pads. A difference in the marks was noted, which was also apparent in supporting photographs. The fretting mark for the unsharpened pad was relatively broader, longer and duller than was the mark for the sharpened pad. No attempt was made to determine the depth of the marks since the borescope used does not have the necessary attachments. Similar borescope inspection of the pressure tube wall prior to this test showed no fretting cor-

The reduction of the PRTR data will not be completed for approximately two months. Preliminary results are reported below which indicate several interesting characteristics of the system:

1. There were no significant variations in amplitude noted on any tube while the primary pumps were maintaining low flow and the rupture loop pump RLP-1A was changed from high to low flow while RLP-1B was switched from low to high flow. All tubes were apparently excited by the same input force.

2. Tube 1946, the rupture loop tube, has larger amplitudes than any other tube in the pump configuration of item (1); however, variations in amplitude on tube 1946 could not be ascertained when the rupture loop pump configuration was changed.

3. When the primary pumps were switched to full flow, the amplitude on all tubes other than 1946 increased by a factor of approximately 4. Tube 1946 indicated an increase in amplitude by a factor less than 2.

4. The amplitude on tube 1946 decreased by a factor of 2 when the primary pump PP-1 was left on high; PP-2 was switched from high to off, and PP-3 was changed from off to high.

5. During the high primary flow when the rupture loop pumps were started, the amplitude on tube 1946 increased by a factor between 2-4.

6. The rupture loop pumps increase the amplitude of variation in the tubes other than 1946 by a factor that is less than 2. This observation was made with PP-1 and PP-2 on high while PP-3 was off.

7. With PP-1 and PP-3 on high there is not a noticeable amplitude increase in the process tubes other than tube 1946 when the rupture loop pumps are switched on.

8. Preliminary information indicates an order of increasing forces imparted to the reactor by various primary pump combinations with and without the rupture loop pumps operating. These results are stated below:

A. Without rupture loop pumps

(1) PP-1 & PP-3 - high
PP-2 - off

(2) PP-1 & PP-2 - high
PP-3 - off

(3) PP-2 & PP-3 - high
PP-1 - off

B. With rupture loop pumps

(1) PP-1 & PP-3 - high
PP-2 - off

(2) PP-2 & PP-3 - high
PP-1 - off

(3) PP-1 & PP-2 - high
PP-3 - off

Design Analysis

PRCF-EBWR Safety Studies - Mixed Oxide Fuels. Additional studies of the time delay in the Doppler temperature coefficient in mixed PuO_2-UO_2 due to heterogeneously distributed PuO_2 have been completed. The conduction model employed in previous studies (HW-78711RD) has been applied specifically to the case of mixed oxide fuel being fabricated for EBWR and PRCF loadings. Autoradiographs of impacted fuel material indicate good uniformity in PuO_2 particle distribution and thermal conductivity measurements lend further validity to the analysis.

Fission fragment heat transport has also been included. It has been determined that approximately 35% of the total heat is deposited directly in the uranium by this process. Assuming the fission fragments to deposit heat isotropically over the surface of a PuO_2 particle, an increase in the size of the prompt heat source results. Thus, for the fuel under consideration, the mean PuO_2 particle size is effectively increased from 22 to about 25 microns. This results in a decrease in the time constant for conduction of heat from PuO_2 to the surrounding UO_2 of 20%.

Consideration has also been given to the transfer of heat by radiation within the fuel. Because of the relatively high

transmission of infrared wave lengths in UO_2, it appears that radiant heat transfer may dominate the other processes during a fast power excursion. Since negligible time delay is associated with this effect, the conduction model is believed to yield a reliably conservative delay time.

A conduction time constant of 1.2 milliseconds was calculated for prototypic EBWR-PRCF fuel. A second case in which the PuO_2 enrichment was assumed to be doubled to 3% was found to have a time constant of 0.47 millisecond. Comparing the first result with attainable periods in the reactors to be employed, it may be concluded that the assumption of instantaneous Doppler feedback in excursion analyses is valid.

Details of this work are presented in HW-78711, "Time Delay in the Doppler Effect Resulting from Inhomogeneities in Mixed PuO_2-UO_2 Fuels." This supercedes HW-78711RD with the same title.

PRTR Moderator Level Control. Document HW-79669, "Moderator Level Control in the Plutonium Recycle Test Reactor," has been completed in rough draft form and is circulating for comments.

Thermal Hydraulic Studies

Boiling Burnout for an EBWR Fuel Element in the PRTR Rupture Facility. Calculations were made for boiling burnout conditions in the PRTR rupture loop when fueled with a type of fuel element designed for irradiation in the EBWR. The fuel will consist of two bundles of rods 0.424-inch OD by 56-3/4 inches over-all length placed end to end. The rods will be in a 5x5 square array on 0.569-inch centers with the corner rods missing. It is desired to operate this fuel element at 1800 kw where the maximum heat flux is calculated to be 580,000 Btu/hr-ft^2.

The boiling burnout data most nearly applicable to this case is that for the PRTR Mark I fuel element. The EBWR fuel element differs from the PRTR Mark I element in that it has a larger rod spacing, has no wire wraps, and the square array fills the cross section less completely than does the circular pattern of the Mark I element. The larger rod spacing should, if anything, increase the boiling burnout safety factor, but this could be counterbalanced by the lack of wire wraps which would decrease coolant mixing. However, limited data from closer spaced bundles (0.050-inch) indicates that the lack of wire wraps has little effect upon boiling burnout.

The open cross-sectional areas of the cornerless 5x5 array are filled by support rods and plates to prevent flow channeling, but the support plates are as close as 0.072-inch from one of the side rods. This must be considered since there is some evidence that a rod facing an unheated surface may have a somewhat lower burnout heat flux than has a similar rod facing a heated surface. In view of the difference between the PRTR Mark I fuel element and the EBWR element, it was decided to use a burnout safety factor of 2.0 rather than the 1.85 which was used for the Mark I application.

Use of the Mark I data and a safety factor of 2.0 shows that the EBWR fuel element may be operated at an 1800-kw tube power at 1200 psia pressure and a flow rate of 123 gpm if the outlet temperature is no more than 550 F. If the flow rate is 171 gpm, which gives the same velocity past the EBWR element as 123 gpm has past the Mark I, the element may be operated up to saturation conditions at the pressure tube outlet.

PRTR Power Levels. A summary of preliminary thermal hydraulics calculations involving a possible increase in power of the PRTR was written. It was concluded that both the liquid phase and boiling convection cooling would be adequate protection following a complete loss of pumping power for a power increase from 70 Mw up to 125 Mw. The present primary system relief capacity was also found to be adequate. The study also showed that decreasing the reactor inlet temperature would provide the most convenient power increase from a thermal hydraulic standpoint.

Computer Code AIH-III. Several improvements were made in the computer code AIH-III which calculates the radial temperature distribution in fuel test capsules considering space and temperature dependent variables. These improvements include the addition of an input data printout routine and a routine to calculate film heat transfer coefficients at flowing liquid-solid interfaces by means of the modified Colburn equation. In previous versions of this program, the film coefficient was required as input data, necessitating a laborious trial and error hand calculation. Special versions of the code were written for the test capsule where a 20% uncertainty factor on the film coefficient is required.

Gas Heat Transfer. A report (HW-79463) was written on the available methods and techniques for the analysis of the heat transfer to gases from typical rotating machinery components. The general characteristics of the hydrodynamic flow fields surrounding heated rotating shafts and disks were examined with particular

attention being given to the critical rotational speeds at which transitions between laminar and turbulent flow occur. A method based upon the analogy between momentum and heat transfer was developed to describe the turbulent heat transfer from rotating disks subjected to a variety of boundary conditions commonly found in practical applications. The heat transfer relations developed analytically were used to augment the empirical relationships found in literature and a collection of recommended equations was assembled.

2. Plutonium Ceramic Fuels Research

Preparation of UN-20 wt% PuN Pellets. Three capsules of UN-20 wt% PuN solid solution pellets are being prepared for irradiation in MTR. The solid solution was obtained by sintering -325 mesh UN and PuN powders five hours at 1800 C in 2/3 atmosphere of N_2. Although densification did not occur, x-ray diffraction analysis showed that the solid solution was obtained with little or no oxidation.

Several UN pellets were sintered in the same cycle. This material reacted with the N_2 atmosphere to form a higher nitride, tentatively identified as U_2N_3 by x-ray diffraction.

Changes in the sintering cycle are expected to produce densified, stoichiometric UN and UN-PuN solid solution pellets.

PuN Sintering Studies. A pellet of PuN was sintered to 97% theoretical density. Sintering was achieved in 76 minutes at 1800 C under 0.7 atmosphere of nitrogen. The pellet contained approximately 5 wt% plutonium oxide, which probably acted as a sintering aid. Total weight loss during the experiment was 0.9%.

High Temperature Furnace. The high temperature Brew furnace has been moved to the 308 Building where it is being equipped to allow passage of 250 ft^3/hr of hydrogen gas through the furnace during heating of plutonium-bearing compounds.

3. Ceramic (Uranium) Fuels Research

UO_2 Fuel Structures Associated with In-Reactor Melting. Final resolution of uncertainties of fuel structure associated with central melting during irradiation of UO_2 fuel elements was achieved by a series of short- and long-term irradiations performed in the MTR and ETR. Relocation of tungsten shot which was randomly distributed throughout the fuel permitted exact identification of

the radial limit of melting. Similar tests, now in progress, involving different irradiation durations and manipulations of the test capsules between irradiations will provide additional information concerning structural changes, thermal conductivity, and the dynamic variation of melt radius.

High Temperature Electrical Conductivity. A new apparatus to measure high temperature electrical and thermal conductivities was put into operation. Initial measurements to 1600 C give an intrinsic activation energy of 2.1 ev and an extrinsic activation energy of 0.29 ev for electrical conductivity of polycrystalline UO_2. These values are in good agreement with other recently published data.

High Temperature Behavior of Uranium Carbide. Several specimens of arc-fused UC crystal that had been stored in air for three years were examined by reflection-electron microscopy during heating in vacuum to 975 C. The initially uniform polished surfaces became etched in a variety of patterns, probably related to the orientation of the several grains revealed in each piece. Another phase, possibly oxide, was evident along the edge of several of the surfaces and at some grain boundaries.

High Temperature Behavior of UO_2-W Cermet. A recently developed high temperature stage makes possible reflection electron microscopy during heating to above 3100 C. Self-resistance heating of a conducting or semi-conducting specimen is utilized. The existing gas reaction device can be used with the stage.

Using the newly developed stage, specimens of impacted 20 vol% UO_2-W cermet (O/U = 2.01) were examined during heating to 3100 C. Rapid vaporization of the UO_2 began below 2000 C. Higher temperatures resulted in marked outlining of tungsten grains comprising the continuous phase, with both etching in the grain boundaries and of a smooth low ridge along the boundaries. A similar ridge appeared around each UO_2 crater.

Replication for Electron Microscopy. Detailed evaluation of the use of polyvinyl chloride for electron microscopy replicas is in progress. Using methods based on those outlined by Padden, and techniques developed earlier in this laboratory using polystyrene, highly satisfactory results were obtained with a variety of specimens. The replication appears readily adaptable to remote application on radioactive specimens; replicas can be decontaminated in hot HF without damage. Replicas of several irradiated specimens being prepared for thermal conductivity studies were

obtained and are being processed for examination.

Development of High Resolution Autoradiography. Improved resolution autoradiographs of irradiated UO_2 fuel rods were obtained using stripping film. Observation at 7X clearly reveals pores which were only partially resolved in previous autoradiographs. The method consists of floating an emulsion onto the sample, exposing for a suitable time, and processing the emulsion while in contact with the sample. After processing, the emulsion is transferred to a glass plate and dried in a gentle stream of cool air. Disadvantages of the stripping film are that it is difficult to manipulate and the UO_2 is stained by exposure to the photographic solutions.

Metallic Inclusions in Irradiated UO_2. Recent studies show that many small pores previously observed in the large grains of irradiated UO_2 fuel may represent metallic inclusions formed at high temperature during irradiation and subsequently removed by ordinary metallographic preparation. This disclosure is significant in the interpretation of the microstructure of irradiated fuel elements.

4. Basic Swelling Program

Irradiation Program. Two controlled temperature general swelling capsules reached their goal exposure and were discharged. Seven irradiated capsules are now being stored in the reactor discharge basin. Four of these capsules contain dilute alloys of U-Fe-Si and U-Fe-Al in addition to high purity uranium and the other three contain high purity uranium with natural, twice natural, and four times natural enrichments, respectively. Two additional capsules were charged into a reactor. These were tandemized so that both could be inserted into a single test hole. Specimens of high purity uranium and dilute alloys of U-Fe-Si and U-Fe-Al were included.

Construction of four additional capsules containing specimens of high purity uranium of various sizes and shapes has continued. These are about 10% completed.

Studies have been initiated to determine ways of increasing the heat transfer of the controlled temperature general swelling capsule without major design changes so that lower temperatures and higher fissioning rates can be examined. These include alternate ways of attaching the present heat transfer fins to the

inner chamber, alternate fin design, and alternate atmospheres between the inner and outer chamber.

Post-irradiation Examination. A replica prepared from a polished and etched cross-section of a tubular uranium fuel element irradiated in a high pressure 1600 psi hot water loop has been given a cursory examination in the electron microscope. The element operated at about 600 C (1112 F) in the center and 400 C (752 F) at the edge resulting in a "normal" looking microstructure in the center and a "swirled, worked" structure at the edge typical of uranium irradiated at these temperatures. No grain boundary tearing was observed in the "swirled" structure, but crystallographically aligned porosity was observed in regions which operated at about 600 C (1112 F) and some general porosity was present in these areas. The aligned pores had the same perpendicular relationship to twin bands as was the case for high purity uranium specimens irradiated in controlled temperature capsules at these same temperatures. Porosity was also evident at the boundary between large, irregular twin bands and the general uranium matrix. Thus, the innate propensity for pure uranium to exhibit grain boundary tearing and crystallographically aligned micro tears has been largely (but not completely) suppressed by either the impurity additions, the bulk of the element, the cladding restraints, or the external pressure restraints. It is interesting to note that the large restraints imposed on this element did not completely eliminate the micro tearing.

A few tiny spherical pores were observed in the edge regions of the cross-section. The central regions contained a large number of pores which were considerably larger. A few isolated, widely scattered areas were present which contained tiny recrystallized grains with porosity at the grain boundaries. The fission gas porosity observed in the center of this element was qualitatively similar to that observed in high purity uranium specimens irradiated under controlled temperature, unrestrained conditions. The effect of pressure is apparently most significant in its suppression of grain boundary tearing and aligned micro tearing. External pressure is certainly not controlling the fission gas porosity and it would further appear that in the temperature range studied, external pressure is not even a contributing factor. Detailed studies conducted under known and controlled conditions are obviously necessary in order to understand the influence of external pressure on the irradiation behavior of uranium.

5. Irradiation Damage to Reactor Metals

Alloy Selection

Materials for the Irradiation Effects on Reactor Structural Materials Program have been received and catalogued under a system designed to maintain control over each item. Several shipments of program materials to participating sites were made during the past month. Tensile specimens of Hastelloy N, Inconel 625, Inconel 702, Inconel 718, R-235 and R-27 from the ETR hot water loop facility are presently being examined. During the past month some specimens were tested, the results of which are being analyzed. Tensile tests at room temperature on specimens of Inconel 625 irradiated at exposures of 5.88×10^{19}, 1.1×10^{20}, 1.8×10^{20}, and 5×10^{20} nvt show only small changes in yield strength, ultimate strength, or percent uniform elongation. One specimen of Inconel 625 irradiated at 80 C (176 F) to an exposure of 1×10^{20} nvt showed a 65% increase in yield strength, indicating a marked effect of irradiation temperature upon the properties of the material.

Post-irradiation testing of Cb-1 Zr, Cb-752, and Ta-10 W is continuing.

In-Reactor Measurement of Mechanical Properties

The in-reactor creep test started last month has accumulated over 1000 hours of testing time. Two reactor outages have occurred during this period. The creep rate just prior to the first outage was about 4×10^{-6}/hr. Just after the reactor outage the rate increased to 6.5×10^{-6}/hr. After completion of the reactor outage and attainment of full power, the creep rate was observed to decrease with time to a value of about 1.8×10^{-6}/hr. This creep rate is probably not the steady state value as the curve appeared to be in transient creep. During the second shutdown the creep rate increased to 3×10^{-6}/hr. This behavior is in qualitative agreement with previously determined data.

Attempts were made to correlate the primary portions of the in-reactor creep curves with the Larsen-Miller, Zener-Holloman and Dorn parameters. None of these parameters worked well. Apparently neutron irradiation affects primary creep in a manner that produces creep behavior considerably different than that usually observed. In order to develop a method of correlation, the neutron effect must be rationalized from first principles. An attempt to do this has produced a very complicated expression for the creep rate. This expression indicates that the activation energy for creep

will be both time and strain rate dependent on the primary range. This is in rough agreement with observed primary creep behavior.

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 will be given to the determination of mechanical property changes produced in metals by irradiation at elevated temperatures.

Small quantities of AISI 304, 348, and Am-350 stainless steel rods were sent to offsite vendors to be fabricated into various types of mechanical test specimens. The types of test specimens requested include round tensile specimens, round smooth fatigue specimens, and round notched fatigue specimens. Pilot lots of specimens are being machined from each material and examined before final approval is given for the balance of the order. Approximately 15 smooth and 15 notched fatigue specimens have been received. The AISI 304 and 348 were in both the annealed (as received) and 25% cold worked conditions. The AM-350 was given a double aging treatment and a cold work and tempering treatment. The double aging consists of an anneal at 1710 F (932 C) for one hour followed by an oil quench to room temperature, a conditioning treatment at 1375 F (746 C) for three hours followed by an oil quench to room temperature, and a tempering treatment at 850 F (454 C) for three hours followed by an oil quench to room temperature. Another rod was cold drawn from 0.600 inch in diameter to 0.420 inch in diameter, to obtain a 50% reduction in area. Cold reduction of the rod diameter was made in 10-mil increments. After cold drawing to a 0.420 inch diameter, the rod was sectioned and tempered at 850 F (454 C) for three hours, then air cooled to room temperature. Test specimens of AM-350 stainless steel are being fabricated in both of these heat treated conditions.

Crystallographic anisotropy, which has a marked effect on the tensile properties of unirradiated Zircaloy-2, continues to influence tensile properties after irradiations at 280 C (540 F) to 2 x 10^{21} nvt. The yield strength of transverse specimens at this exposure is higher than that for longitudinal specimens, and the yield strength in both directions continues to increase with irradiation. Anisotropy plays a greater role in the strength of cold worked Zircaloy-2. At exposures greater than 1.5 x 10^{21} nvt, the yield strengths in the rolling direction are 111,000 psi in the annealed condition and 113,000 psi for both the 20% and 40% cold worked conditions; a difference of only 2000 psi is caused

by prior cold work. However, yield strength in the transverse direction increases to 113,000 psi in the annealed condition, to 114,000 psi in the 20% cold worked condition, and to 122,000 psi in the 40% cold worked condition, an increase of 9000 psi with 40% cold work. This difference in strength with direction in the cold worked material is attributed to differences in deformation mechanisms, which is principally slip in the rolling direction and twinning in the transverse direction. Deformation by twinning appears to be less affected by thermal recovery at the irradiation temperature than is deformation by slip.

Damage Mechanisms

The objective of this program is to determine the mechanism by which neutron produced defects interact with dislocations to modify the plastic deformation characteristics of the metal. The investigation is presently concerned with the role of interstitial impurities in α iron.

Additional work on interpreting the results of strain-rate change experiments has been directed toward estimating the magnitude of the long range and short range stresses contributing to the flow stress of iron over the temperature range 77 K (-321 F) to 300 K (80 F) and the mechanism of work hardening. The following general conclusions can be drawn from the experiments performed to date:

(a) The long range stresses are much greater in polycrystals than in single crystals.

(b) Work hardening during deformation at 300 K (80 F) and 200 K (-100 F) must be attributed to an increase in the long range stresses, for the $\Delta\delta$ values obtained from strain rate change tests are constant during the course of the deformation.

(c) The mechanism of dislocation forest hardening as a thermally activated process can be eliminated for these two tests although the increase in forest density may still contribute to the increase in non-thermally activated stresses.

Several attempts to zone level titanium into a $\frac{1}{4}$-inch bar of Ferrovac "E" have been only partially successful because of the large fluctuations in filament current when the element passes over a titanium rich region. The efforts to produce a homogeneous bar by this technique are continuing.

Several tests at temperatures ranging from 90 C (194 F) to room temperature have been performed in a silicon oil bath. Strain aging in the tests above 55 C (131 F) was extremely rapid and produced a serrated stress-strain curve. Below 55 C (131 F), the effects were not so marked as to control the rate of deformation, but diffusion was still rapid enough to pin dislocations when the test was stopped for a few seconds.

Environmental Effects

In-reactor weight gain results for 24 pre-autoclaved Zircaloy-2 coupons irradiated in quadrants 73 and 74 during ETR cycles 55 and 56 were obtained during the report period. These coupons were exposed to G-7 loop water at 540 F (282 C) for 38.8 days. Average weight gains and estimated fast flux exposure values are as follows: 80.1 mg/dm^2, 8 x 10^{13} nv, and 2.7 x 10^{20} nvt for quadrant 73; 70.6 mg/dm^2, 7.0 x 10^{13} nv, and 2.3 x 10^{20} nvt for quadrant 74. These gains are reasonably consistent with ETR Zr-2 corrosion data previously obtained and represent more than a 10-fold increase over estimated gains in the absence of high energy neutron irradiation.

Metallographic examination of four Zr-2 coupons exposed out of the flux zone to irradiated G-7 loop coolant only at a temperature of 515 F (267 C) for 17.6 reactor full-power operating days during ETR cycle 57 showed low weight gains comparable with out-of-reactor experience. Weight gains for pre-autoclaved and bright etched coupons averaged approximately 0.5 and 6.5 mg/dm^2, respectively. However, hydrogen pickup results for these coupons indicate that the fraction of theoretical corrosion hydrogen absorbed during the initial phases of oxidation in irradiated coolant may be higher than normally experienced in the absence of irradiation: 67% average fractional hydrogen pickup for the etched coupons as compared with a normal out-of-reactor range of 10% to 50%. Coolant transport time between the active zone and the location of these specimens was approximately one minute. All coupons were smooth, glossy and showed interference colors over large areas. There was no visible evidence of crud deposits on any of the surfaces.

ATR Gas Loop Studies

Insulation Studies. An analysis was made and reported on "The Requirements and Selection of Insulation Applicable to the ATR Test Section" (HW-79336). Insulation in various parts of the ATR test section must: (1) permit attainment and maintenance of

ultra-high helium purity; (2) be removable and rechargeable; (3) have high temperature strength -- 1093 C or 2000 F; (4) generally assure low heat losses; (5) retain structural dimensions; (6) withstand charge-discharge of test specimens; and (7) withstand intense radiation and gamma heating without deterioration.

Various brands of the three major types of insulation generally available: fibrous; solid, blown, or particulate; and metallic heat shields, were compared on the basis of meeting the specified requirements. Metallic foil heat shield insulation most nearly fulfilled the desired requirements, the main problem being a higher thermal conductivity.

Model Gas Loop Test Section. The model gas loop Mark I test section* consists of the following: a tube of type 347 stainless steel to bear the internal pressurization; metallic foil insulation at the lower and middle sections of the tube to maintain a relatively low pressure tube temperature; and a Haynes alloy 25 flow guide tube at the top of the test section for attemperation. Access to the top of the test section is provided by a Grayloc type blind flange.

Construction of the Mark I test section has continued. Machining work on the transition joints has been completed and assembly started. The metallic foil insulation has been received from Solar Aircraft.

Equipment Testing. Feasibility of using an ultrasonic leak detector for locating pinhole leaks in double contained piping with cast Fiberfrax between pipes was investigated. A three- to four-mil hole was placed in the pressurized inner pipe. By using a contact probe on the bare outer pipe at 300 psi, the leak could be narrowed to about a 4-inch long band near the leak. Increasing pressure to 500 psi did not permit further resolution although sensitivity was increased, making the leak more readily detectable. No detection resulted when one inch of insulation was placed around the outer pipe. With no insulation on the inner pipe and with only 8 psi internal pressure the leak could be detected without contact from a 15-foot distance.

Fluid Flow. Theoretical calculations indicated an excessive pressure drop (130 psi) for the tensile specimen assembly proposed

*D. R. Doman and R. L. Knecht, "The Model High Temperature Gas Loop Test Section - An Interim Report," HW-69159, October 3, 1963.

for the ATR gas loop. An experimental test was made in which the results substantiated the theoretical calculations. In this test the pressure drop was measured in air for one, two, and three layers of the proposed twelve layer assembly. Pressure drop calculations on the ATR loop test assembly agreed with those of the architect engineer when similar assumptions were used. A more exact calculation was computer programmed and found to accurately predict the results of the tests run with compressed air.

An engineering study of the tensile specimen holder was made which gave acceptably low pressure drop calculated values (~ 15 psi). The modified design incorporated the following changes: (1) minimal number of tensile specimens compatible with providing the required data for the testing program, (2) slight modification of the tensile specimen design to provide fewer abrupt changes, and (3) minimum structure for holding the tensile specimens.

Thermal Strain Cycling Test. The test configuration and heater element design of the thermal-strain cycling test was checked out by building a stainless steel prototype unit. The test section consists of a 2-inch pipe inside a 6-inch pipe, with a heavy plate welded across both ends. The annulus between the pipes is filled with insulation, and both pipes are pressurized to about 10 psig with helium. The heater is a coiled tungsten wire. A temperature of 1093 C (2000 F) was attained on the inner pipe in 45 minutes. Temperatures will be cycled between 260 C (500 F) and 1093 C (2000 F). Materials for the inner pipe will be Haynes 25, Hastelloy X, and other candidate materials for the model gas loop.

Stress Rupture Tests of Haynes 25. Haynes 25 alloy is a candidate structural material for the ATR gas loop. A program for evaluation of the stress-rupture properties of Haynes 25 at 2100 F (1149 C) has been completed. The following results were obtained for stress-rupture life of Haynes 25 at 2100 F (1149 C):

Stress (psi)	Time to Rupture (hrs)
4294	0.78
3800	1.51
3000	3.49
1504	239.00

The stress-rupture curve plotted from the above data predicts a 1000-hour rupture life at 1100 psi for 2100 F (1149 C) temperature.

Corrosion of Inconel 600. Tests of Inconel 600 are being conducted at 1200 C (2192 F) in various pressures of oxygen. In oxygen atmospheres ranging in pressure from 5 mm Hg to atmosphere air, the alloy exhibits better corrosion resistance than either Haynes 25 or Hastelloy X. Though Inconel 600 also loses weight when exposed to conditions of high vacuum ($\sim 10^{-4}$ mm Hg) at 1200 C, the weight loss is not as rapid as that exhibited by Haynes 25 or Hastelloy X, most likely because of its lower chromium content.

Design Test 1172. Thermal and bending tests of a Grayloc connection for the ATR Gas Loop is scheduled to be complete by January 15, 1964; however, a considerable amount of test data will be available before that date.

Exposure Normalization

The exposure of graphite specimens has been often expressed as MWD/AT (megawatt days per adjacent ton of fuel). This unit is useful to relate radiation effects to reactor operation and life; however, it can lead to serious error unless it is derived and used properly. Recently conversion factors relating fast flux to power have been revised by use of the most advanced computational techniques available.

Data for hot test hole facilities in the Hanford K and C Reactors and for graphite supporting the process tubes in N-Reactor are presented below. They illustrate the differences that can be experienced in graphite reactors.

Reactor	$\phi > 0.18$ mev MWD/AT	$\phi > 0.18$ mev / $\phi > 1$ mev	Fe ($\phi > 0.18$)	Ni ($\phi > 0.18$)
N	1.88×10^{17}	2.54		
K	1.19×10^{17}	2.60	30.4	40.8
C	0.953×10^{17}	2.69	29.0	38.8

The ratios of fast flux to power shown above were derived from core analysis calculations which gave fission rate in the fuel and fast flux in the zone of interest for graphite studies. These theoretical ratios agree well with those obtained from irradiation experiments in C and K Reactors in which fast flux was calculated using the effective cross sections presented above, and local reactor power obtained from operating data. It

is evident from these ratios that an error by a factor as great as two could be made in estimating a fast neutron exposure if the flux distributions in these graphite moderated reactors were all considered identical.

6. Gas-Cooled Reactor Studies

Thermal Oxidation of Large Graphite Samples by Water Vapor. Measurements of the rate of oxidation of relatively large graphite samples by small partial pressures of water vapor (7000 ppm in helium) have been extended to TSX graphite samples 12 inches long, 0.5 inch in diameter, and weighing about 65 grams. The results obtained to date are compared below to the rates observed on samples 1.5 and 2 inches in diameter.

Temp., °C	Oxidation Rate, hr^{-1}		
	0.5" Dia.	1.5" Dia.*	2" Dia.*
725	4.80×10^{-6}	1.16×10^{-6}	1.22×10^{-6}
763	9.22×10^{-6}	2.84×10^{-6}	2.99×10^{-6}
813	1.87×10^{-5}	8.40×10^{-6}	8.86×10^{-6}

*Based on interpolation to the temperature of interest.

From this table it is readily apparent that the oxidation rates of the 0.5-inch diameter sample are appreciably higher than those for the larger samples, whose rates are probably the same within experimental error.

The Radiolysis of Carbon Monoxide and Water. In order to investigate the effect of inert gases upon the radiolysis of carbon monoxide-water mixtures, samples containing various amounts of argon were irradiated in the cobalt-60 facility.

Known quantities of the gases were sealed into ampoules. The argon pressures were varied but the pressures of carbon monoxide and water were held constant for this series.

The hydrogen produced from energy absorbed by argon was calculated by subtracting that formed by direct absorption of energy by water and carbon monoxide from the total hydrogen yield. (The 100 ev yield for hydrogen production by absorption of energy in

CO + H_2O was taken as 7.35, from a single run with no argon present.) The efficiency of energy transfer ranged from 0.6 to 0.8.

Although there is considerable scatter in the data, it seems probable that the efficiency of energy transfer from argon to CO and H_2O is in some way dependent upon the fraction (CO) + (H_2O)/(CO) + (H_2O) + (A).

EGCR Graphite Irradiations. The H-3-7 capsule continues to operate satisfactorily in the GETR. It has successfully completed one cycle of reactor operation.

Effect of Impregnation on High Temperature Contraction. The fourth irradiation has been completed in a test to determine the effect of impregnation with furfuryl alcohol on the high temperature radiation-induced contraction of two types of graphite, TSX (N-Reactor graphite) and HLM-85 (used in HTGR fuel elements). Added impregnant carbon amounted to 8% for TSX and 4% for HLM-85. Irradiation temperatures are near 1000 C and the total accumulated exposures range from 2.5 to 4.5 x 10^{21} nvt, E > 0.18 Mev.

The most interesting results were obtained from the TSX transverse samples. The samples heat-treated to 900 and 2650 C after impregnation, showed, for the first time, expansion - respectively 0.008 and 0.08% during the last irradiation. Both the control and the impregnated 2900 C sample displayed a contraction. For all samples in this set, the accumulated length change was approximately -0.5%. This result is consistent with similar expansion in the transverse direction in EGCR graphite at higher exposures but at roughly the same total contraction.

The fact that expansion is noted in samples which were impregnated and heat treated to 900 and 2650 C, but not in the control nor in the impregnated 2900 C sample, could be explained in the following way. The contraction rate of the impregnant carbon is higher for lower heat treatment temperature. Thus saturation of contraction takes place more rapidly in the samples heat treated to lower temperature. With the impregnant carbon no longer contracting, the net crystallite effect, which is an expansion, becomes evident.

In the parallel direction greater contraction was noted in the graphitized impregnated samples than in the control samples. In all instances the impregnated parallel samples heated to 900 C were more stable dimensionally than the graphitized impregnated samples.

7. Graphite Radiation Damage Studies

Hanford Graphite Irradiations. Radiation-induced dimensional changes were measured on samples from a series of molded Texas-coke graphites in which the heat treatment temperature was varied from 1370 to 2700 C. Total exposure at 500 to 650 C for this group of samples was extended to 9500 MWD/AT_C or about 9×10^{20} nvt, E > 0.18 Mev during the last irradiation period.

The history of dimensional changes during previous exposure periods* is characterized by a high contraction rate for samples with the lowest heat treatment temperature and lower rates of contraction with successively higher heat treatment temperatures. Evidence of saturation of contraction at 1.85% was observed after 7800 MWD/AT_C in the samples heat treated to 1370 C.

Between 7800 and 9500 MWD/AT_C, the 1370 C samples grew 0.2% leaving a net contraction of 1.65%. Other graphites in the series continued to contract at the rates measured in previous irradiations. As a result, the curves in which length changes are plotted versus exposure for the 1370 and 1800 C heat-treatment samples intersect at approximately 8800 MWD/AT_C. Contraction of the 1800 C samples at 9500 MWD/AT_C is 1.85%. This is the same value at which saturation of 1370 C samples was noted at 7800 MWD/AT_C. There is, however, no indication that contraction of the 1800 C samples has saturated.

Irradiation of this series will be continued to determine whether the magnitude of contraction at saturation is a function of heat treatment temperature or whether it is the same for all materials in this series. Characteristics of growth and contraction as a function of processing temperatures will also be followed with interest.

Density Decreases in Graphite During Plastic Deformation. Density decreases observed in graphite during creep and other plastic-flow processes can be explained by considering the criterion for continuity of material in polycrystals during deformation. This criterion is that five independent means of plastic deformation must occur simultaneously in order to prevent void formation at grain boundaries. It originates from the fact that the state of strain at a point in a body is completely described by the six

*HW-SA-2467, "Summary of Recent Hanford Irradiations," R. E. Nightingale, March 2, 1962, Figure 7.

principal strain components (three normal and three shear). Equating the sum of the three normal components to zero (single crystals have never been observed to change volume during plastic deformation) leaves five undetermined principal strain components. Graphite possesses only two independent means of plastic deformation, which are by glide on the (0001) <2110> slip systems.* Hence plastic deformation will occur in graphite only with the formation of voids and a subsequent decrease in density. This conclusion provides an explanation of the brittle behavior of graphite at room temperature and the formation of voids during high temperature creep.

The pre-existence of pores in graphite should not hinder density decreases during plastic deformation because slip increases the surface area of the crystallites through the formation of slip offsets. Consequential increases in pore volume can only be prevented if five independent means of deformation exist simultaneously so that crystallite deformation is restricted to the same changes in shape as their adjoining grains.

8. Boronated Graphite Studies

Long-term Irradiation Test. The final design of the long-term irradiation capsule has been completed. Expected sample temperatures are 1000 ± 150 F and 650 ± 150 F.

The basic sample length was changed from 1.00 inch to 0.75 inch to increase the number of samples from 33 to 42. This will allow a better characterization of the effect of boron level and type of material (black and grey), yet will provide sufficient accuracy and precision in measurement.

The capsule will be instrumented with 12 thermocouples. The thermocouples are located in the samples and inner capsule parts so as to provide a good approximation of the irradiation temperature of all noninstrumented samples.

Instrumentation. Adequate instrumentation has been provided for the long-term irradiation tests in the 2-C facility. A multipoint potentiometric recorder was modified to record temperature over the full range expected for the 12 capsule thermocouples. The printing speed was modified to minimize chart changing.

*Groves, G. W., Kelly, A., Phil. Mag., 69, p. 883 (1963).

Pressure changes due to the evolution of helium within the test capsule will be measured by a pressure gauge that is accurate to ± 0.1% over the range of 0 to 150 psia. This gauge, which is connected to a calibrated volume, will also be used to measure the volume of the test capsule and the connective tubing. From pressure measurements taken at intervals during the capsule irradiation, the amount of helium (from the (n,α) reaction) liberated will be determined.

9. Metallic Fuel Element Development

Irradiation of Thorium-Uranium Fuel Elements. Three tubular Zircaloy-2 clad thorium - 2.35 wt% U-235 - 1.0 wt% Zr fuel elements have completed four cycles of irradiation in the ETR P-7 loop. The fuel elements were irradiated for a part of the current cycle. The maximum exposure achieved is 1.3×10^{20} fissions/cm^3 (3600 MWD/T). Weight measurements made at the end of cycle 57 showed a total fuel volume increase of 0.5% for the highest exposure fuel element and 0.3% volume increase for the two lower exposure fuel elements.

Irradiation Test of Uranium Containing a Submicron Dispersion of Uranium Carbide. Data from several sources indicate that a finely dispersed second phase in uranium can reduce swelling. The mechanism whereby this occurs is believed to involve the limiting or stopping of the migration of small fission gas pores, thus preventing their agglomeration into large pores. For a given amount of fission gas generated, the small gas pores, through their greater surface tension restraint, will result in considerably less uranium swelling than larger pores.

To investigate the ability of a submicron dispersion of uranium carbide to reduce uranium swelling, fuel rods were produced from chill cast uranium shot by extrusion techniques. NaK capsules are being prepared for irradiation of these samples in the ETR and MTR. Fabrication, assembly, and final testing of three capsules is complete and the capsules have been shipped to the MTR-ETR site for irradiation to start in December. Of these three capsules, two will compare the swelling performance of two fuel rods identical in uranium composition, but with one having a uranium carbide size of 2-5 microns produced from the arc-melted uranium and the other a carbide size of less than 0.5 microns produced from the uranium shot. These two capsules will be irradiated to approximately 0.3 and 0.6 at% burnup. The third capsules will contain two fuel rods with the fine carbide and will be irradiated to approximately 1 at% burnup.

Thorium-Uranium Alloys. The fabrication, irradiation and defect corrosion behavior of the Th - 2.5 wt% U - 1.0 wt% Zr alloy has been described. The effects of higher zirconium and uranium compositions on the structure, fabrication and defect corrosion behavior of thorium base alloy fuels have not been thoroughly studied. A program has been initiated to study these effects.

Nine additional alloys have been primary melted and are being prepared for extrusion into electrodes for remelt. These ingots are 2.9" diameter x 11" long and weigh approximately 32 pounds. Room temperature hardness readings were taken on the primary ingots. The tests show a hardness minimum between 2 and 3 wt% Zr on each of the uranium levels, while the difference in hardness between the uranium levels remains nearly constant for each zirconium addition.

Thoria Development. Twelve thoria fuel elements were canned and charged into a production reactor. These elements are scheduled for reactor discharge on November 22, 1963.

An Appropriation Request was approved to procure and install thoria fuel development equipment. Hood fabrication and equipment procurement is in progress.

Preliminary tests of pneumatic impaction of thorium oxide were performed. Pot calcined ThO_2, impacted at 1200 C and 326,000 psi impact pressure, reached density of 8.98 g/cc (89.5% TD). Spray calcined ThO_2, impacted at 1200 C and 357,000 psi impact pressure, achieved a density of 9.60 g/cc (95.7% TD). A ceramic grade ThO_2 powder having a tap density of approximately 25% TD was impacted at 1200 C and an unknown impact pressure, to a significantly lower final bulk density, 7.97 g/cc (79.4% TD).

10. Aluminum Corrosion and Alloy Development

C-1 Loop. Following overheating of the dummy fuel train, an attempt was made to remove only the inner tube of the C-1 test section for replacement. It could not be pulled free, presumably because it had been deformed by the swelling of the aluminum dummies on being converted to oxide. The entire in-reactor test section was removed, and a new test section fabricated and installed. The outer tip of the dummy train was removed and examined; it was found to have melted but did not undergo heavy corrosion.

A new dummy train was charged and acceptance tests were run. Fueled tests are expected to begin about the first of the year.

11. Advanced Reactor Concept Studies

Fast Supercritical Pressure Power Reactor. A draft of the report on economic evaluation of the Fast Supercritical Pressure Power Reactor was completed and is being issued for comment.

A request was received from the Chicago Patent Office for additional invention reports covering certain features of the core and fuel design.

A paper describing the moderator segmented core concept was delivered before the ANS Annual Meeting on November 19.

Calculations to approximate the burnup in a 1000 Mwe Fast Supercritical Pressure Power Reactor have been completed. These calculations are based on the assumption that the flux distribution and energy spectrum remain relatively constant over the core life. Another assumption is that there are no gross reactivity changes over the core life. The total breeding ratio, based on these assumptions, was calculated to be 1.136 with the core breeding ratio 1.04 for a fuel exposure of 97,500 MWD/T. These numbers indicate that the last assumption, no gross reactivity changes, may be incorrect because the reactivity will probably increase over at least part of the fuel cycle.

The presence of a flux trap region between the core and the blanket reduces the plutonium buildup in the blanket, thus a thinner blanket is required than for the 300 Mwe FSPPR.

An 18-group cross section set for use in large sodium-cooled fast reactors has been assembled. However, it has not yet been checked with respect to cross section accuracy. The cross section set was assembled on magnetic tape using the "Physics Chain" which has been expanded to handle 20 energy groups. "Physics Chain" computes cross sections and assembles them in the form of "HFN" input data, thus reducing the manual data handling to a minimum. The 18-group cross section set is a debug run on the revised "Physics Chain".

Military Compact Reactor - Plutonium Fueling Studies. A document, HW-79449, summarizing the studies reported last month, was completed and forwarded to AEC-DRD. Discussions were held with personnel of the DRD office of the Reactor Evaluation and the Army Reactors Branch, resulting in a request for some brief supplemental studies to be completed by the end of the year. The study reported in HW-79449 indicated that significant reduction in size and weight of the reactor, increase in reactor power, or increase in core

[redacted]

endurance could be achieved by using plutonium fuel. Four plutonium compounds (PuO_2, β-Pu_2O_3, and PuN) were considered as fuel materials, in ceramic form or as a niobium-base cermet.

Results of the studies include:

1. Retaining the basic MCR fuel-pin core design, 15 Mw(th) power level, and 3600 Mwd endurance, core size could be reduced from the original 15-inch diameter (L/D=1) to from 8.5 to 9.5 inches diameter (L/D=1), depending on which plutonium fuel material is used and whether the fuel is in ceramic or cermet form. For a 9-inch diameter core, a total shield weight savings of 28% would be made.

2. Retaining the MCR core design, size, and power level, a doubling of the core endurance is readily achievable, and potential endurance increases of from 4 to 6 times that of the original design appear feasible.

3. Again retaining the original core design and size, and requiring a core lifetime (240 days) equal to that of the reference core, a power increase to approximately 3.5 times the 15 Mw reference power appears feasible.

4. Utilizing Hanford-developed "honeycomb" or "gridplate" compact fuel designs, a reactor power of 15 Mw(th) and a 3600 Mwd endurance should be achievable with a core from 7 inches (ceramic fuel) to 7.5 inches (cermet fuel) in diameter, again with L/D=1. Shield weight savings for the 7-inch core would total over 14,000 pounds, or better than 33% of the original shielding weight.

All cases included in the studies involved some increase in fuel burnup. Even for the smallest cores considered, the maximum burnup is on the order of 136,000 Mwd/T, a figure comparable with goals of several other advanced fuel programs. However, the problems of fuel integrity and of fission gas retention or disposal at the higher burnups would require a solution in order to utilize the full potential of plutonium fuel.

The study also pointed out the problems involved in maintaining a negative temperature coefficient of reactivity in compact cores. From this aspect the cermet fuels may be most promising for compact cores.

[redacted]

SONAR Power System. Some study has been given to possible nuclear power sources for underwater sonar systems. Previous designs of compact power reactors have been examined. Progress of this study is handicapped by lack of experience with sonar transducers and lack of knowledge of possible application requirements. At month-end, a visit was made to GE-HMED in Syracuse to obtain information on sonar systems.

Engineering Calculation Codes. Some of the recurring problems which are handled in DAO studies are being prepared for analysis by existing computer codes.

FUGUE is a boiling liquid coolant thermal hydraulics program prepared by Atomics International. Currently the physical and thermodynamic properties of potassium have been correlated into a form compatible with the code to assist in the pressure drop and film temperature drop calculations for the current MCR study.

STHTP is a steady state heat transfer program written by F. J. Mollerus of NRD. The input for heat conduction calculations for pin-type fuel, square channel gridplate fuel, hexagonal gridplate or honeycomb fuel, and the round channel honeycomb or inverted fuel element fuel temperature is being prepared for use with this code. The analytical models used were simplified to limit the input requirements where various parameters are changed in the course of a study. To date the pin-type fuel and square channel gridplate models have been run.

Pipe Flexibility Analysis (PFA) is a code written by F. J. Mollerus for the analysis of stresses, etc., in a three-dimensional piping system. This code is currently being used for the analysis of the flexibility of the pressure tubes in the FSPPR fuel elements.

Four computer codes were ordered from the Argonne Code Center for the evaluation of their applicability to DAO work. They are:

1. SEAL-SHELL I for calculating stresses in a thin shell.
2. IV Restraint Pipe Stress, Forces and Moments.
3. CROCK - space power plant design optimization code.
4. SHOCK space power plant radiator design optimization code.

[redacted]

Hydride Moderator Studies. A variable concentration zirconium hydride moderator is under consideration as a means of reactor control. An investigation of the equilibrium composition and kinetics of the hydrogen-zirconium hydride system will be undertaken to assess the feasibility of control concepts. The pertinent variables are temperature, pressure, hydride composition, size of hydride particles, hysteresis effects, and contamination of the hydrogen with other gases. In the reactor under study temperature will vary from 1400 F to 2100 F (760 C to 1150 C) max., hydride composition will be approximately 5×10^{22} hydrogen atoms per cc ($ZrH_{1.27}$), size of hydride pellets may be up to one inch, and pressure will range from approximately 0.5 to 2.0 atmospheres. The equilibrium studies will check literature data and determine whether the system is clean. The kinetic studies will provide the basis for control in the proposed hydride moderator. Control of the composition can best be accomplished in the single phase, beta region by a simple temperature or pressure adjustment. The maximum hydrogen content with beta phase only being present would be $ZrH_{1.16}$ which can be obtained at 930 C (1706 F) with pressures up to two atmospheres. To maintain control in the range $ZrH_{1.16}$ to $ZrH_{1.27}$ requires either a high pressure system (up to 10 atmospheres) to maintain the single beta phase or a two-phase hydride. A two-phase system is more difficult to control because at a given temperature and pressure the composition is not uniquely defined. Thus, a control system must be used which meters the amount of hydrogen put into or taken out of the system by a given change of temperature or pressure. This is obviously not as desirable as simple temperature or pressure control in the beta field but can be done with proper design.

Preliminary hydriding studies to establish the adaptability of available equipment for the equilibrium studies have shown that a vacuum of 10^{-6} mm mercury is sufficient for this work.

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.

Strain-rate cycling experiments are being continued on unirradiated polycrystalline molybdenum. In this type of experiment the strain rate is instantaneously increased or decreased by a factor of ten for several cycles during tensile loading of the specimen. The increase in flow stress resulting from an increase in strain rate is termed Δt. The results of recent tests indicate that the variation of Δt with t, the flow stress at the lower strain rate at the instant of strain-rate change, is not, as reported previously, a parabolic function. The variation of Δt with t has been found instead to be linear up to flow stresses approximately twice the yield stress; above this limit the relationship becomes exponential. Low temperature strain-rate cycling experiments have been delayed because of difficulties in gripping the tensile specimen. The yield stress of the annealed polycrystalline molybdenum under investigation is almost three times greater at 77 K than at room temperature, and under this combination of stress and temperature the materials used in the existing grips exhibit brittle behavior. Several designs for low temperature grips are presently being evaluated.

An investigation on the effect of impurity content on quench-hardening in nickel is in progress. Three materials are being studied: (1) spectroscopic grade nickel, 99.997% Ni, supplied by Johnson-Matthey, Ltd.; (2) high purity nickel, 99.97% Ni, type 270; and (3) low purity nickel, 99.4% Ni, type 200. The specimens are in the form of foil 0.002 inch thick, $\frac{1}{2}$-inch wide, and 1-7/8 inch long. The gage section of the specimen is 1/8 inch wide and 3/4 inch long. The specimens are quenched in a purified helium atmosphere in a chamber surrounded by liquid nitrogen. All quenches to date have been from 1675 K, with resultant quenching rates of the order 10^4 degrees per second. Tensile tests conducted at 77 K reveal that considerable quench hardening is achieved in the higher purity materials. The ratio $\Delta\sigma Y/\sigma Yo$, where $\Delta\sigma Y$ is the increase in yield stress after quenching and σYo the yield stress in the fully annealed condition (with equivalent grain size) was found to be as follows: Johnson-Matthey (99.997%) Ni, 0.80; 270 (99.97%) Ni, 0.15; 200 (99.4%) Ni, 0. The quench hardening in the type 270 nickel may arise in part from quenching strains, but the magnitude of the increase in yield stress observed with the nickel of 99.997% purity is too great to be thus accounted for and so must be due largely to the presence of quenched-in vacancies. This result is notable in view of the fact that other investigations of quench-hardening in nickel have observed no hardening until subsequent aging treatments near the Curie temperature (638 K). Further verification of these observations, as well as post-quenching aging treatments, is in progress.

Previous irradiations of annealed and cold-worked molybdenum foils disclosed a marked difference in the occurrence of defect clusters. Foils which were heavily deformed by rolling failed to show defect clusters, while annealed samples irradiated to comparable exposures showed a high concentration of clusters. In order to investigate this phenomenon further, molybdenum foils 0.003 inch thick x 3/8 inch wide x $1\frac{1}{2}$ inches long have been deformed in tension to strains of 1%, 2%, and 5%. The foils were annealed for 20 minutes at 2175 K before deforming at room temperature. Deformation was carried out on an Instron testing machine in a special fixture designed for testing thin foils. This fixture permits accurate alignment of the specimen and prevents bending or wrinkling of the specimen during attachment to the pull rods. A molybdenum foil specimen given the above annealing treatment and deformed to fracture at room temperature exhibited the following properties: upper yield stress - 34,400 psi; lower yield stress - 33,420 psi; uniform elongation - 20.5%; total elongation - 25%.

Transmission electron microscope examination of 0.075 mm thick molybdenum foils containing 10 ppm carbon and 100-200 ppm carbon, irradiated to 1, 4, and 7 x 10^{18} nvt (E > 1 mev), is in progress. Foils containing 10 ppm carbon irradiated to 1 x 10^{18} nvt did not show an appreciable number of defect clusters (spots), only about 4 x 10^{13} /cc; the foils containing 100-200 ppm carbon revealed a spot defect concentration of 1 x 10^{16}/cc at this exposure. After an exposure of 4 x 10^{18} nvt, the defect concentration in the low carbon foils increased to 1.5 x 10^{16}/cc, while that in the foils containing 100-200 ppm carbon had increased only slightly, to 2 x 10^{16}/cc. Prismatic dislocation loops formed by defect clusters were observed only in the foils with the higher carbon content. The size of the spot defect clusters appears to remain constant with varying exposure, at approximately 50 A in diameter. The abrupt increase in the number of clusters in the low carbon molybdenum between 1 x 10^{18} and 4 x 10^{18} nvt, while the higher carbon material reaches a higher concentration at lower exposures, suggests that the formation of clusters in low carbon molybdenum occurs by means of homogeneous nucleation, whereas nucleation of clusters in molybdenum with 100-200 ppm carbon occurs heterogeneously, at sites such as interstitial carbon atoms or large substitutional impurity atoms (in the case of vacancy clusters). Thus, in the absence of such nucleating sites, the concentration of single defects must be much higher (i.e., a greater exposure) before nucleation will occur. Once this critical concentration is achieved, then the number of spot defects (clusters) increases rapidly. After nucleation is complete, cluster growth should predominate and further irradiation would

then result in an increase in the size of the clusters with no great change in density. Examination of foils irradiated to higher levels is expected to substantiate this hypothesis.

A commercial wafering machine has been installed in the laboratory to cut thin sections from crystals suitable for electrothinning and subsequent transmission electron microscopy. Techniques for cutting wafers parallel to predetermined planes from molybdenum single crystals are now being evaluated.

Molybdenum foils containing < 10 ppm and 150 ppm carbon, irradiated to 1×10^{18}, 3×10^{18}, 6×10^{18}, and 1×10^{19} nvt (E > 1 mev), have been examined by x-ray diffraction. Line breadths increased very slightly, from about 0.24° 2θ to about 9.28° 2θ for the (400) line after 10^{19} nvt. Broadening was approximately linear with exposure and was the same for both materials. The peak profile is slightly changed, becoming broader at the base in a manner suggesting thermal agitation of the atoms. A random distribution of defects would account for the same effect. The peaks are being further analyzed by a Fourier method. Lattice parameters also increased with exposure in about the same manner for both materials. These results are in marked contrast to those obtained in earlier experiments, where little or no change was observed in the lattice parameter of low carbon (< 10 ppm) molybdenum after irradiation to 10^{19} nvt. This discrepancy is as yet unexplained; it may be due to possible differences in pre-irradiation annealing treatments.

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 these properties.

The steady state creep characteristics of the beta and gamma phases of plutonium were determined as a function of temperature in the range of 160 to 190 C and 185 to 290 C (320 to 374 F, and 365 to 554 F), respectively, at a compressive stress of 112 kg/cm^2. The steady state creep rate of the beta phase varied from 0.06% per hour at 163 C (323 F) to 0.8% per hour at 190 C (374 F), whereas the creep rate of the gamma phase varied from 0.01% per hour at 190 C (374 F) to 0.95% per hour at 285 C (545 F). Accordingly, at 190 C the creep rate of the beta phase is 80 times the creep rate of the gamma phase. The lower creep strain of the gamma phase is due to the more close packed structure of the gamma phase, and hence a

slower diffusion rate. The activation energies for creep of the gamma and beta phases under these conditions were computed to be 24,000 and 38,000 calories per mole, respectively. This activation energy for beta creep is 8000 calories per mole higher than that previously reported at compressive stresses of 150 and 217 kg/cm^2 in the temperature range of 120 to 150 C (248 to 302 F). One significant experimental variation that may have contributed to this difference was that the activation energy of 38,000 calories per mole was obtained utilizing 1.16 cm diameter specimens whereas the value 30,000 calories per mole was obtained at higher stresses and with 0.64 cm diameter specimens.

The transformation strain during the alpha to beta transformation was very nearly a linear function of the applied stress. It increased from zero at a compressive stress of 7 kg/cm^2 to 2.5% at a compressive stress of 112 kg/cm^2 for an average transformation rate of 1 to 2% per minute. At an average transformation rate of 0.1% per minute the transformation strain was 3.5% at 112 kg/cm^2. A new experimental apparatus which has been installed will improve future work on transformation strain and should yield more precise data.

In order to elucidate the effect of beta phase deformation by compression on the subsequent beta to alpha transformation, 23 specimens were plastically deformed varying amounts at temperatures between 110 C and 180 C (230 and 356 F) and transformed to the alpha phase at 80 C (176 F). There are yet insufficient data to clearly understand deformation in the beta phase and how it relates to the mechanisms of the beta to alpha transformation. However, two factors are clear: firstly, plastic deformation at 150 to 180 C (302 to 356 F) increases the beta to alpha incubation time and decreases the maximum rate of transformation; secondly, decreasing the temperature of beta deformation decreases the incubation time and increases the maximum rate of transformation. For example, the beta to alpha incubation time at 80 C after beta heat-treating one hour at 180 C and plastically deforming (80% change in length) at 180 C was $\sim$ 1750 $\pm$ 250 seconds. On the other hand, the incubation time was less than 50 seconds after beta heat treating one hour at 180 C and deforming (80% change in length) at 110 C. The maximum rates of transformation after deforming at 180 C and 110 C were approximately 0.35 and 3.5% per minute, respectively.

Further analysis of the diffractometer patterns of alpha-rolled plutonium, both from the rolling plane and from the surface of the transverse section of the strip, have confirmed the basic crystallographic orientation reported earlier.

Attempts to extend the rolling investigation employing the cast strips produced at the same time as those used in the initial experiments indicate a subtle change in properties. This material, now some three months after casting, can no longer be readily reduced in thickness. In fact, reductions of no more than about 10% have been possible prior to what amounts to catastrophic crumbling. This is true even in material which has previously been rolled to some 40% reduction in the beta phase prior to the attempt at alpha rolling.

It would appear that this deterioration in workability is the result of self-damage. In view of the fact that prior working in the beta phase and the subsequent transformation does not materially improve the working characteristics, it must at this time be assumed that the accumulation of helium atoms resulting from the decay process is the overriding factor involved. The elevated temperature treatment and the subsequent atomic mobility involved in the phase transformation should remove the effects of the dislocations introduced by the energetic particles inherent in the decay process. Preliminary vacuum annealing studies suggest that this helium cannot be effectively removed from solid plutonium.

Further evidence of the development of significant lattice strain with age arises from the diffraction pattern of the as-cast material. It had been found in the case of the freshly cast material that the $(\bar{1}13)$-(201) and the (020)-(211) line pairs could be readily resolved. After some three months, however, the same diffractometer conditions are no longer able to distinguish between these associated reflections. A similar situation has been found with those planes having higher two theta values whose $K\alpha$ doublet is resolvable in the freshly cast material.

A more definitive study of these effects can more advantageously be conducted on material of greater purity. It would appear at this time that self-damage can exert a significant effect upon the properties of plutonium and is worthy of the type of investigation which would lead to a better understanding of the manner in which these changes are effected.

A metallographic technique for plutonium which involves coating specimens with a thin film of plastic is under investigation. Examination of identical areas of a sample of furnace-cooled 1040 steel before and after coating with a layer of Formvar approximately 0.5μ thick revealed that the coating definitely increased contrast and resolution. This was particularly noticeable at 1500X. In addition to improving image quality, the plastic film

should mitigate the toxicity problem of plutonium. Preliminary observations showed that the oxidation rate of alpha plutonium is less for samples coated with the Formvar.

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.

The following listing includes major items of work done during the month:

a. Metallography

Samples Processed	59
Photomosaics	6
Autoradiographs	9

b. Chemistry

Burnup dissolutions	14
Decladding dissolutions	15
Rare gas collections	4

c. Physical & Mechanical Testing

Tensile Tests (Room Temp.)	80
Hardness Tests	32
Density Measurements	20

d. General

Negatives processed	543.

Microhardness Tester Blister for "I" Cell. A purchase order was written for a Riehle testing machine including design features and accessories required for remote operation.

Remotized 15,000# Tensile Grips. The fabrication of components was completed by Tech Shops.

Waste Handling and Disposal. Evaluation and design of improvements to the solid waste handling operation in 327 Building was continued.

An evaluation of commercially available waste containers was completed. The design of improved equipment for handling and transporting waste to the burial ground is under way.

High Temperature Tensile Testing Machine. Acceptance testing on the modified furnace, hydraulic equipment, furnace controls, and the complete assembly is scheduled for December at the vendor's plant. Upon satisfactory completion, the machine will be shipped.

"E" Cell Metallography Facilities. Design of the cathodic etcher is complete. Procurement and work orders are being issued. The cell castings and lead glass viewing ports have been received and are ready for installation.

Orbital Polishers. Two orbital polishers have been ordered for test grinding and polishing of nitrides, carbides, and other materials.

Plug-Mounted Sander. Fabrication of the sander was completed and is ready for testing.

2. Metallography Laboratories

During the report month 767 samples were processed, a total of 1004 macrographs and micrographs taken, 2124 negatives printed, and 7025 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.

Continued and extensive support has been given to NRD, N-Fuels Product Engineering, in the examination of spot welds for fuel element supports and locking clips. Test welds were made by NRD to duplicate production conditions which existed as much as 15 months ago. These welds were nondestructively tested first, then sectioned and examined metallographically to establish the nondestructive test pattern for acceptable and nonacceptable welds. A good deal more work may be required before a complete test pattern is established for nondestructively testing existing stocks of completed fuel elements.

Ultrasonic tests performed by Physical Testing Operation for cracks in Incoloy 800 tubing provided no definable results

because of high background noise. Metallographic sections of this material revealed numerous inclusions of variable size uniformly distributed throughout the metal and a few microcracks on the surface. The relative size of the microcracks versus the inclusions was such that the inclusions would return a signal large enough to mask that obtained from the microcracks.

The construction of an all-metal vacuum cathodic etching apparatus has been virtually completed. The new etcher was designed and installed in a radio-frequency shielded enclosure to attenuate the RF signal of the high frequency generator according to F.C.C. regulations. An operating frequency has also been assigned by the F.C.C. and a crystal-regulated high frequency generator purchased to assure that the assigned frequency is maintained. The new etcher is currently undergoing startup testing.

3. N-Reactor Design Testing

N-Reactor Magazine Loader. The magazine loader was used during a recent series of N-Reactor charging tests at the 314 Building. The loader successfully passed all prescribed tests. Final modifications, painting, and preparation for shipment were accomplished and the loader, tongs, spare parts, and associated test equipment were shipped to the N-Reactor site on schedule. Results of Charging Machine Design Test No. 23 were reported in HW-79535.

N-Reactor Charging Mockup. The initial testing period of N-Reactor fuel elements was completed after 12- and 18-inch fuel elements were run through the magazine loader and charge machine mockup. No significant self-support damage was observed. In an attempt to determine a threshold force for damaging of the 24-inch self-supports, charges were run at backpressures of 700, 900, 1100, 1300, 1500, 1700, and 1900 lb. No significant (> 5 mils) damage to the supports was observed, thus completing this phase of the testing.

4. High Temperature Lattice Test Reactor Mockup

HTLTR Prototype. Detailed layout drawings of the core insulation, and structural drawings of the shell, are 93% and 87% complete, respectively. Drawings of the recirculating gas system are 35% complete.

A small heating element test mockup is being fabricated in which to test shortened heating elements. The heating element test mockup will consist of a 4-foot heating element pass through a single graphite core piece. Insulating brick enclosed in a metal shell

will surround the graphite core piece. Detail drawings of the heating element test mockup are completed. All the materials to be used in the mockup are now on site and fabrication of the containment shell is in progress.

Orders have been placed for the following HTLTR prototype materials and equipment:

(1) Insulation
(2) Transformer
(3) Blower
(4) Structural Steel
(5) Vacuum Pump.

A portion of the brick and all of the required structural steel have been received, and fabrication of the containment shell for the prototype has started.

The initial detailed drawings for the safety and control rods have been completed and are being checked. A significant change has been incorporated in the control rod drive mechanism. This change eliminates the bellows seal and adds a "dash-pot" seal combination which will both seal the penetration and act as a snubber during rod scram. Control rod materials under consideration are boron carbide and uranium carbide. Safety rod materials being considered are boron carbide overlayed with TD nickel.

HTLTR Shield Design. Applied Physics Operation requested that MAC calculations be performed for evaluation of the HTLTR vault wall thickness. These calculations will be used to supplement the approximate calculations of Vitro Engineering. Preparation of MAC input data has been completed, and the problem is ready to be run on the computer.

HTLTR Mockup. The evaluation of cladding and structural materials for use in the HTLTR was continued. Mechanical test data for the candidate alloys after exposure at 1000 and 1200 C in a simulated reactor environment of nitrogen gas and graphite have indicated that the chromium-bearing alloys become embrittled and undergo a pronounced decrease in room temperature tensile properties. Because of this, Hastelloy X and Inconel 625 have been dropped from further evaluation tests. Testing is continued on Nickel A, TD Nickel, Hastelloy B, and Inconel 600.

These alloys have been placed in a test chamber for exposure to the nitrogen gas-graphite environment at 1200 C for 1000 hours. Included

in this test is a capsule of UO_2 clad in nickel and specimens of the carbon felt and structural ceramic which will be used in the HTLTR.

5. EBWR Fuel Elements

EBWR Plutonium Fuel Element Fabrication. Preparation of EBWR fuel (U^DO_2-1.5 wt% PuO_2) is proceeding. The mixed oxides are being compacted to greater than 98% TD by high energy impact, then crushed, sized, and vibrationally compacted into Zircaloy tubes.

Preliminary studies have indicated that the moisture content of air in the glove box hoods of the new vibrational compaction facility is appreciably increased from in-leakage of room air. To reduce this moisture content, the hoods were mechanically sealed, and silicone rubber sealant applied around the Plexiglass windows. The initially high leak rates of these hoods were reduced considerably by this operation. Installation of an air drying system is being considered as a means of eliminating the moisture problem.

Cladding Alloy Procurement. A shipment of 561 EBWR cladding tubes was received, bringing the total obtained to 2102. Of 752 tubes completely tested, 63 rejects were found and returned to the vendor. Reworking operations at HAPO are necessary before 789 tubes with internal burrs and scratches can be adequately tested by ultrasonic methods. A centrifugal hone was designed and fabricated to accomplish this reworking. Ultrasonic inspection of a trial run is being conducted to evaluate the honing process. The inspection of these reworked tubes and of the 561 newly arrived tubes will be delayed until the current inspection of PRTR cladding has been completed. An additional 600 tubes were added to the order to insure that enough acceptable tubes are available for EBWR requirements in the event the burred tubes cannot be repaired, as well as providing material for special test elements.

An order was placed with Bridgeport Brass for 500 tubes, 0.750" OD by 0.680" ID, for use in fabrication of PRTR fuel rods by swaging. Delivery is expected during December 1963.

The remainder of the cladding needed for fabrication of 12 Mark-II, nested tubular, PRTR fuel elements was ordered after receiving revised bids. Delivery is expected by March 1964.

Pneumatic Impaction of UO_2-PuO_2 for EBWR. Samples of depleted UO_2 to be used in EBWR fuel elements were pneumatically impacted for thermal diffusivity specimens. Three different initial particle

sizes were used, -65 mesh, -200 mesh, and -325 mesh. The -65 mesh size fraction is typical of that to be used for EBWR fuel. Sieve analysis of typical -65 mesh UO_2 is as follows:

Tyler Sieve Size	Wt% UO_2
-65 +100	17.6
-100 +150	18.1
-150 +200	13.9
-200 +325	13.4
-325	37.0

Densities of the impacted -65, -200, and -325 mesh specimens were 10.80, 10.88, and 10.82 g/cc, respectively. Oxygen/uranium ratios were 2.014, 2.031, and 2.042, respectively.

Mixtures of -65 mesh, -200 mesh, and -325 mesh depleted, fused UO_2, with 1.5 wt% -325 mesh PuO_2, were pneumatically impacted at 1200 C. Alpha autoradiographs of samples of each mixture were prepared and made available to the Reactor Engineering Development personnel for use in estimating the time delay in the Doppler effect resulting from inhomogeneities in mixed UO_2-PuO_2 fuels.

6. Other Off-site Customer Work

Irradiations of UO_2-Tungsten Cermet Plate. A tungsten clad, UO_2-W cermet fuel plate was evaluated. A plate temperature of approximately 3000 C, as indicated by tantalum temperature monitor wires, was attained during a 4-hour irradiation. Metallographic examination revealed thermal etching of the tungsten cladding and vaporization of UO_2 particles from near the surface of the plate during irradiation.

A second plate was irradiated using a different capsule size and radiation shield geometry to raise the temperature of the fuel.

Electron Beam Evaporation of Gold. High purity gold has been evaporated using the new electron beam evaporator. The purpose of the evaporation was to determine the operating procedures necessary to bond gold films to copper. The gold coated copper will be used in diffusion and irradiation experiments. Uniform films, 0.0003" (7.6×10^{-4} cm) thick, were deposited on the copper samples in a vacuum of 1×10^{-6} Torr. Plating time was 45 minutes at a power level of 7.5 Kw, with the substrate 10 inches from the source. Gold foils of the same thickness were stripped from nearly stainless steel substrates. Measurements on these foils indicate that with

the present configuration, uniform film thicknesses (no detectable variation) can be maintained over an area of approximately 4 inches square. This area could be increased with appropriate substrate geometries and placement.

Pneumatic Impaction of Cermet Fuels. A new method of loading a grid assembly was used with very gratifying success on the most recent 80 vol% W-UO_2 grid. The 1/8-inch square bars which form the mandrel were held and aligned at the bottom with a broached holding plate. Vibration was used to load the cermet powder easily and uniformly to an initial packed density of 60-75% TD. The top ends of the bars were held by another broached plate. The assembly was contained in a 2.5-inch OD, 0.025-inch wall impaction can with a 0.090-inch sleeve insert in the center of the cermet portion of the can. After impaction and machining, a 2-1/8-inch diameter, 2-1/16-inch length fuel element was obtained. Can wrinkling was minimized, and no apparent distortion of the cermet grid occurred during the impaction of the initial assembly from a 4-inch length to 3-3/8 inches. Visual examination of the cermet revealed no segregation of the cermet powders through the use of the vibratory compaction technique of loading. Several other 1/8-inch hexagonal honeycomb grids also were fabricated, using different loading techniques.

Fuel Fabrication. A can and die assembly for impacting cermet plates was tested. Some bonding was achieved between tungsten foil and tungsten-UO_2 cermet. Stainless steel foil which was intended to keep the tungsten from bonding to the mild steel spacer plates, actually bonded to both the mild steel and the tungsten.

Off-site procurement of tungsten vapor deposition work was initiated. An order was placed for the coating of hexagonal steel rods and circular steel plates with tungsten for use in the development of a NASA cermet.

Extrusion Studies. A third composite billet of Mo and W was extruded during November, employing a lubricant can of common polycrystalline graphite rather than pyrolytic graphite. Conditions and observations from the first two extrusions are included below, for comparison.

All three of the billets were extruded at 60 in/min ram speed through 0.75-inch ID, 90° cone, H-21 tool steel dies at an extrusion ratio of 9 to 1. All three billets consisted of two concentric 0.125-inch thick W sleeves around a solid Mo core, sealed by vacuum electron beam welding in a 0.125-inch thick Mo can. The purpose of employing the two concentric W sleeves in each billet was to investigate the extrusion parameters required to obtain a good W-W diffusion

bond. The table shows extrusion conditions. The billets were preheated in argon by induction (3000 cycles/sec), and manually transferred to the extrusion container in air (about 10 seconds total time from removal to application of force for Extrusions 2 and 3; about double this time for No. 1).

Extr. No.	Lube Can Material	Preheat Temp.	Max. Force	Die Coating
1	Pyro Graphite	2200 C	170 tons	None
2	Pyro Graphite	1800 C	210 tons	ZrO_2
3	Normal Graphite	1800 C	210 tons	ZrO_2

The surfaces of the extrusions varied from extremely poor (rough) on No. 1 which severely washed the uncoated die, to very good on No. 3. The two coated dies used for Extrusions 2 and 3 stood up very well, with only 0.010 to 0.015-inch wear in the throat, and appear good for subsequent use. The pyrolytic graphite apparently did not flow properly to produce a smooth surface. It broke up in large pieces which went through the die and imbedded in the surface of the extrusion. It was noted on all three extrusions that imperfections in the outside Mo surface were much less apparent in the surface of the tungsten.

Partial bonding at the W-W interface occurred in all three extrusions, beingly nearly complete in Nos. 2 and 3, but less so in No. 1. In all three cases bonding was improved toward the rear of the extrusion - which at this time seems an anomaly since the trailing end of the billet goes through the die at a lower temperature than the front. Evidence of this temperature difference was indicated by a significant variation in both Mo and W grain size, from large at the front to small at the rear. Also, the over-all grain size was larger in No. 1 than 2 and 3 because of the higher preheat temperature. No. 1 billet was cleaned prior to assembly by etching in HNO_3-5% HF. Imperfect rinsing or a possible billet leak during preheating may account for the poorer bond on this billet. Nos. 2 and 3 were cleaned by cathodic sputtering in argon and welded without being reintroduced into the atmosphere. It is concluded that W-W bonding can be accomplished with proper cleaning (possibly by heating in hydrogen) at 1800 C and 9 to 1 extrusion ratio. Such conditions appear feasible for the extrusion of the proposed honeycomb.

Removal of the Mo sacrificial material was readily accomplished using 1/3 each HNO_3, H_2SO_4, and H_2O.

The remaining billet of the design described above will be extruded to gain additional information on bonding and surface control. Procurement of materials for a honeycomb billet is now proceeding. This may take one or more of three routes:

1. Procure extruded W-clad hexagonal Mo rods to be assembled into a honeycomb billet.

2. Same as (1), but coated by vapor deposition.

3. Procure components, either assembled or not, for billets from which can be extruded the W-clad Mo rod for the honeycomb billet.

Some observations from metallographic examinations of the three tungsten-molybdenum coextrusions are summarized below:

Extrusion No. 1

Evidence was noted of both a surface and subsurface carbide layer along the length of the extrusion. The W-W interface was completely bonded in some areas but appeared to have separated in others. Bonding was more nearly complete near the rear. Extremely large grains were noted in both the W and Mo at the front of the extrusion, becoming smaller by a factor of 10 to 100 at the rear end. Some cracking of the W was noted in areas other than the W-W bond.

Extrusion No. 2

There was no evidence of severe W-W bond separation or W cracking after the first nine inches of the extrusion. Metallurgical bonding at the W-W interface was complete throughout most of the extrusion. The bond region was only discernible by employing 500X magnification and observing the slight amount of dirt present at the interface. The tungsten grain structure at the front was finer than in Extrusion No. 1, and was equiaxed; the rear showed elongated grains, recrystallized, and about 10 times smaller than the front.

Extrusion No. 3

Similar to No. 2 with little or no cracking at the front. The tungsten grain size was equiaxed at the front; elongated, recrystallized, and much finer at the rear. The Mo cladding

at the front showed a duplex grain size with a band of fine grains at the outer surface and large grains near the tungsten. This changed to a uniform finer grain size at the rear.

A portion of the No. 2 extrusion was swaged from 0.730-inch diameter to 0.467-inch diameter from a 1200 C air atmosphere tube furnace. The furnace temperature was gradually lowered to 1075 C during the swaging. The resulting surface was rough and exhibited numerous longitudinal cracks. However, it is believed that successful swaging of this material can be achieved through the use of a 4-die swage instead of a 2-die swage; using smaller reductions per pass; and utilizing slower feed rates.

J J Cadwell KB

For Manager, Reactor & Fuels Laboratory

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PHYSICS AND INSTRUMENTS LABORATORY

MONTHLY REPORT

NOVEMBER 1963

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

K-Lattice PCTR Experiments

The work in support of the retubing of K-pile has been completed. The results are contained in a report, HW-79325, which was issued during the month.

N-Reactor Lattice Parameter and Spectral Measurement Tests at Startup

Procurement and fabrication of materials are more than 75% complete in preparation for physics startup tests. Lattice parameters to be measured include ϵ, p, f, the neutron temperature, and the r-value of the epithermal flux. The spatial and energy dependence of the flux will be measured using Pu-Al, U-235, Al, fuel enrichment uranium, depleted uranium, Lu, Eu, Au, and Cu pins, both bare and cadmium covered. Similar tests are planned for both cold and hot core tests.

The fuel elements required for cold tests are on hand, but minor modifications remain. The fuel elements required for hot tests are nearly completed at the uranium shop. All pins except lutetium oxide have been fabricated and are nearly ready for loading into the fuel elements. The lutetium oxide pins have been ordered from a commercial supplier, and delivery has been promised within a few days.

Irradiations have been made in the PCTR mockup of N-lattice. The data from these irradiations are being analyzed to determine the relative activation rates of the various materials in N spectrum, and the dose rate to be expected from the experimental fuel element when discharged from N-reactor.

Instrumentation

Progress was achieved on the design and fabrication of mechanical equipment planned for N-Reactor physics startup tests. The required vertical and horizontal traverse tube sections were completed. Vertical traverse thimble

design was modified to permit the use of a larger neutron counter, and the water-cooled thimbles are being reworked to correct fabrication errors. Temperature and flow monitoring systems were completed in design and are being assembled. New coaxial cable was ordered to replace the original cable, which was not acceptable. The required laboratory and shop work is now 90% completed.

Installation of the primary instrumentation for the PRTR Rupture Loop was completed, and the loop conditions were monitored for background data following startup of PRTR. The background spectra and decay data were obtained for eventual reference use during later regular operation when fuel elements are charged into the loop. Difficulty was experienced initially with the flow controlling systems; however, modifications were made to provide generally satisfactory operation.

Investigations were continued to determine appropriate materials for use in fresh fuel storage casks, and data were obtained with the fuel containers empty, filled with water, and both bare and cadmium covered. This essentially completes the experimental work originally planned. Data analysis is being started, and the results will determine future experimental work.

Discussions were held with Irradiation Testing, IPD, and Design Analysis, HL, regarding the possible use of Hanford shielding facilities for irradiation of large instrument system packages. Such use appears feasible.

System Studies

The steam generator vessel portion of the N-Reactor primary coolant system simulation is functioning properly; however, the complete primary simulation with closed loop controllers has not yet been made successfully operational. Correction of a recently disclosed error in one of the controller circuits is expected to eliminate the difficulty.

The secondary loop model, which excludes the steam generators, has been formulated. It includes the main steam header and associated pressure controls, one dump condenser model, an equivalent surge tank model complete with pressure and level controls; and a condensate pump and header model. The simulation includes five control systems. The simulation circuit has been patched and scaled and is presently being debugged. Preliminary tests indicate that difficulty being experienced in a high gain integrator circuit may be readily corrected with a simple circuit modification.

The N-Reactor injection system simulation developed earlier was used to study the effects of measurement time lags on the differential pressure-pump speed control system performance. In addition, a check on the con-

troller output versus pump speed was made to confirm earlier calculations. As predicted, a relatively small change in the measurement lag (a few tenths of a second) produce significant changes in the system response to disturbances and to the maximum allowable controller gain settings. The results obtained indicate that changes will be required in the injection system controls. This study is based on a simulation which includes a fluid-drive type of pump speed control device and the characteristics of this device are not accurately known. Therefore, before any changes are made, it would be advisable to obtain actual operating data on the installed system. A request was made to N-Reactor Field Engineering Operation to consider the inclusion of appropriate tests in their forthcoming injection pump testing program.

A decision was made to build a special control system simulation panel for use with the N-Reactor simulation studies. Use of this panel will release approximately 30 integrators on the analog computer for other circuitry. Present plans are to build groups of standard plug-in circuits for rate limiters and proportional-plus-reset and proportional-plus-rate controllers which can be used with low-cost unstabilized operational amplifier manifolds. The panel will include the necessary initial condition and readout circuitry for synchronization with the analog computer and will be capable of accelerated time base and repetitive operation. A requisition has been processed for the operational amplifier manifolds.

The primary flow system for the N Reactor was simulated with the MIDAS program. Equations derived from previous runs were checked out. The simulation showed substantial agreement with previous runs with the simulation done on the EASE analog computer.

Using the Boonshaft and Fuchs transfer function analyzer measurements on the two-dimensional reactor simulation were made. These measurements were used to calculate coupling coefficients between different nodes. This was accomplished by making Nyquist and Bode plots for different node combinations. Data are now being analyzed.

Consultations were provided NRD on the proposed N-data system (HW-78939), rupture monitor equipment, process tube monitoring, and the raw water emergency cooling system.

SEPARATIONS

Critical Experiments with PuO_2-Plastic Mixtures

Critical mass experiments were continued with the Remote Split-Table Machine. Fourteen critical experiments were completed with PuO_2-polystyrene compacts

as the fuel. Comparative data were obtained for the relative reflector savings of 8-in. of water to that of 8-in. of Lucite, and of Lucite in various thicknesses from 1-8 inches. Other reflector combinations studied consisted of 0.06-in. stainless steel and Lucite, 0.03-in. cadmium and Lucite, 0.19-in. stainless steel (0.3 w/o gadolinium) and water, and an 8-in. thick reflector of 3% enriched uranyl nitrate solution at $\sim$ 374 g U/ℓ. In each of these experiments, the cross sectional dimensions of the rectangular prism core were 12-in. x 12-in.; the critical length was determined for the various reflector combinations above, with the reflectors being located only on the end(s) of the fuel core.

The experiment with the gadolinium steel plate and water, and that with the uranium solution reflector were performed to obtain data applicable to the proposed design of a dissolver for uranium; a design in which enriched fuel elements ($\sim$ 2% U^{235}) would be placed in a gadolinium steel basket during dissolution.

Some interesting results have been obtained from these experiments:

1. Although the density of hydrogen is less in Lucite than water (5.78×10^{22} atoms/cc for Lucite to 6.68×10^{22} atoms/cc for water), Lucite was found to be a slightly better reflector than water (δ Lucite - δ $H_2O \sim 0.6$ cm). This is due to the fact that the number of carbon and oxygen atoms in Lucite is nearly equal to the number of hydrogen atoms in Lucite. These atoms, having a scattering model more nearly isotropic than hydrogen, tend to reduce neutron leakage for neutron energies above 0.01 MeV.

2. The reflector savings of a 6-in. thick layer of Lucite is effectively the same as that for an "infinitely" thick layer.

3. With a 0.03-in. thick cadmium sheet positioned at the Lucite core interface, the reflector savings were $\sim$ 0.9 cm, or less than that of a 1-in. thick Lucite reflector for which δ was $\sim$ 1.69 cm. The cadmium, when used in this manner, is thus not only effective for Pu solutions (previous solution data), but for intermediate spectrum systems corresponding to Pu precipitates or PuO_2-slurries having an H/U ratio of $\sim$ 15.

4. The 0.19-in. stainless steel plate containing $\sim$ 0.3 w/o gadolinium was less effective than expected (there may be some question concerning its gadolinium content).

The analysis of the experiments with the uranium solution reflector has not been completed.

Measurements were also carried out to evaluate the effect of the rubber coating used on the PuO_2-plastic blocks for contamination control. Critical experiments were conducted with a non-reflected array (~12-in. x 12-in. base) with added amounts of rubber inserted between each fuel cube. For the non-reflected array, an increase of about 5% in the critical mass (33.7 to 35.5 kg Pu) was indicated by extrapolating the data to zero thickness of rubber coating.

Pulsed neutron source measurements were taken concurrently with the experiments on plutonium-polystyrene compacts. Results which were obtained showed the neutron lifetimes to vary from 1.36 μ sec for an unreflected 12" x 12" x 12.54" array to 6.71 μ sec for a 12" x 12" array reflected on one end by Lucite. Measurements were also made of the reactor noise in several cases. The results of these measurements compare very well with those obtained on the same arrays utilizing the pulsed neutron source technique.

The Poisoning Effect of Gadolinium when Placed Adjacent to a Two Percent Enriched Uranium-Water Lattice

Calculations were made to determine the worth of gadolinium as a poison for a uranium-metal dissolver containing uranium fuel elements at an enrichment of 2 w/o U^{235}. The dissolver's radius was large enough so that the calculation could be carried out as an infinite slab in two dimensions. This is a reasonable approximation since any neutron which escapes two or three mean-free paths from the system is essentially lost from a reactivity standpoint. The concrete was of sufficient thickness so as to approximate an infinite reflector. The thickness of the dissolving region was 6.5 inches for most of the calculations but 6.0 inches was used occasionally. The gadolinium-stainless steel (0.3% Gd by weight) thickness was varied from zero to 1.25 inches. The reflector consisted of UNH solution with uranium densities of 500 g/ℓ and 600 g/ℓ (2 w/o U^{235}). Some calculations were also carried out using water as the reflector.

Cross sections were obtained from the GAMTEC lattice parameter and cross section code. Twelve energy group macroscopic cross sections were obtained for each region and these cross sections were used in the HFN diffusion code for the k_{eff} calculations. The infinite multiplication factor has its largest value for a two percent enriched uranium-water lattice with a rod diameter of about 0.6 inches and a water-to-uranium volume ratio of about 2.7. Therefore, this type of lattice was used in all the calculations. The infinite multiplication factor for the two percent enriched lattice was found to be 1.390 by using the HFN diffusion code. This is conservative when compared with the value of 1.366 as obtained by the GAMTEC code.

The basic results of the calculation are:

(1) When UNH solution with a uranium density of 600 gm/ℓ is used as a reflector instead of water, a reactivity increase of about 65 mk results.

(2) When the uranium density of the UNH solution is increased from 500 gm/ℓ to 600 gm/ℓ, a reactivity increase of about 8 mk results.

(3) The gadolinium-stainless steel sheet has its greatest poisoning effect with a thickness of between 0.125 inches and 0.500 inches. If the gadolinium-stainless steel sheet is made too thick (greater than one-half inch), a large percentage of the high energy neutrons are reflected back into the lattice by the steel with a resulting increase in k_{eff}.

(4) The total worth of the gadolinium-stainless steel sheet at the optimum thickness is about 67 mk for the system reflected by the 600 gm/ℓ UNH solution. Therefore, about all that can be gained from the gadolinium is a cancelleation of the reactivity effect of the uranium in the water reflector.

(5) A reduction in k_{eff} of about 32 mk occurs when the width of the dissolving region (uranium-water lattice region) is reduced from 6.5 inches to 6.0 inches.

In summary, these results indicate that this method of poisoning the system is not too effective. The gadolinium-stainless steel sheet will help some if the optimum thickness is used.

There are a number of safety factors "built into" the calculations: 1) Optimum rod diameter and optimum water-to-uranium volume ratios in the uranium-water lattice, 2) the poisoning effect of the acid was neglected, 3) water was used as the moderator in the uranium-water lattice but in general there will be a certain amount of uranium in the water which would decrease k_{eff}, and 4) the maximum density of the uranium in UNH solution was taken to be 600 gm/ℓ but this is probably greater than the actual densities that will be encountered.

Slightly Enriched Uranium-Water Moderated Lattices - Correlation of Experiment and Theory

A new correlation between theory and experiment for slightly enriched uranium-water moderated lattices was begun that will utilize an η averaged over an equivalent Wigner-Wilkins spectrum for the lattice in question.

C. L. Brown previously obtained a correlation by adjusting η to fit the experimental data; calculations of other parameters were carried out with the IDIOT code which contains several outmoded theoretical models. These outmoded models were replaced by more current models which have been included in the GAMTEC code.

Calculations were made for 0.387-in. diameter rods having enrichments of 1.027, 1.143, and 1.299 w/o U^{235}, corresponding to BNL experiments. Calculations also were made for several rods of 3.06 w/o U^{235}; the rod diameters were 0.30, 0.60, and 0.926 inches, corresponding to Hanford experiments.

The calculations completed to date agree quite well for the above enrichments and rod sizes. The results, with few exceptions, fall within the experimental errors (approximately 4-6%); those that do not only slightly exceed the expected uncertainties. Over-all, the calculations thus far performed agree well with the experiments; this being accomplished without any major empirical adjustments to force a fit.

Also under study are theoretical methods for calculating the criticality parameters for slightly enriched uranium rods in a uranyl nitrate solution. The results will be applicable to the problem of dissolving slightly enriched uranium in chemical processing operations.

Criticality of U^{235}, U^{238}, Pu^{239}, and Pu^{240} in Unmoderated Systems

As reported last month, a series of Monte Carlo calculations of k_∞ were made for systems containing only one of the isotopes U^{235}, U^{238}, Pu^{239}, and Pu^{240}. An error has since been found in the fission spectrum built into the Monte Carlo code which changes these results slightly. The error (key punch) caused all the neutrons from fission to originate with energies greater than one MeV. New calculated values for the infinite multiplication factors are: 2.344 instead of 2.382 for U^{235} metal; 2.733 instead of 2.785 for Pu^{239} metal; and 2.284 instead of 2.561 for Pu^{240} metal. The values for U^{235} and Pu^{239} are thus only slightly different, but the Pu^{240} system is now calculated to have a slightly smaller value of k_∞ than the U^{235} system (the critical mass is estimated to be about equivalent to that of U^{235}). The revised value for U^{238} is not yet available.

Series Expansions for Theoretical-Experimental Flux Comparisons

One possible means of comparing theoretically calculated fluxes with those in an experimental configuration is to expand the experimental flux in a series as

$$\overline{\Phi}(E) \simeq \overline{\Phi}_o(E) \left[B_o + \sum_{k=1}^{n} B_k X_k(E) \right]$$

where $\overline{\Phi}_o(E)$ is the calculated flux normalized in magnitude by the constant B_o. With this consideration, the $X_k(E)$'s become a set of correction functions of magnitude B_k to be applied to $\overline{\Phi}_o(E)$ to yield the experimental flux $\overline{\Phi}(E)$. By irradiating foils in the experimental flux and calculating their reaction rates (R_m for the m^{th} foil material) then

$$R_m = \int_o^\infty \overline{\Phi}(E) \sigma_m(E) dE$$

$$= B_o \int_o^\infty \overline{\Phi}_o(E) \sigma_m(E) dE + \sum_{k=1}^{n} B_k \int_o^\infty X_k(E) \overline{\Phi}_o(E) \sigma_m(E) dE$$

where $\sigma_m(E)$ is the activation cross section of the m^{th} foil material. If the experimental irradiation were to include n foil materials (assuming the $\sigma_m(E)$ to be linearly independent) then n equations and n unknowns (B_1, B_2, ---, B_n) would result and the magnitudes of the correction functions could then be found.

The problem of finding the appropriate set of correction functions, $X_k(E)$, is of importance and two separate types are now under consideration. Both methods have enough strength to warrant further investigation. One type is to let the $X_k(E)$'s be an infinite set of orthonormal functions and allow this series to be terminated after n terms. Under this consideration, from the definition of orthonormality (it is equivalent to a least squares fit), the series will be converging and the rate of convergence will be designated by the type of orthonormal function used. Another type of correction function under investigation is a set of n step functions defined over the non-zero $\overline{\Phi}_o(E)$ range such that

$$X_k(E) = 1 \qquad (E_k < E < E_{k-1}).$$

If $\overline{\Phi}_o(E)$ were similarly represented as a set of step functions then $\overline{\Phi}(E)$ could be solved for exactly (disregarding cross section and experimental

Buckling of Partially Filled Spheres

Several remaining small bugs in the program to calculate the buckling of truncated spheres were found and corrected. The number of fit points used for derivative approximation were limited to near points to prevent the introduction of large cancelling contributions from distant points. This change introduces no substantial error, but has limited the complete generality of the formulation.

As a result of these changes the code is now operative. It is being used to produce the curve of B^2 vs. h (with unit radius) for truncated bare spheres. A description of the work, with results, is being prepared for possible journal publication.

INTERSET - Interaction Code

The code was improved to allow solution of problems in which the matrix of interaction does not converge with the normal eigenvalue iteration technique. Specifically, when an array consists of cylinders in a line, the interaction matrix will have equal and opposite roots and will not converge in the analogous "root squaring" process for a polynomial. Since the eigenvector is independent of the size of the eigenvalue, a simple transformation may be made as follows:

$$\underline{\underline{P}}\ \underline{\phi} = \lambda\ \underline{\phi}$$

$$\underline{\underline{P}}\ \underline{\phi} + \epsilon\ \underline{\underline{I}}\ \underline{\phi} = \lambda\ \phi + \epsilon\ \underline{\underline{I}}\ \underline{\phi}$$

$$(\underline{\underline{P}} + \epsilon\ \underline{\underline{I}})\ \phi = (\lambda + \epsilon)\ \underline{\phi}$$

$$\underline{\underline{P}}'\ \phi = \lambda'\ \phi$$

The matrix $\underline{\underline{P}}'$ will be convergent if ϵ is large enough. If ϵ is too large the iteration will take longer. If λ represents the albedo, a good number for ϵ is .2, since .25 is the maximum interaction between two cylinders. If λ represents k, a good number for ϵ is 1 (i.e., critical). One input number in INTERSET has been added to enable solution of any linear array.

A formal document is being prepared.

Expansion of the Plutonium Critical Mass Laboratory

The design criteria for the addition to the Critical Mass Laboratory were approved by RLOO on October 25. A project proposal, prepared by Facilities Engineering, requesting these design funds has been submitted for AEC approval. The proposed addition will include facilities for decontamination, a storage vault for fissile materials, a mockup room for experimental assemblies, and other needed laboratory and office space.

Instrumentation

Preliminary work began to develop an on-line noise analysis system for the Critical Mass Laboratory. This system will be used in measurements of β/ℓ and k_{eff}. A prototype system is being developed to plot the transfer function for the assemblies with eight different frequency components. The number of frequency components will be increased in the final system. The development of eight RC parallel-T filter networks and their use in the feedback loop of eight operational amplifiers is now in progress.

A method was developed to vary the pulse rate of a pulse generator exponentially. This will be used in testing the period meter circuit in Channel #2. This channel is comprised of a log count rate meter and period amplifier. The response time and saturation effects of this channel will be studied. Satisfactory results have been obtained in the preliminary tests performed to date.

Consulting Services on Nuclear Safety Hazards

1. Nuclear Safety in HL

   Comments were submitted to PRTR Technical Planning concerning the storage of used ion exchange columns in the 309 Building ion exchange vault. The spent columns will normally contain very little plutonium; nevertheless, the mass limit per column is 230 g. For vault storage limits, two alternates were proposed: 1) Twelve isolated columns, each containing 230 g of plutonium (max.) and each column isolated from the other columns by 10 inches of concrete; or 2) twelve columns in four rows of three, with 230 g (max.) per column and 460 g (max.) per row, and each row isolated from other rows by 10 inches of concrete.

   A suggestion to place a neutron poison in the PuO_2-UO_2 pellet centerless grinder coolant tank in the 308 Building was reviewed for Fuels Testing and Analysis. Nuclear safety in this tank is presently assured by an accountability check system and a conservative mass limit. The neutron poison would be a secondary control. Two poisons are under

consideration: A boron steel grid, and pyrex raschig rings. The grid would be one-inch squares of 1 w/o boron steel plates, about 1/8 inch thick. The raschig rings would be the one-inch size containing 4 w/o boron.

A storage array for EBWR fuel rods in 309 Building was reviewed for PRTR Technical Planning. These rods are:

0.372-in. diameter,
1.5 w/o PuO_2 in UO_2,
0.22 w/o U^{235} in the uranium,
8 w/o Pu^{240} in the plutonium, and
about 88% of theoretical density.

The storage arrangement will be a single layer array of six boxes mounted on a wall of the PRTR cell. There will be 100 rods per box in a "close packed" bundle. In the close-packed arrangement, the rod bundles are critically safe in any arrangement. At optimum spacing between rods, however, 600 rods represent about 1.7 critical masses.

Four new nuclear safety specifications were issued:

B-7 - Storage of Fissile Materials in Buildings 305-B, 326, and 3731.

B-8 - Storage and Handling of 35 w/o U^{235} Enriched U-Al Alloy in 305-B and 3731 Buildings.

K-7 - Processing 1.61 w/o U^{235} Enriched Uranium in 306 Building.

M-2 - Storage and Transporting of Mark I PRTR Fuel Elements in the PRCF, 309 Building.

2. Nuclear Safety in CPD

Service was provided as requested on a Criticality Hazard Review Committee in connection with the fire that occurred in 233-S Building on November 6. Participation on this committee will continue until Redox processing operations are restored to normal.

3. Nuclear Safety in NRD

A nuclear safety review to cover the fabrication of 1.25 w/o U^{235} enriched NPR fuel elements in the 333 Building was begun. Four review

sessions have been held to date. The review will cover the 333 Building process step by step, and considerations will be given to all equipment and all areas of the building. To support this review, the necessary critical parameters were calculated.

4. Nuclear Safety Training and Education

Course B, Group I lecture series in nuclear safety was completed on November 7. The course covered eight sessions. The second group of Course B will begin on November 21. The "B" series is oriented toward the non-technical supervisors and specialists who have need for further understanding of the bases for nuclear safety procedures and operating limitations.

NEUTRON CROSS SECTION PROGRAM

Scattering-Law Measurements for H_2O at 95°C

The triple-axis spectrometer was inoperative most of the month because of delays in getting the spectrometer shielding reassembled following the installation of the step plug for time-of-flight work in the 4-B hole. A series of measurements were completed on the elastic scattering from vanadium to calibrate the detection efficiency of the analyzing spectrometer. Measurements are now in progress on H_2O samples at 95°C of different thicknesses which are being made to study the effects of multiple scattering. Work has continued on the organization of a Monte Carlo program to calculate multiple-scattering effects in scattering-law measurements.

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. Neutron reflectivity measurements were made on possible monochromating crystals for this spectrometer. One, a beryllium crystal ingot 1-in. diameter and 4-in. long, proved to have two approximately equal reflections separated by 3 degrees in angle. Although the indicated mosaic was about 0.13 degree the integrated reflected intensity was only about one-half that of the copper crystal reported last month. Measurements were also made on a Hanford Al crystal No. 12-B for calibration purposes. The Al No. 12-B reflectivity was about two-thirds that of the Be crystal. The high voltage breakdown problem in RIDL preamplifiers was solved for 15 units by improved insulation and replacement of defective connectors. Six units will be returned to the manufacturer. Analysis of detector-shielding experiments was completed and a manuscript prepared on

this work. Measurements were continued on time-jitter in He^3 proportional counters for different gas fillings.

Fast-Neutron Cross Sections

Another series of measurements of neutron total cross sections from 3 to 15 MeV was completed during the month. Repeated measurements were made on Sc, Eu, Tm, and Lu and first measurements were made on samples of Ru, Rh, Re, Os, Ir, and a separated isotope - Cr^{53}. No data previously existed for these samples. The performance of the accelerator was very poor for these runs but the time-of-flight equipment performed well. The results of total cross section measurements of seven additional elements were transmitted to BNL and LRL for inclusion in data compilations.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Approach-to-Critical Experiments Using High Exposure Pu-Al Fuel

The 0.80" lattice has been reloaded to make measurements on the worth of the top template (aluminum) in the lattice. Also, buckling measurements will be made for a simulated bare assembly, i.e., the reflector will be isolated by 0.040" of cadmium around the curved surfaces of the cylindrical assembly. The effects of varying the thickness of the reflector were measured by lowering the water level as multiplication data were taken. The change in the multiplication caused by changing the reflector thickness was less than 0.1% for a reflector thickness greater than 5". The worth of an 80 gram Pu-Be neutron source centrally located, but spaced 4" from the end of the assembly, was measured to be less than the worth of one fuel rod in the outer ring.

Low Exposure PuO_2 Lattice Studies

Ceramics Research has resumed work on fuel for these experiments. Pellets that were previously sintered are now being circumferentially ground to size and end ground to a flat surface. Thus far, 1000 pellets have been successfully ground. It is estimated that there will be 1700 good pellets after all have been ground. These will make about 44 inches of a 19-rod cluster, enough for the central cell. A PCTR loading requires about 324" of cluster. These pellets are 1/2" in diameter, approximately 1/2" long, and contain 0.90 w/o Pu which has a Pu-240 content of 7.56 a/o. Alternate methods of fabrication of the remainder of the core are being considered.

Experiments in the PRCF

An informal report on the reactivity measurements of mixed oxide fuel elements in the PRCF has been completed.

The PRCF was shut down, as scheduled, on November 1, for conversion to H_2O moderator. The shutdown terminated 7-1/2 months of successful operation. During this time, startup experiments and other post-start-up experiments were completed. The capability of the facility for measurements of irradiated fuel elements was successfully demonstrated and excellent short- and long-term stability was observed. An important fact from an operational standpoint is that loss and degradation of the D_2O moderator were minimal.

A final draft report on the initial experiments is about completed.

Preparations have begun for the experiments in the PRCF which use H_2O moderator. The experiments have been outlined and problems which must be solved before experiments with Pu-Al fuel can be accomplished have been listed. Approval to operate the PRCF with H_2O and EBWR fuel has been obtained from the AEC.

Since no fully automated reactivity lifetime calculations can presently be carried out, with the required detail, for the Phoenix fuel reactor cores (Mark I, II, III), extensive desk calculations and considerable card handling is required. For many of the Phoenix reactor calculations, rather crude time steps have been used to reduce labor and cost. In a brief study, just completed, the effect of halving the time steps from $\sim$4000 efph to 2000 efph has been investigated for a specific Phoenix core. The shorter time step analysis results in a more rapid reactivity loss with time than the longer time step analysis. Comparison of these results with very early studies, in which beginning-of-life cross sections were used for the entire life cycle, suggest a rather rapid convergence of the k_{eff}-time curve.

The standard reactivity calculation method used in the Mark I parameter survey is to obtain microscopic cross sections from the TEMPEST and GAM codes, and then to use them in the HFN multi-group code. The Pu-240 resonance is isolated in a single group from 0.683 to 2.38 eV. As a check on the method, cross sections for this group were calculated using the THERMOS code. Both the GAM and THERMOS libraries were obtained from the RBU Basic Library at 0°K. For a typical problem, the standard method produced a k_{eff} at 1.0622, and a microscopic Pu-240 burnup to buildup ratio of 2.969. The same set of four-group cross sections with the exception of

the third group (0.683 - 2.38 eV) which was obtained from THERMOS yielded a k_{eff} of 1.1115 and a burnup to buildup ratio at 2.390. This difference is attributable to the effective cross section change of Pu-240 from 2097 to 1514 barns. The GAM code provides only four-energy groups across the resonance while the THERMOS calculation uses twenty-six. The finer energy mesh provides a more accurate picture of the flux depression which, in general, lowers the average cross section in the resonance. This work is being extended to a re-examination of the thermal group, and preliminary calculations indicate even further deviations in k_{eff}, but somewhat smaller differences in the burnup to buildup ratio.

An additional deficiency in the initial MTR analysis has been observed. Thermal group cross sections were calculated using THERMOS with all 30 allowed "speed-groups" disposed up to the defined thermal group upper energy limit. This forces the simple assumption of a 1/E flux distribution as the slowing down source. With a significant amount of Pu-240 present in the problem, some provision should be made for the resonance flux dip and its influence as a slowing down source on the thermal spectrum. Tests are being made to determine the number of speed groups which should be assigned to the resonance group for source purposes.

Tentative fuel specifications are being prepared for Phoenix experiments in the split table assembly.

k_∞ calculations using the PHYSICS CHAIN are under way for a series of Be-Pu systems which range in Be/Pu ratio from 20 to 5000, and in temperature from 293 K to 3000 K. The plutonium compositions considered are 5 and 20 a/o Pu-240, and our standard "reactor grade" which consists of 50.03% Pu-239, 31.18% Pu-240, 16.59% Pu-241, and 2.21% Pu-242.

In addition to the above systems which contain only Be and Pu, calculations have been carried out on "reactor" systems which include sodium coolant and iron structure. This reactor type has been designated Mark IV. It contains 5 v/o iron and at least 50 v/o beryllium. The remaining composition is made up of sodium which is varied from 0 to 45 v/o. It is assumed that the fuel occupies no volume. Core sizes considered are 250, 500, and 1000 liters with fuel loadings at 0.1, 0.2, and 0.3 kg of Pu/lit. The plutonium is of the "reactor grade" mentioned above. The results of these calculations are presently being analyzed.

The Modified Heavy Gas Equation and Water

In the case of water, neutrons slowing down into the thermal region produce a source nearly uniform in energy. Consequently, to study the suitability of the modified heavy gas equation with water as a moderator, it is necessary

to consider a source term. The numerical techniques for including a source term have been developed following those for the modified gas equation without a source. These techniques have been incorporated into program SMGAS, which solves the modified heavy gas equation with a source. Currently, program SMGAS is undergoing final debugging.

Thermalization and RBU

Debugging is nearly complete on the revised SIGGAS subroutine in program SPECTRE S. As a test of the programming, this subroutine has reproduced results of program IDEAL GAS for both upscattering and downscattering in appropriate cases. Studies on the effects of varying the temperature and mass of scatterer as a function of initial neutron energy have been started.

PRTR Burnup Studies

Fuel elements 5092 and 5111 have been processed through the PRTR Gamma Scan Facility. Eight rods from fuel element 5111 and two rods from 5092 were scanned. Element 5092 is a physics test, Pu-Al element which has had complete chemical and mass spectrometry analysis performed on three of the rods. It was hoped that the gamma scan would aid in predicting the variation of burnup along the rod, but the data were not of sufficiently high quality. The results of the scanning of element 5111 were not entirely satisfactory, either. Some difficulties have occurred in positioning the rods with respect to the scanning head, which makes it difficult to obtain consistent data. Measures to improve the situation are being considered.

The results of the burnup determinations are being carefully scrutinized by members of the Chemical Laboratory in order to isolate and solve some problems encountered in the collection of data from fuel element No.'s 5092, 5095, and 5051. The scrutiny will continue for subsequent fuel elements.

Code Development

RBU

The burnup calculation on the PRTR 19-rod cluster, using RBU, is now ready for the computer. Owing to the complexity of the problem and the limited availability of machine time, progress on this study has been slow. Concurrent with the above work, the checkout of the two-mass approximation to the scattering law of bound scatterers is continuing.

RBU Basic Library

The inelastic spectrum tables in the RBU Basic Library are apparently in error. In all $(n, n'\gamma)$ inelastic scattering processes, the neutron energy losses are equal to the excitation energies of the target nucleus. Several of the neutron energy losses reported for various nuclei in RBU are not in agreement with experimental data for the excitation energies of these nuclei. Furthermore, some of the probabilities of a neutron suffering a given energy loss are questionable. A careful scrutiny of the situation is being undertaken.

COMBO

This code, though primarily designed to combine and simplify TEMPEST and GAM input, is being expanded to occupy a more important role in the PHYSICS CHAIN. Minor changes and revisions are still being made. Debugging continues.

HFN Modifications

HFN-II is now being modified to allow homogenization with the critical flux from a search calculation.

Transport Analysis Improvement

To alleviate computer memory-capacity limitations being encountered in radiation-damage studies conducted by the Materials Development Laboratory, Hanford's multi-energy transport analysis (GE-HL program S-XII) was extended to continually wheel about half of the working storage in and out of the fast memory, by the use of opposed-phase twin channel magnetic tape shuttles running in parallel-logic simultaneity with core calculations. This modification nearly doubles the maximum problem size. Check-out runs have indicated that the parallel-logic shuttles have no measurably significant effect on calculational speed, in accordance with expectations, and that the super-density tapes (800 bits/inch) impose no discernible reliability limitations at this intensive level of utilization (contrary to expectations).

CLERK

The CLERK code document, HW-75521, has been distributed. Binary decks complete with sample problems are available for the author.

PHYSICS CHAIN

An incompatibility between the GAM data tape and the GAM code that is currently being used by the PHYSICS CHAIN has been removed. Cases using U-238 or Th-232 were the only ones influenced since the bug occurred only in the special handling of resonance isotopes, and since this version of GAM recognizes only these as resonance isotopes. Users of the PHYSICS CHAIN need not be concerned over possible error in output from cases run in the past due to this bug unless they were using U-238. Output from the GAM portion of the chain now agrees to eight places with output from the "model" GAM code.

HRG

The Hanford Revised Version of GAM has been made compatible with the PHYSICS CHAIN, and is now operating as one of the Part IV chains. Since HRG expects 180 isotopes in its library rather than the 131 available in the old GAM library, it was necessary to make a special Composite Library Tape containing this 180 isotope library. Caution must be taken in using the Part IV PHYSICS CHAIN to insure that the program has the proper C.L.T.

Isotopic Analysis of PRTR Samples

Isotopic analyses were provided on 15 samples of PRTR-irradiated fuel elements in support of the Plutonium Recycle Program. Of these 6 were plutonium samples from element No. 1101, 2 were plutonium samples from element No. 1501, and there were 5 uranium and 2 plutonium samples from element No. 1006. Analyses were also provided on 4 samples supplied as tests of chemical reagent purity and of chemical process yield.

The high uranium content of plutonium samples derived from uranium-oxide fuel elements, reported last month, has been satisfactorily reduced. Difficulty occurred this month in the appearance of a high ion background in the plutonium mass region which made the measurement of sample Pu^{242} difficult. A search for the origin of the interfering ions revealed very intense spectra of barium, barium-double-carbide, and barium-chloride. These spectra were also found in samples of HCl process reagent. The problem has been circumvented by eliminating the final chemical procedural step involving HCl.

Instrumentation and System Studies

Operational tests were conducted on the detector and instrumentation of the PRTR underwater gamma scanner for fuel rods and activated wires. Tests indicate that an increase in viewed incremental area of the fuel rods might

be necessary to provide adequate counting rates and hence reduce the data-taking time. To date, some mechanical difficulties have arisen which preclude keeping the fuel rod centered under the collimator. All electronic instrumentation has continued to perform correctly, and preparation was started on a complete instruction manual for the system. Minor changes are being made on the collimator to provide higher counting rates.

The spare PRTR liquid effluent gamma monitor was converted to the new circuit configuration and tested. All four monitors have now been converted and three are in continuous operation. All drawings have now been updated and preparation of an instruction manual was started.

Experiments were started in an effort to develop specialized aural signaling monitoring instrument for use with the PRTR automatic controller. Tests with several circuit configurations were carried out to facilitate development of a prototype.

Vibration data were taken at the PRTR on ten different process tubes for six different operating conditions. Five different accelerometers were mounted in different positions on each tube. Three-hundred runs of vibration information were recorded on a 7-channel tape recorder. The information is noise signals containing frequency components of 20-200 cycles. Power density curves (vibration amplitude squared versus frequency) will be plotted to determine the frequency content of each signal. These data will be run periodically to determine the effects of equipment changes on vibration and hence fretting corrosion. Two preliminary response curves have shown resonances at different points in the spectrum.

Meetings were held on October 21 and 22 with officials of the Industry Control Department, General Electric Company, in Phoenix. Subjects discussed were the programming, course content and schedule, input-output signal requirements, and general system capabilities of the new process control computer recently ordered. It was agreed that the programming course be presented at HL on the seven working days beginning December 9. Twelve Laboratory personnel have been suggested as possible course attendees. A detailed list indicating input-output signal requirements for 321 Building application has been prepared and forwarded to the Industry Control Department. A special equipment request was written and submitted to Transportation and Maintenance for a semi-trailer to house the computer.

HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

Room temperature yield strengths and metallographic examination of the materials exposed to a nitrogen atmosphere at high temperatures for 200 hours have been completed. Two runs were made, one at 1000°C, the other

at 1200°C. The three materials that show greatest promise for use in HTLTR are nickel A, TD nickel, and Hastelloy B. Information on these materials includes a bend test, room temperature yield strength, metallographic examination of the microstructure before and after each exposure, and the weight gains during each exposure. The nickel A sample heated adjacent to graphite at 1200 C showed some carbon transport, a slight lowering of room temperature strength, and no change in ductility. TD nickel showed no change in microstructure at 1000 C, but some grain growth at 1200 C, and possibly agglomeration of the ThO_2. Hastelloy B retained some ductility even in the 1200 C test and showed no grain growth in either test. A third test of these materials and others has been started. It is being run at 1200 C, and is to last 1000 hours. This is comparable to the expected operating time of the HTLTR at 1000 C or above during a year's use.

Preparatory to planning the details of a research program using the HTLTR, a series of calculations has been started in which the k (multiplication factor) of a variety of lattices as a function of temperature will be determined. Some initial calculations have been completed. These include lattices of uranium metal rods in graphite for which room temperature experimental data is available. A part of the planning included a meeting with the AEC, Division of Research and Development, in Germantown in which some preliminary suggestions for work were discussed.

A design and performance specification for the time-of-flight neutron spectrometer is being prepared. The HTLTR chopper will use a 16 cm diameter rotor spinning about a horizontal axis. Between the rotor and the source, which are separated by 2.9 cm maximum, will be a collimator 1.5 meters long. The rotor will spin at 120 rps maximum, and will have 16 straight and parallel slits of minimum width 0.08 cm. The effective slit width is adjustable to a practical maximum of 0.16 cm. An evacuated flight path 25 meters long will be used.

NEUTRON FLUX MONITORS

Four regenerating detector plutonium samples, irradiated to different exposures, were chemically processed to remove fission products and are now prepared for mass spectrometer analyses. The four uranium isotope samples, which serve as integrated neutron flux monitors, have not been chemically separated from fission products and prepared for spectrometer analysis because of technical difficulties. It appears a satisfactory technique has now been developed and the samples have been scheduled for separation. If the data obtained from these samples verify the theoretical calculations, the experiment will be concluded and a final report prepared.

Specifications were started to permit offsite fabrication of several in-core ionization chambers which will utilize regenerating material. Procedures were started to obtain 100 milligrams of U-234 from Division of Research, AEC, Washington. Appropriate request forms were completed and submitted.

Fabrication of new cables for use with the Boron-11 beta current detectors was accelerated, and all necessary insulators have been assembled. The four elements to be used in the scheduled tests were fabricated into usable form; these isotopes are Aluminum-27, Boron-11, Rhodium-103, and Technetium-99. All four elements are being enclosed in the same chamber and separate conducting leads are being brought out through the new cable assembly for each. In addition, the cable will contain one extra wire for direct use in measuring any generated cable noise current. The assembly will be charged into KW Reactor at the first opportunity. Calculations are being continued in an effort to better determine the theoretical beta currents to be expected from the various isotopes under given reactor conditions.

The 200 milliwatt klystron recently ordered for experiments regarding the microwave neutron flux monitor concept has been received. Assembly of the necessary hardware to permit laboratory testing of the unit was partly completed. Word was received from a manufacturer that attempts to fabricate the necessary flexible waveguide were unsuccessful and that another attempt will be made. The units failed quality control tests.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

Conversion of the multiparameter eddy current nondestructive testing device for evaluation as a tubing tester is proceeding. Initial tests have revealed several problem areas in the tubing test not present in the original multilayer test demonstration, including problems due to different pulse shapes obtained on each channel as the test scans the defected tubing irregularities, and the need for increased instrument sensitivity and stability. These problem areas are being explored using two tubing testers operating at 200 Kc and 400 Kc, and it is expected that the results of these tests will lead to the required changes in the descriptor determining sections of the multiparameter tester. The most severe of these problems is that due to the differing pulse shapes between the various channels. This effect was not unexpected and is due to the scanning technique and the necessary use of different test frequencies. During scanning of a tubing irregularity, each channel of the equipment produces an output pulse, and the descriptors to be used by the transformation section of the equipment are determined by some measure of these individual channel output

pulses. However, since different frequencies are used in order to increase the dimensionality of the signal, the pulse shape for the channels associated with the different frequencies is different. This in itself is a result of the higher dimensionality of the signal and the differing pulse shapes actually carry additional information which will be eventually of interest. However, the varying shape of the pulses presents a difficulty in extracting a single descriptor from each pulse which is needed now for applying the basic principles of the multiparameter test to the tubing test. Several methods to do this are under consideration.

A review of eddy current methods of making absolute determinations of the electrical conductivity of metals is being made. Measurements are being made in the laboratory using a single frequency method and a pulse method to determine their relative merit. The selected method will be used to measure the conductivity of Wood's metal used in a study of eddy current diffusion.

Heat Transfer Testing

Reassembly of the heat transfer testing equipment, in preparation for experiments using the new emissivity compensation method, is partially complete. It had been necessary to disassemble the equipment to allow removal of the N Reactor charging machine and magazine loader from 314 Building.

Zircaloy-2 Hydride Detection

The new stabilized eddy current hydride detection circuit gave a clear indication of purposely hydrided areas believed to contain 250 ppm hydrogen in two N-Reactor process tube samples. Destructive metallographic analysis was used to determine the hydrogen content. Vacuum diffusion analysis to check this result is planned. Comparison of the circuit output during tests on the samples before and after hydriding would ordinarily be required for detection of hydride concentrations this low. However, it was not required in this case, since the background resistivity of the samples was uniform. Preliminary information from tests performed on the samples by Materials Engineering, HL, indicates that 250 ppm hydride may have a significant effect on the mechanical properties of Zircaloy-2.

Eddy current tests on two butt welded N-Reactor tube samples confirmed the earlier result that no unusual variations in resistivity occur in the weld zone. Thus, there is no strong dependence of the resistivity on grain size since the grains are larger in the weld zone.

The new prototype transistorized eddy current hydride detection circuit has been completed and used to detect hydride in a bench standard Zircaloy-2

tube. Tests showed that several modifications in this circuit would improve the performance, and these are now being made.

Tests were made to determine whether or not flaws in process tube walls could be detected with the hydride detector. A 1/16 inch diameter hole drilled to within .020 inch of the inner wall of a Zircaloy-2 N-Reactor tube sample gave a readily detected output signal.

Fundamental Ultrasonic Studies

Experiments with critical angle boundary wave effects continued. Feasibility studies were conducted to determine if boundary waves are sensitive to surface stresses and to the condition of bonds between acoustically dissimilar solids.

Compressive stresses, which produced a range of surface strains, were applied to samples of brass, aluminum, and stainless steel. Ultrasonic signal changes were observed for strains of about 300 micro-inches per inch in brass and 300 to 500 micro-inches per inch in aluminum. Signal amplitude plots across the reflected beams showed that the observed changes were mainly due to a lateral shift in the reflected beam. In all cases this lateral shift increased with increased strain. A maximum strain of about 500 micro-inches per inch in the stainless steel did not produce a measurable change in signal. The experiments with stainless, however, demonstrated two items of interest. The reflected beam amplitude plots for zero strain and maximum strain were extremely close in contour. This indicated that a high degree of experimental reproducibility was achieved since, between zero and maximum strains, the sample was removed and replaced five times in obtaining intermediate amounts of strain. Secondly, the amplitude plot for stainless was entirely different than has been observed for other materials. The reason for this is not evident at this time.

In order to more definitely determine the depth of boundary wave propagation in the preceding experiments, an aluminum sample was drilled with flat bottom holes to within varying distances from the surface. Hole depths beyond about 0.035 inch were found not to appreciably influence the propagation. Since a frequency of 5 Mc was used, this depth is about half way between the wavelength of shear and longitudinal waves. Other materials, having different wavelengths, would effect the propagation to different depths.

Bond layer experiments are not complete. Preliminary work, however, has shown that critical angle tests are sensitive to bonding defects such as unbonds. To demonstrate that boundary waves can be generated at a bonded interface, a Lucite sample was bonded with adhesive to the aluminum sample.

having the drilled holes. Experiments with these holes demonstrated that boundary waves can be generated at the interface. Penetration of these waves in the aluminum was found to be about the same depth as that observed in the experiments reported above.

Efforts to further examine the nature of boundary waves also continued. In addition to previous work in which narrow frequency band equipment had been used, some preliminary work was done with broadband equipment. Boundary waves were generated with this equipment, along several water-metal interfaces. As the waves propagate along the surfaces, the amplitude decreased uniformly and did not appear to exhibit the energy exchange behavior and this suggests that the periodic narrow band signal changes that were observed previously were actually due to interference effects. During these measurements, another effect was noticed. Along with the amplitude decrease there was also a uniform decrease in frequency. At a distance along the interface of about one inch from the main reflected beam, the boundary wave signal decreased in frequency by about a factor of two. This reduction of frequency was previously noticed with narrow band equipment but not to a measurable amount.

Analyses on Lamb waves in a hollow cylinder continued. The method used for transforming the frequency equation into the form of the flat-plate equation plus corrections requires the expression of products of Bessel functions in a series of trigonometric functions in a manner previously unknown (at least not published, as far as can be determined). The method appears analytically sound. The laborious work of obtaining an explicit expression for the flat-plate correction term continued. Efforts to find short cuts have, so far, failed.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

The prototype sheath tubing tester was set up to test 35 mil wall, 0.75 inch diameter Zircaloy tubing. By utilizing the 3/16 inch diameter, spherically-focused crystal in conjunction with the prototype electronics, and propagating shear energy at 45° within the tube wall, linear notch depth amplitude relationships were obtained. Notches down to five percent of the wall by 20 mils in length were detected with equal amplitude on the inside and outside tube surfaces. Some problems were encountered from small particles in the water; however, by recirculating the ultrasonic couplant water through a filtering system and covering the test tank, these problems were corrected. Up-to-date drawings incorporating all the design changes prompted by operating experience are under preparation. Preliminary copies have been forwarded to the Machine Shop so that fabrication

of the final tubing test mechanical system can begin. With the exception of amplifiers, all the components for the electronic system have arrived. Fabrication of the tubing test electronics has begun.

In applying the tubing test to thicker tubing deeper notches were required. The punch used on the thin wall tubing did not produce a good notch at depths greater than two mils. By redesigning the punch to a wedge shape, uniform notches up to five mils in depth were readily punched. The displaced metal which erupts around the edges of the notch is polished off leaving an accurate defect surface which is measured optically for depth.

Development of ultrasonic transducer measurement techniques continued. All the types of ultrasonic transducers presently employed by Hanford have been measured, completing the development work required. Quartz, lithium sulphate, and ceramic crystals ranging from 2.25 to 20 Mc with flat, cylindrical-focused and spherical-focused lenses were evaluated. Variations in transducer strength and frequency are recorded. These variations can be attributed to bonding or backing differences along with variations in the thickness or ultrasonic velocity of the crystal. Utilization of this transducer information should assist in standardization of ultrasonic tests, assist in determining transducer specifications, and possibly predict transducer life. A report on the ultrasonic transducer measurement techniques with results attained is under preparation.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples during the month in accordance with current schedules.

Studies were carried out on the performance characteristics of the scintillation-type ion detector installed on the mass spectrometer last month. The dead time of the tunnel-diode discriminator circuit was reduced and measured to be 0.23 microseconds. The stability of the ion-beam magnetic-switching system was improved so as to define the ion-beam position at the analyzer exit slit to within 0.001 inch. The new detector gives a constant response over a variation of 0.005 inch in the ion-beam position. Measurements of the U^{235}/U^{238} ratio of three different NBS standards have agreed within the measurement precision (0.3 percent) with the certified value. Repeated measurements, however, have shown a large variability in result with sample-emission life in the spectrometer. This apparent effect will require further investigation. A faulty gasket seal in the spectrometer was determined to be the cause of abnormally high pressures in the spectrometer tube.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

In diffusion research, significant progress was made in expressing the lateral growth of a plume solely in terms of meteorological parameters which can be readily calculated from wind velocity data on strip charts. Experimental values for the cross-wind variance of the exposure distribution have been shown to be related by

$$\sigma_y^2 = At - A\alpha + A\alpha e^{-t/\alpha},$$

where σ_y^2 is the cross-wind variance, t is the plume travel time, and the parameters A and α can be determined independently from wind measurements at the source. These relations are

$$A = 13.0 + 232.5\ \sigma_\theta \bar{U},$$

and

$$\alpha = (13.0 + 232.5\ \sigma_\theta \bar{U}/2(\sigma_\theta \bar{U})^2,$$

where σ_θ is the standard deviation of the wind direction at the source over the period of emission and $\bar{U}$ is the average speed for the period. This functional form for the variance was first proposed in 1921 from theoretical considerations, but the Hanford results are the first to relate A and α to wind velocity data, making application possible to distances 16 miles from the source.

The replacement instrumentation for the portable mast was received, and underwent testing this month. The sensors (6 anemometers, 6 wind vanes, and 6 thermocouples) were mounted on a boom near the 622 Building. Here, under identical meteorological conditions for all sensors, comparisons of the recording signals are being made. To date, the tests show that the system is operating within the design specifications.

Two successful field tests were conducted with zinc sulfide releases made from 200 feet on the meteorology tower. Exposure measurements were obtained to 2 miles in one test and to 1 mile in the other. The real-time sampler was successfully operated during the tests.

Further analysis of data obtained with the real-time sampler have been made. Last month data were presented which were obtained over a 19-minute period at 600 meters downwind from the source. Similar data have been reduced for the same field test at the 1600-meter arc for a 29-minute period. The table summarizes the percent of time the ratio of the instantaneous concentration

and the average concentration exceed a given value, A. Some minor corrections have been applied to the 600-meter data which were reported last months.

Percent of Time Instantaneous Concentration/Average Concentration Equals or Exceeds A

A	Percent of Time 600 m	1600 m
0	100.0	100.0
1	15.8	22
2	10.9	14.2
3	8.2	10.4
5	5.2	6.3
10	2.1	2.1
20	.14	.32
30	.039	.053
34.6	Peak Observed	.041
40		.022
50		.006
58.5		Peak Observed

Radiological Physics

Work on the plutonium scintillation counter was almost at a standstill because of failure of the multichannel analyzer used with it. Some measurements of its sensitivity for plutonium in the human (phantom) lung were made. They were in the range of sensitivity expected. A proportional counter being built for this same purpose was tested with an argon-methane counting gas and found to operate well and have very good resolution. Xenon gas will be used when the system has been made tight.

The coincidence beta ray counter described last month was tried on a patient receiving P^{32} therapy. The signal to background ratio was much better than obtained with the bremsstrahlung counter. Furthermore, the background seems to be more nearly the same from one person to the next (on the basis of a small number of tests). Only a small lead shield around the head of the subject is required for this counter. These facts indicate that the counter may be suitable for P^{32} detection in the truck whole body counter (the bremsstrahlung counter is not suitable because of the large amount of shielding required).

An attempt was made to correlate the radioactivity measured in the air at the mill of a uranium mine with that found by whole body counting in a

person who had been in that atmosphere. The attempt was unsuccessful because too long a time elapsed between departure from the mill and arrival at Hanford Laboratories for counting.

The positive ion accelerator operated satisfactorily during the month.

A proportional counter for low energy fast neutron spectrometry was assembled and leak-checked. Tests with a methane filling were satisfactory.

The neutron generator for the University of Washington spermatogenesis project was received, assembled, and put into operation.

Difficulty was encountered in calibrating the calorimeter being used to measure the half-life of Pm^{147}. The thermistor used for measuring the calorimeter temperature was found to be changing in sensitivity by about 0.1% between calibration runs. Fortunately some other thermistors, in the calorimeter but not used before, were found to be stable to about 0.01%. The measurements are proceeding using these thermistors. The first model of a smaller calorimeter was put together and is being tested.

Instrumentation

Operational tests were successful on the entomological species counter developed for use at Radioecology, Biology Laboratory, and the instrument is now in routine operation. A report was written and drawings are being prepared.

A method was devised for control of the respiratory valve used at the Biology Laboratory in the measurement of radionuclide deposition in animal lungs. The pneumotachographic transducer used proved to be hypersensitive to animal body movements, and some spurious switching was noted. A method of circumventing this difficulty is being investigated in which the valve is synchronized with the animal's breathing by a method that does not require contact with the animal. In addition, work is continuing in an effort to develop a suitable transducer to measure breathing volume both accurately and indirectly.

General development work progressed regarding a special alpha air monitor, which will use alpha energy analysis techniques, for use in the Biology Inhalation Laboratory. Portions of the system are commercially available items, and purchase specifications for these have been completed.

Progress was achieved on the experimental animal biological function telemetry system for measurement of temperature, respiration, and pulse rate. For temperature measurement, a miniature oscillator was developed which uses

a thermistor as a frequency control element. As the temperature changes, the oscillator frequency varies proportionally and this modulates the carrier, which was about 100 Mc/sec in frequency for this particular test. It appears that the complete transmitter package, for the temperature measurement portion, will be about 0.5 cubic inches in volume. Satisfactory transmitting-receiving operation to distances of 20 feet was achieved.

Initial satisfactory testing was achieved in the field with major portions of the portable mast data logging and recording system. Following extensive laboratory tests and final calibration of the temperature measurement portion, the system was field tested. The developed portions of the system performed well; however, some difficulties were experienced with the commercial digital voltmeter, and modifications may be in order. Calibration is now proceeding with the wind speed and direction segments of the system.

Several of the necessary work sheet drawings were completed for the revised data station of the HAPO Radiotelemetry System, and a new solid state radio transceiver was ordered to replace the old vacuum tube unit. Fabrication work was completed on the power supply portion of a second data station, and fabrication was partly completed on the three-channel radiation monitor, to be used in each data station.

Circuit development was started on the Atmospheric Physics wind component meter in an effort to materially improve operational performance. It appears that extensive filtering may be required.

Work continued on the 400-channel analyzer during the month. Power supply noise and ground loops were found to be causing bits to be lost in the read/write memory cycle. Filtering the power supply and improving the grounding system corrected these problems. Digital to analog decode networks for the oscilloscope display were developed and installed in the analyzer.

TEST REACTOR OPERATIONS

The PCTR operated routinely during the month. There were two unscheduled shutdowns due to electrical failures. A set of foils was irradiated for standardization purposes for the Graphite Research and Development Operation. The experimental determination of the fluxes to be expected in N Reactor in special columns for isotope production was nearly completed during the month.

The TTR was operated on a two nights a week basis for the University of Washington Graduate Center. There were no unscheduled shutdowns. Several sets of foils were irradiated for normalization and standardization.

The fuel for the split table assembly has been received from KAPL and stored. The mechanical components also have been received.

Weather Forecasting and Meteorological Services

Meteorological and climatological consultation services included 1) participation in the Radiological Emergency Control Center practice sessions, 2) response to a request for climatic data from the State of Washington, 3) summarization of climatic data pertinent to reactor operations, 4) a study of wind variability for N-Reactor licensing hearings, 5) 233-S renovation, and 6) determination of the 10-year expectancy of frost penetration at Hanford.

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	90	83.6
24-Hour General	60	82.0
Special	195	86.7

November was the fourth straight month of much-above-normal temperature and closed out the warmest fall season (average temperature 56.6) ever recorded in the Hanford Area. The previous fall season high since 1912 was 55.5 in both 1937 and 1953.

Rainfall totaling 0.19 inch on November 5 brought an abrupt end to a long dry spell, which had seen only 0.07 inch in the previous 117 days. Subsequent rains brought the November total to 0.74 inch, which was practically normal.

Mass Spectrometry

Isotopic analyses were provided on eleven samples of uranium in support of HAPO U^{233}-production studies and three analyses were provided in support of research in Nucleonics Instrumentation.

Instrumentation and System Studies

Three scintillation and solid state coincident count type alpha air filter counters were completed, tested, and delivered. The instruments, which

performed correctly in demonstration tests, were for Control Operation, CPD, and for Fission Products Processing Operation, CPD.

All necessary laboratory testing was completed on the soil moisture measuring probe developed for use by Advance Technical Planning, CPD, and Chemical Effluents Technology, HL. Remaining work is to complete the water-tight aluminum housing and finish the assembly.

General design work was started on a special scintillation mixed fission product detection instrument for use in measuring face mask contamination. The instrument is being designed for Protective Equipment Decontamination, HUPO.

Assistance was rendered to Calibrations Operation, HL, regarding modifications of a number of Hanford developed, offsite fabricated solid state annunciator units to be used with HAPO portable radiation survey instruments. Incorrect parts had been used by the manufacturer. Proper parts installed provided fully satisfactory performance.

Bid reviews were completed and orders placed for the commercial instrumentation portions of the U-235 fuel enrichment monitor to be used by Metal Fabrication Development, HL. Drawings were started on the items to be fabricated locally.

A device was fabricated to enable pre-heat and post-heat cycles to be used on a resistance welder located in the 325 Building. This extra heating will be used in an attempt to eliminate the cracking encountered when welding beryllium fuel elements. The device alters the phase of a 60-cycle input signal used to fire the welding current control tubes. The piece to be welded is then heated for a short time before welding with currents of less than 10,000 amperes, then welded at higher currents, then heated for a short period after the weld. The entire system has been tested but no welding has yet been done.

Design and construction on Chemical Effluents Technology's ground water analog system has continued during the month. The system will be used to simulate the ground water conditions in the Hanford area.

The 662 node test model assembly has been moved to 200-W Area where the wiring is being done. Ninety percent of the resistors have been installed and all of the rear panel wiring is complete. It is expected that the remaining resistor installation and connector and input wiring will be complete within a week.

After completion of the assembly, tests will be made to check the accuracy

of the wiring. Data will then be obtained from the system in the form of node potentials. These values will then be compared with the predicted results obtained from a digital computer program. The analog system reliability and feasibility will then be determined.

A prototype temperature comparator designed for use as an out-of-limits detector on the uranium swelling program was evaluated for accuracy, repeatability, and temperature compensation. The prototype unit performed within requirements. A specification for a complete system consisting of four channels of high-low comparators with adjustable setpoints was written subsequent to the evaluation.

Advice and design assistance was rendered Reactor Metals Research on the problems of a) using a servo-repeater to follow a ± 10 volt error signal by positioning a feedback potentiometer and an output potentiometer which functioned as the slidewire input to a three-mode controller, and b) providing a method of measuring dynamically the resistance of a metal specimen as a crack propagates its length by measuring the voltages associated with a current passed through the specimen.

A General Electric GEMAC control system with the unique anti-reset windup circuit in the controller was purchased on a rental basis for evaluation and testing with option to buy later. A letter justifying procurement by those means was written to AEC Purchasing and was approved.

The creep program data logger and controller were run on automatic for a period during the month with success. Shakedown will continue through December to allow more operating time and more complete debugging prior to on-line installation.

A calorimeter control circuit was used as the feedback element in the control of sample temperatures. The circuit was strictly a proportional control circuit and could not control the temperature within the desired limits of ± 0.1°C. The high gain that was required in order to obtain one ampere of output current at the high temperatures caused the system to oscillate. After several design changes, it was found that proportional control alone would not be sufficient. A new circuit was designed which contained proportional-plus-reset-plus-rate-compensation at the input. The circuit has just been built and, as yet, has not been tested.

Simulation of the simplified C-column model was completed. The automatic random minimizing circuit was tried again and found to give acceptable results. However, the time required for it to converge on the optimum values of B and K was considered excessive.

The simulation of the complex model was reprogrammed to improve patchboard layout and many of the external relays were replaced by internal relays. The new patchboard was designed to accommodate the automatic random minimizing circuit. Test results have shown that the solution is extremely sensitive to the initializing calculations, indicating that the simulation provides only rough values of the optimum values of B, K, and D.

A rough draft for the formal report on the C-column was written. The report cannot be completed until all of the 17 remaining computer runs are finished.

Optics

A study has been made for the purpose of devising a camera which will restore the true relative dimensions to a picture of a metal surface taken at a large angle of incidence in an electron microscope. Two methods of rectifying the picture are possible. One employs two crossed cylinder lenses adjusted so that the magnification of one lens is the reciprocal of that of the other lens; the other uses conventional lenses but stretches the image in one dimension by translating the film during exposure.

A new traverse mechanism has been designed for use in the old reactors. Improvements are expected to increase the accuracy by a factor of two. This unit will supplement the unit now in service.

During the five week period (October 13 to November 17) included in this report, 520 manhours shop work was performed. This work included:

1. Fabrication of 10 glass bearings for waste solution pumps at CPD.
2. Fabrication of a rotating film holder for a B&L metallograph.
3. Repair of crane periscope heads for Purex.
4. Fabrication of parts and repair of one tank periscope for CPD.
5. Repair of camera adapter for an IPD borescope.
6. Fabrication of two quartz prisms.
7. Fabrication of a gap measuring probe for Physical Testing, HL.
8. Modification of the N-Reactor seismoscope.
9. Fabrication of parts for the new traverse mechanism.

Physical Testing

An ultrasonic test was developed to evaluate the quality of spot welds on the locking and support clips of N fuel elements. The test has been demonstrated to measure uranium penetration in the weld, lack of penetration,

voids and unbonds. Metallographic evaluation of test specimens verified the ultrasonic test results.

A complete nondestructive inspection was completed in record time to analyze the condition of a 40-ton crane hook assembly that was dropped in the Purex Canyon area. Radiography was used to confirm the condition of the disassembled and decontaminated critical parts. All major parts were not damaged; however, some bolts and retaining pins required replacement.

An emergency call was answered to find propane leaks in the 108-F Building with the ultrasonic translator. Propane content in the atmosphere was sufficiently high that individuals had been sent to First Aid. All major leaks were identified and repaired. Potential leaks were also scanned in one of the N-Reactor steam generators during the repair program. No leaks were detected with either the ultrasonic translator nor with the helium mass spectrometer.

Work continued on the Eddy Current Motion Analyzer for use in studying the fuel assembly vibrations which have been encountered in the PRTR. A prototypical fuel rod, containing an eddy current sensor, was successfully fabricated and tested; work is proceeding on the fabrication of the complete nineteen-rod, sensor equipped fuel assembly.

The eddy current test which was developed to obtain an initial condition signature for the N-Reactor process tubes was employed on only a few tubes. Testing was discontinued when significant amounts of sediment were found in most of the tubes checked. The tubes inspected included most of the two top rows and several randomly selected tubes in the center row. Difficulty was experienced in traversing the coil along the length of some process tubes, presumably due to insufficient clearance between probe coil and tube wall. The 55 tubes inspected provided traces that show each tube will provide its own unique signature for future comparison.

The eddy current test employed on the N-Reactor steam generators was used again to check the results of mock-up studies at the Combustion Engineering Company in Windsor, Connecticut. Sections of corroded and non-corroded steam generator tubing were operated for a total of 700 hours at simulated temperature and pressures. The eddy current test revealed no significant changes had occurred in any of the samples.

Eddy current and ultrasonic test development are continuing to reduce some of the problems encountered in testing Incoloy 800 tubing. The tubing, one half inch in diameter with 0.037 inch wall represents material from two vendors and with two metallurgical conditions. Excessive background noise associated with this particular tubing has limited test sensitivity with

normal inspection practices to approximately twenty-five to fifty percent defects; however, laboratory developments have reduced this figure to about ten percent. Work is continuing on this tubing as well as on the Inconel 600 tubing to be used in the retubing of the 4-A N-Reactor steam generator.

Work is progressing on the charge-discharge console for capacitor discharge equipment whose function is to provide large amounts of electrical energy pulses for heating metal samples. The ultimate objective of this program is to measure property changes which may occur in the sample as a result of extreme temperature transients. Orders have been placed for the power supply, charging resistors, vacuum relays, and the special low-inductance storage capacitors.

High temperature bonding mediums are being investigated to inspect metals at high temperature with ultrasonic longitudinal and shear waves. One system meeting with success so far has been the bonding of an aluminum specimen to a titanium rod and successfully detecting the longitudinal wave at 650 C, the melting point of the aluminum. The shear wave was lost in the titanium rod; the geometry of the titanium rod is being changed to eliminate the problem.

ANALOG COMPUTER FACILITY OPERATION

Computer utilization was as follows:

265	Hours Up
23	Hours Scheduled Downtime
8	Hours Unscheduled Downtime
0	Hours Idle
296	Total In-Service Hours

Approximately 112 hours of off-shift (swing and graveyard) operation are included in the totals shown above.

Problems considered during the month were:

1. HTR (Two-Dimensional Model).
2. Critical Mass Power Spectrum Analysis.
3. C-Column.
4. NPR Steam Generator.
5. NPR Injection System.

The latest vendor information indicates that the new computer should arrive on-site by December 2. The delay of nearly two weeks from their target date of November 15 was caused by some difficulty in completing the Digital Output/Input Translator (DO/IT) and a certain amount of component rework which was necessary.

The step-by-step procedures for the acceptance tests are approximately 95% complete.

INSTRUMENT EVALUATION

Four of the seven offsite fabricated combined alpha-beta-gamma scintillation hand and shoe counters have now passed all acceptance tests and are ready for plant service. The acceptance testing work is being carried out by maintenance technicians who will eventually be servicing the instruments; thus, the work entails both testing and training.

Two months of satisfactory field testing at 105-B Building were achieved on the experimental gamma background compensated beta-gamma hand and shoe counter, and the instrument was moved to the 321 Building for further tests and demonstrations. Measured detection efficiencies were considerably better than for the regular Hanford hand and shoe counters.

All testing, except for temperature dependence effects, has been completed on a commercial portable beta-gamma dose-rate meter. General performance has appeared to be satisfactory in most tests; however, the unit cost of about $600 may preclude Hanford use.

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CHEMICAL LABORATORY

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Reactor Film Adsorption Studies

Two coupon holders which had been in the flux zone of the C-Reactor for two and one-half months were removed for analysis. One coupon holder will be analyzed for corrosion effects; the other coupon holder was studied for the effects various coatings had on the concentrations of radionuclides in the film formed on the coupons from the reactor cooling water. The coupons tested had been left uncoated, or had been coated with Teflon, a silicone, or with one of two different black ink formulations. After removal from the reactor, all of the coupons were found to be identical in appearance, each having the characteristic red reactor cooling surface film. There was no visual evidence to indicate that any film had remained intact for any significant time. The surface films were removed from the coupons with 3 N HNO_3 and analyzed for Cu-64, As-76, Np-239, Ga-72, P-32, Cr-51, Fe-59, Zn-65, Sc-46 and rare earths. Comparison with results from uncoated coupons exposed at the same time showed no significant differences in the concentrations of any of these radionuclides. In reactor tests where these coatings were exposed without a high neutron flux, the coatings remained largely intact and strongly inhibited the processes by which radionuclides were adsorbed on or produced in the film. These studies indicate rather definitely that organic surface coatings are not able to survive the high-flux, high-surface-temperature environment in the reactors, and cannot therefore greatly inhibit the radionuclide production process.

Electrodeposition of Nickel on Uranium

Laboratory studies of procedures for nickel plating uranium metal have continued with the development of a new coating porosity test and with attempts to produce thin nickel coats with low porosity. In the porosity test, the plated uranium specimen is anodized briefly in an electrolyte containing potassium ferrocyanide. The anodization, at constant current and for a known length of time, allows the formation of a uranyl ferrocyanide deposit at the surface of the specimen, in amount directly related to the porosity of the coat. The amount of iron in the deposit is then measured by X-ray fluorescence.

To standardize the test, coats of varying thickness were plated from a Watts bath onto a series of uranium coupons. Plotting coat thickness versus the log of the relative pore area, measured as above, gave a straight line for coat thicknesses of 8 to 17 microns.

This porosity test was used to evaluate the effect upon coat porosity of (a) superimposing an AC ripple upon the DC current used for plating, in a Watts bath; (b) the use of a high initial current density (500 ma/cm^2 for 10 seconds), in a bromide bath; and (c) the use of a sulfamate plating solution, which is claimed to yield plates with low tensile strength and hence low porosity. The applied AC voltage did not show any significant improvement in plate porosity. The high initial current density in a bromide bath reduced the relative pore area by a factor of five, presumably by causing the nickel to nucleate evenly on the uranium surface, with consequent reduction in grain size. The sulfamate bath decreased the relative pore area by a factor of two. At this point, the most promising avenues to production of a thin ($\leq$ 10 micron) nickel plate on uranium metal appear to be in the use of a bromide or a sulfamate bath.

Corrosion of Aluminum Reactor Nozzles during Decontamination

A study was made to determine possible cause for severe corrosion of cast aluminum (356) reactor nozzles which occurred recently during decontamination of the nozzles. The decontaminating bath used was Turco 4306-C, an inhibited sulfamic acid cleaner. The bath had been used previously on similar nozzles without corrosion problems. Corrosion tests and titration studies indicated that the temperature of the bath, during the decontamination in which severe corrosion occurred, had been higher than that desirable for this cleaner. At temperatures above 80 C, Turco 4306-C and its principal hydrolysis product, ammonium hydrogen sulfate, are severely corrosive to both aluminum and mild steel.

Emissivity of Aluminum-Clad Fuel Elements

Tests were run to determine the emissivities of aluminum-clad uranium fuel elements. These data are desired to improve calculations of the hazards of overheating of irradiated fuel elements which do not fall into the water-filled basin upon discharge. Six elements have been used, to date, in two types of tests.

In the first series the fuel elements were induction heated to 1000 C and allowed to cool in air. The temperatures were taken with a thermocouple. The cooling curves were then used to determine surface heat-transfer coefficients, which were a function of both radiation

and convection. The convective coefficients were calculated, and the radiant heat loss was found by difference.

The second series was similar; the fuel elements were heated as before, but this time they were allowed to cool in an evacuated (5 in. Hg. abs.) quartz tube. All the heat loss was assumed to be by radiation. The average values from the two methods are shown below:

Temperature (° C)	Uranium Phase	Emissivity	
		Air Cooled Series 1	Vacuum Cooled Series 2
1000 to 770	Gamma	0.35	0.46
770 to 655	Beta	0.38	0.57
655 to 600	Alpha	0.39	0.63

The higher emissivities noted in Series 2 may be due to significant heat losses to the residual air in the quartz tube. Additional tests at better vacuums will reduce these losses and improve the accuracy of the emissivity measurements.

SEPARATIONS PROCESSES

Uranium-233 Production Studies

A one-slug dissolver, for use in B-Cell of the High Level Radiochemical Facility, was completed and tested in the laboratory by dissolution of a "cold" (unirradiated) 6-inch thorium slug. The aluminum cladding was removed by reaction with $NaNO_3$-NaOH solution and the bare thorium then dissolved by use of fluoride-catalyzed nitric acid. Since total dissolution was desired, and the rate of dissolution decreases markedly as the area of the piece and the concentration of residual nitric acid decrease, three incremental cuts with nitric acid were employed. These contained about 44, 8 and 4 moles of nitric acid, respectively, and gave a final solution containing about 2 M thorium, 5 M HNO_3, and not-yet determined concentrations of iron, chromium and nickel.

Total dissolution required 68 hours. For plant application, dissolving time could be considerably improved and terminal acidity decreased by use of a heel of unreacted metal. After minor modifications, the dissolver was installed in B-Cell for dissolution of the first of the irradiated thorium metal slugs.

Other laboratory work was directed at defining a flowsheet for in-cell recovery of U-233 from the three dissolved slugs. It is planned

to use MnO_2 scavenging for Pa-233 removal (and subsequent decay to U-233) and batch solvent extraction for U-233. The partially decontaminated U-233 will be removed from the cell for final purification.

Direct Conversion of Thorium Nitrate by Spray Calcination

The thorium product from a chemical separations plant is expected to be a thorium nitrate solution, which will require conversion to fuel grade thoria (ThO_2) for recycle to the reactors. Several scouting runs were accordingly made in the 8-in. x 10-ft. "cold" spray calciner to evaluate possible application of this versatile calcination technique to this service. Initial results were quite promising. The feed was a 1.2 M thorium nitrate solution which happened to be available from other work (a more concentrated feed would result in improved calciner capacity). The product of the calcination, in which the column was operated at a nominal 750 C, was an excellent-appearing, free-flowing powder. Estimated particle size was ca. 50 microns, and iron pickup (43 ppm) was less than with pot calcined material. Moisture content was only 0.067 percent, and was reduced to an undetectable level by heating to only 150-180 C. Tap density was ca. 23 percent of theoretical. Cold pressing at 40 tsi increased the bulk density to about 60 percent of theoretical, adequately high for low-density target elements. Some of the powder was pneumatically impacted (by CFDO personnel) to 96 percent of theoretical density and combined with pot calcined material for fabrication of the thoria test elements currently being charged to the reactors.

Apparent advantages of spray calcination of thorium nitrate include (1) process and equipment simplicity, (2) high purity of the product, and (3) small particle size, which may materially improve dissolution kinetics-as compared to Sol Gel or pneumatically impacted material.

Plutonium-Neptunium Separation by Batch Amine Extraction

Preliminary experiments show that plutonium can be nearly quantitatively separated from neptunium by a simple, batch, amine extraction from a dilute HNO_3 solution in which plutonium(IV) and neptunium(V) are stabilized with 0.1 M $NaNO_2$. Using a feed containing 35 g/l Np, 1.75 g/l Pu, and 1 - 1.72 M HNO_3, two room temperature extractions (one-half volume each) with dodecane containing 15 v/o trilauryl amine and 7 v/o n-octyl alcohol extracted 99.3 percent of the plutonium and only 0.5 percent of the neptunium. Following the extraction, the two extracts were combined, scrubbed with one-tenth volume of 1 M HNO_3, and stripped with two quarter-volumes of 1 M HNO_3 containing 0.2 M ferrous sulfamate. Overall recovery of the plutonium was 99.1

percent. The neptunium decontamination factor was 350. It is believed that the neptunium decontamination factor can be increased at least three-fold by better phase separation and by improved stripping conditions.

Americium Extraction

The distribution of nitric acid between DBBP solutions and sodium nitrate-nitric acid solutions has been measured, and americium extraction can now be correlated with the acid content of the system. It is found that the distribution of Am-241 at constant nitrate concentration is determined almost entirely by the free DBBP concentration.

The Harned salt effect is small, but among the ions of process interest the salting effect is in the order of $Al^{3+} > Mg^{2+} > Ca^{2+} \gtrsim Na^{+} > Fe^{3+}$.

Polonium-210 Separation

Preliminary experiments have been carried out to test the separation of Po-210 from bismuth by carrying the polonium on cerium bismuthide precipitated from molten bismuth metal.

A Vycor bulb with a long stem was charged with 46 grams of bismuth (traced with Po-210) and 0.6 gram of cerium metal, evacuated and sealed. The bulb and contents were then heated to 500 C and held at this temperature, and agitated, for 28 hours. Following this treatment, during which the cerium metal dissolved, the bulb was so positioned as to allow the metal to flow into the stem. The system was then cooled to 400 C and held at this temperature for four hours, to permit insoluble cerium bismuthide to float to the top of the narrow column of bismuth metal. Po-210 assays of segments of the bismuth metal column have been made and show that 81 percent of the Po-210 concentrated in about five weight-percent of the bismuth in the top centimeter of the 15 cm column. The remaining Po-210 was evenly distributed along the bismuth column. A blank run with no cerium addition is currently in progress. Preliminary indications are that no segregation of Po-210 occurs in the absence of cerium.

An obvious advantage of a procedure of this type is that it would permit immediate recycle of most of the irradiated bismuth to the reactor for production of more Po-210. Moreover, only a small percentage of the irradiated bismuth need be processed further, or shipped off-site for further processing, to recover pure Po-210.

Szilard-Chalmers Reactions for Thallium-204 Production

A series of reactor-irradiation experiments have been carried out to scout the potential of "hot atom" reactions for the production of Tl-204 of high specific activity. Two schemes were tried: (1) irradiation of thallium(I) loaded zeolites (on which the thallium had been fixed by heating) followed by post-irradiation leaching with dilute acids or salt solutions; and (2) irradiation of insoluble, thermodynamically-unstable thallium(III) oxide followed by water dissolution of any soluble thallium(I) produced. Negative results were largely obtained. No enrichment was seen with the zeolites and less than a factor of 2 with the oxide.

Trilauryl Amine Extraction of Plutonium and Neptunium from Purex 1WW

Highly encouraging results have been obtained in using hydroxylamine sulfate solutions to co-strip plutonium and neptunium from TLA-Soltrol solutions. At 50-60 C, contact of 0.3 M TLA-Soltrol solution with an equal volume of 0.05 M $(NH_2OH)_2H_2SO_4$ solution for sixty minutes removes over 99 percent of the plutonium and 95 percent of the neptunium present. The hydroxylamine sulfate strip solutions can be routed directly to the Purex J-cell package (3A-3C) cycle for separation of the neptunium from plutonium and accompanying fission products. With oxalate strip solutions the oxalate must first be destroyed because oxalate interferes with neptunium extraction in the 3A column.

Isotopic Problems in Thorium Recycling

Calculations have continued to define any radiation problems that could cause trouble in either the recycle thorium or the U-233 product. The major result of the calculations completed this month has been to show that, at 1 ppm U-232 in U-233, the Th-228 concentration in recycled thorium will decrease each cycle instead of increase as was reported last month. The Th-228 concentration will also decrease each cycle at higher U-232 concentrations but the U-232 concentration that results in a Th-228 increase has not been defined. The decrease in Th-228 concentration results from the removal of Ra-228 during initial thorium processing. Ra-228, a precursor of Th-228 in the natural thorium decay chain, has a relatively long, 6.7-year, half life. After processing from the ore, the Th-228 content in natural thorium decreases for several years until the Ra-228 builds up to a point that Th-228 is produced at a greater rate than the residual Th-228 is decaying away. For short irradiation times and a U-232 concentration of 1 ppm in the U-233 product, the amount of Th-228 produced from the decay of U-232 will

be much less than the amount of residual Th-228 (from the natural decay chain) that initially decays away.

Other findings, based on U-233 containing 1 ppm U-232, are as follows:

1. The U-230 concentration in U-233 will be insignificant at cooling times of several hundred days.

2. The Th-227 and Th-229 concentrations in recycle thorium will be insignificant.

Calcination of Thorium Nitrate

Studies were initiated on the direct calcination of thorium nitrate. Approximately 22 pounds of ThO_2 were prepared for subsequent target element fabrication by agitated-bed, direct denitration of thorium nitrate to a terminal temperature of approximately 700 C. Calcinations proceeded smoothly from thorium nitrate tetrahydrate crystals, through liquid, mastic, and finally powder stages. The resultant powder, on loading (by Ceramics Research and Development) into cans prior to pneumatic impaction, had a density of 37 percent of theoretical. On subsequent work by the Mallinckrodt Chemical Company, direct agitated pot calcination under steam blanket resulted in a thoria product, after firing to 1100 C, with a nominal tap density of 75 percent of theoretical. Adaptation of this process to existing Hanford equipment is being explored.

Flowsheet for Separation of Plutonium from Uranium and Thorium

In a study flowsheet concept for recovering uranium, plutonium and thorium from scrap materials, uranium is extracted with a TBP-CCl_4 solvent; thorium and plutonium(III) are left in the aqueous raffinate. The raffinate, after addition of aluminum nitrate is fed to a static column containing 30 percent DBBP (dibutylbutyl phosphonate)-CCl_4 solvent to extract thorium and leave plutonium in the raffinate. Batch contact studies show that extraction of plutonium into the DBBP-CCl_4 solvent is significant and that adequate separation of plutonium and thorium during the extraction step only is not feasible. Selective stripping techniques to improve the plutonium-thorium separation are under study.

Disposal to Ground

No significant changes were noted in the areal extent or the concentrations of radionuclides in the contaminated ground water beneath the 200 Areas during the current month.

A correlation of beta-emitter concentrations in samples from well 699-34-39A, located two miles southeast of the Purex plant and in the path of the contaminated ground-water plume, with the discharge of activity to ground from the Purex plant shows a travel time for radioruthenium of about three years to this point. Peak discharges from Purex in 1957 and 1960 were noted in samples taken from well 699-34-39A in 1960 and 1963, respectively. On this basis, increased activity in the Purex process condensate waste noted in late 1962 should be evidenced in that well in late 1965 or early 1966.

Well sampling results for October and November, 1963, showed the persistence of low but detectable concentrations of gross beta-emitters in several wells within 0.5 mile of the Columbia River and in intervening wells back to 200-East Area. Detectable and nearly equal concentrations of contaminants (1.5×10^{-7} μc β/cc) were noted in all piezometers installed in wells 699-20-E12 and 699-10-E12 located close to the river. A nominal travel time of seven years for radioruthenium to move from the Purex plant to these wells is thus established.

I-131 in Plant Streams

Charcoal cartridges continue to be tested for iodine retention with Redox sand filter inlet air. Additional measurements were made using laboratory iodine on charcoal with NO_2. When dry air containing 0.1 percent NO_2 was passed through the charcoal, no iodine was removed during the two hour test. Some iodine was removed from this charcoal when it was subsequently exposed to air containing normal amounts of moisture. This demonstrates that the NO_2 as such does not cause elution of iodine from charcoal but that the acid products of NO_2, air and water can. The NO_2 levels of this test were much higher, however, than the ∿ 3 ppm of NO_2 in the sand filter inlet air.

The effect of methyl iodide, which is a compound found by the British to exist during some iodine release experiments, on charcoal has been examined. The adsorption of 0.8 mg CH_3I per gram of charcoal caused the release of 45 percent of the iodine previously on the charcoal. Subsequent testing showed the charcoal would retain iodine with 95 percent efficiency. The methyl iodide displaced adsorbed iodine but did not greatly hinder the later adsorption of iodine.

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WASTE MANAGEMENT AND FISSION PRODUCT RECOVERY

BAMBP Radiation Stability

Evidence that BAMBP, in a BAMBP-D2EHPA-Soltrol CSREX solvent mixture, may be more prone to radiation destruction than are the other components of the mixture - perhaps to a degree which would make Waste Management use of CSREX uneconomic - was reported several months ago. The suggestion was recently made that this destruction may be due not to a direct interaction of radiation with BAMBP, but rather to attack by free radicals produced in the Soltrol or the D2EHPA. Some pure BAMBP (no diluent) was irradiated in the cobalt source to test this hypothesis. No detectable destruction had occurred at 400 watt hours or 1.5×10^8 R, tending to confirm the above hypothesis. Additional experiments are under way to determine the source of radicals and to see whether a free-radical scavenger will afford protection.

Irradiation Stability of CSREX Solvent

Equipment has been assembled and tested which will permit studying the effects of radiation on a solvent while the solvent is in intimate contact with an aqueous phase. The equipment is installed in the Graphite Research and Development Co-60 source. Dosimeter runs are currently in progress to determine the radiation level in the mixing chamber. The equipment will be used to study radiation effects on BAMBP in CSREX solvent when in contact with appropriate aqueous phases.

Waste Packaging

An analytical solution to the problem of transient centerline temperature behavior in waste containers (strontium-90 and cesium-137 sorbed on zeolites) during the post-drying period was developed and programmed for the IBM-7090. The analytical solution specifically considers the temperature-time relationship existing following completion of high-temperature through-gas drying of the heat generating cylinder. On termination of the gas drying step (ca. 800 F), the heat removal mechanism is shifted to one of radial transfer to the surroundings. The analytical solution provides a history of centerline temperature in a heat-generating cylinder (initially at uniform temperature) when it is suddenly allowed to lose heat to a constant temperature sink with a constant heat transfer coefficient. The initial conditions represent the state of a dry, insulated, cylindrical, heat-generating molecular sieve bed through which a stream of hot gas is being passed. The final set of conditions represents the bed after gas flow has been stopped and insulation has been removed. A report summarizing this study is in prepa[redacted]

In-Tank Solidification

In-tank solidification of intermediate level plant waste will be accomplished initially by evaporation with air heated to 1200 F. Development work in support of this technique has been essentially completed except for extended runs to study scale deposition as a function of time. Examination of the finned hot air circulator from a previous one-week test revealed no detectable scaling.

Primary emphasis is now being given to development of an alternate, less expensive approach for in-tank solidification. In this concept electric heaters are immersed in the solution and a minimum amount of air is used to maintain circulation in the tank (instead of evaporation by heating the air, as discussed above). Detailed discussions have been held with several companies specializing in industrial heater design and manufacture. Each concurred as to the feasibility of this method in terms of reduced cost and heater element life. Proposals from each vendor are being evaluated and prototype heater units have been specified for further evaluation and test.

Technetium Recovery - Plant Test

Successful absorption of about a kilogram of Tc-99 from Purex alkaline tank supernate on an anion exchange bed contained in an STT was described last month. During the current month, the cask was eluted (with ca. 3000 gallons of 6 M HNO_3) and the eluate successfully evaporated (at Hot Semiworks) to a volume of only 100 liters and shipped to Chemical Research for final processing. Analyses show that technetium losses were very low and that the amount received is almost exactly one kilogram. The solution does, however, also contain 1 M iron (and lesser amounts of other corrosion products), a quantity which implies unexpectedly high corrosion of the concentrator and which suggests that pertechnic acid may be corrosive to stainless steel (in a manner analogous to chromic acid). This amount of iron will complicate further purification, as will the presence of an as-yet unidentified precipitate and a moderately high concentration of Zr-Nb-95 and Ce-144 (both probably leached from the Semiworks plant equipment by prolonged acid reflux). Experiments are under way to develop a purification scheme which will handle these impurities.

Amine Extraction of Technetium

Routine recovery of Np-237 and waste Pu-239 from Purex first-cycle (acidic) waste by amine extraction has been under investigation for some time. Some experiments were accordingly undertaken to determine

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the fate of technetium in this process. The extraction coefficients (E_a^o) of Tc-99 into 0.3 M trilaurylamine-Soltrol were found to be three from 1WW and 30 from STW (sugar treated waste). The former is more than adequate for quantitative technetium recovery in a counter current contactor (such as a pulse column), and the latter for batch contactor operation. The extracted technetium is readily stripped with NaOH or Na_2CO_3 solutions, but not with the oxalic acid wash.

Use of amine extraction would permit direct recovery of about 80 percent of the technetium which enters the Purex Plant and halt routing to the tank farm. (The remaining 20 percent, which leaves Hanford with the purified uranium stream, could also probably be routed to the acidic waste by minor flowsheet changes.) Due to coextraction of substantial quantities of ruthenium, the recovered technetium will require further purification in shielded facilities, possibly by anion exchange.

Cesium Removal from Alkaline Waste

A water scrub, recycle flowsheet was tested for ion exchange removal of cesium from Redox alkaline high-level waste supernatant solutions. In this flowsheet the same volume of water was used as the volume of 0.2 M $(NH_4)_2CO_3$ sodium scrub in previous flowsheets. After elution of cesium from the Duolite C-3 exchanger, the eluate was boiled to expel ammonium carbonate, and the solution was then pumped through a column of Linde AW-400 for a second ion exchange cycle. A water scrub was used for this column also. The Na/Cs ratio in the eluate from the C-3 column was 700 as compared to about 150 where an ammonium carbonate scrub was used. The Na/Cs ratio in the eluate of the AW-400 column was 4.5 as compared to 1.7 where the ammonium carbonate scrub was used. These data indicate that water scrub is allowable in processes for cesium recovery from Redox wastes. This change provides considerable savings in essential materials costs as well as simplifying disposal of waste from the scrub step.

A 6-liter column of AW-400, loaded with cesium from a simulated Purex alkaline supernatant waste solution, was eluted with 48 liters of 3 M $(NH_4)_2CO_3$ - 2 M NH_4OH solution. The 48 liters of eluate were concentrated to 2 liters by evaporation to determine the characteristics of the solids formed. Solids precipitated during boiling and formed brown particles in the solution and a brown scale on the surface of the flask. The solids are about 1 percent of the total volume (2 liters). Previous results have shown that the amount of solids decreases with each loading cycle. The scale in the boiling flask was readily dissolved with 8 M HNO_3. Since the solids settle rapidly after mixing, their separation is expected to be no problem.

Cerium-Sodium Equilibrium on Linde 4A and 13X

It was reported previously that Linde 13X and 4A were capable of removing appreciable amounts of trivalent rare earths from solutions containing other cations. Because trivalent rare earth removal could be of interest for the Hanford Waste Management Program, the thermodynamics of the exchange reaction in a Na-Ce system were determined. A total solution normality of 0.5 was used in the equilibrium experiments, so that solution concentration corrections were necessary. Glueckauf's equation was modified for use in a system containing trivalent and univalent cations. Concentration corrections for the cations on the zeolite also were made. The resulting thermodynamic equilibrium constants were 0.158 with 4A and 9.5 with 13X for the reaction

$$3Na_{zeolite} + Ce^{+3}_{solution} \rightleftarrows Ce_{zeolite} + 3Na^{+}_{solution}.$$

Equilibrium constants also will be determined in Sr-Ce and Sr-Na systems to ascertain the accuracy of the equilibrium results.

EQUIPMENT AND MATERIALS

Non-Metallic Materials

In static immersion tests, polyvinylidine fluoride (Kyner-Pennsalt Chemicals) and a chlorinated ether (Penton-Hercules Powder Company) were unchanged after 41 days at room temperature in 1 M HF - 13 M HNO_3 or 25 percent DBBP-CCl_4. After 39 days in the latter solution at 50 C, a polyethylene rashig ring had increased five percent in length and 38 percent in weight. The exposed ring also increased some in flexibility but was still quite strong.

Effects of Heat Treatment on Titanium Corrosion Resistance and Hardness

A series of welded A-55 titanium samples was heat treated in water-saturated air at temperatures from 1000 to 1400 F and times from one to 60 minutes. The heat treated samples were exposed to boiling simulated Purex 1WW for about 500 hours. All samples corroded at rates less than 0.1 mil/mo. Another series of A-55 samples was heat treated for ten minutes in water-saturated air at temperatures from 800 to 1900 F. Although surface colors were produced on the samples by the heat treatment, hardness changes, as measured by instruments producing two and three mil indents, were not of a magnitude to indicate significant changes in the physical properties of the metal.

Scaling of Titanium Heat Transfer Surfaces

Tests for possible scaling of titanium heat exchanger surfaces, and consequent loss of heat transfer, in Purex process waste solutions were continued. An improved design of test bayonet is being used which permits more accurate detection of heat transfer loss. A bayonet boiling simulated Purex 1WW (butted on a weekly basis with 0.01 M silica) for three weeks has shown no loss of heat transfer. A bayonet boiling simulated Purex H-4 solution at 8 M HNO_3 has shown no accumulation of silica scale; a similar bayonet boiling simulated H-4 solution at 9 M HNO_3 has acquired some silica scale but not as heavy as occurred in previous tests on simulated H-4 solution at 10 M HNO_3.

PROCESS CONTROL AND DEVELOPMENT

Air Pulser Development

A solid state controller has been developed for frequency control of air pulsing systems. The frequency of a multivibrator is set by a special oscillator circuit and controlled rectifiers are used to energize a two-way solenoid valve. Elimination of relays provides a high reliability system with costs substantially below alternate control devices.

Additional work was performed on the measurement of pulse velocity (which is directly related to the pulse amplitude-frequency product). A preamplifier with a voltage gain of 12,000 was tested in conjunction with a strain gage flowmeter. This combination provides a relatively simple system with an accuracy of 99 percent or better.

A spare pulser system using a variable speed motor and poppet valve is being procured as a back-up unit for the Plutonium Reclamation Facility. Modifications were incorporated into the earlier design to improve reliability and all parts are now on order.

Plutonium Electrowinning Instrumentation

Instrumentation has been devised for remote weighing of the plutonium product from an experimental electrowinning unit. The device is based on a spring deflection which is measured with a linear variable differential transformer; output from the LVDT is demodulated and the DC signal supplied to a potentiometer. Preliminary test results indicate a short-term weighing accuracy to within ± 1 percent with a long-term accuracy within ± 3 percent. Since short run times preceded by tare weight adjustment are planned, this unit should meet the requirements of the application.

Scintillating Glass

Corning No. 7913 scintillating glass suffered no observable physical changes during immersion in 0.25 to 4 M HNO_3 at 50 C for 440 hours and at room temperature for 128 hours. Inspection for possible changes in scintillating characteristics is under study.

Pyroelectric Temperature Measurement

A barium titanate transducer was evaluated as a means of measuring small rates of temperature change; this measurement has potential application for determining heat generation rates of fission product process streams. The transducer produces a charge proportional to temperature change. This charge is impressed across a megohm resistor and the voltage drop is measured with a vibrating reed electrometer. The combination of this megohm resistor and the internal capacitor of the electrometer result in differentiation of the generated charge, producing a voltage proportional to $dt/d\theta$. In laboratory tests temperature changes in the range 0.004 to 0.1 degree Centigrade were obtained by allowing heated water to cool in an insulated flask. Over this range the electrometer output was 0.2 to 5.5 millivolts, indicating very good sensitivity. Barium titanate is useful over the range 20 to 80 C, and other pyrcelectric devices are available for higher temperatures.

Advanced Process Control Development

Analog computer determination of the parameters for the solvent extraction model equations is essentially completed. Digital data fitting has been plagued by a numerical instability problem, requiring extensive revision of the mathematics of the computer code. In another phase of the program, solution of the non-linear model equations is being accomplished by a process of stepwise linearization. Basically, this means replacing the S-shaped uranium equilibrium distribution curve by a series of straightline segments. The line segments which result in minimum error are then determined using the principles of dynamic programming. The required expressions for the problem have been derived and a 7090 computer program has been coded and placed in service.

Development and demonstration of computerized control of a pulse column will utilize a General Electric 412 computer, which is now on order. Discussions were held with representatives of the Industry Control Department to arrange a programming course for interested Hanford personnel and to exchange information concerning equipment performance, input-output signal requirements, and installation arrangements. Extensive discussions concerning available software point up the fact that existing 412 programs and sub-routines will aid considerably in preparation of the program for optimized column control.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process Engineering Development

Operation of Salt Cycle process equipment in C-Cell of the High Level Radiochemical Facility has progressed to the plutonium-uranium stage. Unirradiated, cold-swaged, Zircaloy-clad UO_2-PuO_2 fuel rods were bisected and 10 half pins were declad in 1.5 hours in the cell with a vibratory decladder. About 32 pounds of UO_2 containing 58 grams of plutonium were removed from the cladding and partially oxidized to U_3O_8 at 450 C with air at 1.5 liters per minute for 18 hours to reduce the particle size. The U_3O_8-PuO_2 mixture was introduced into a 65-pound molten LiCl-KCl salt bath and sparged with 1.5 liters/minute chlorine - 1.5 liters/minute hydrogen chloride gas mixture. About 90 percent of the uranium was dissolved in the first 24 hours of dissolution, but none of the plutonium had dissolved at this point. After 48 hours at least 96 percent of the uranium had dissolved and about 50 percent of the plutonium had dissolved to give a concentration of 26.5 w/o U and 0.06 w/o Pu. Subsequent efforts to dissolve the plutonium with hydrogen chloride gas over a 10-day period gave only a maximum salt bath concentration of 0.079 w/o plutonium representing 64 percent of the plutonium in the fuel. The following conclusions were formed during these tests:

1. Much of the plutonium which could not be dissolved was probably in a black ring which was observed on the quartz vessel wall about 2 inches above the salt. The Pu/U ratio in a sample of this ring was 5.7 to 1.

2. No plutonium dissolved until nearly all of the U_3O_8 dissolved.

3. The plutonium content of the last sample taken of undissolved solids was 9 w/o plutonium, about 50 times the concentration in the feed material.

4. Removal of the salt pot lid for brief periods reduced the plutonium content of the batch to less than 40 percent of the maximum value because of hydrolysis of plutonium caused by atmospheric moisture.

5. Sparging with hydrogen chloride gas for 16 hours redissolved nearly all the moisture precipitated plutonium.

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6. Operation with an argon blanket does not cause appreciable quantities of salt to freeze out in the upper, colder parts of the salt system and may reduce the moisture leakage into the system.

7. Attempts to remove the plutonium above the melt interface by washing it down with salt from a gas lift were unsuccessful.

8. Elimination of moisture in the cell and reduction of air leakage into the salt pot are beneficial to keeping the plutonium in solution.

Salt Cycle Flowsheet Studies

Laboratory studies of Salt Cycle flowsheet parameters have continued to place emphasis upon determination of the factors important in the electro-codeposition of uranium and plutonium dioxides from molten chloride salt solutions. It now appears that with the use of a low current density and with a low oxygen/chlorine ratio in the sparge during the electrolysis, a rare earth decontamination factor of 30, and a plutonium enrichment factor of 50 can be achieved. Thus far, however, electrolysis under these conditions has recovered only about half the plutonium from the melt.

Chloride Contamination in Electrolytic UO_2

In an effort to determine the mechanism for the incorporation of large amounts of chloride in some samples of electrolytic UO_2, a qualitative comparison was made between the foreign metal content and the chloride content of a number of samples. The data thus far indicate aluminum, molybdenum, and niobium to be "strong carriers" of chloride. It is entirely possible that other metals will be added to this list as the work continues. It is postulated that the mechanism of chloride carrying during UO_2 electrodeposition involves the formation of metal oxy-chlorides, with consequent codeposition of such species in the UO_2 lattice.

Chemistry of Uranium in Chloride Melts

Using spectrophotometric techniques, the equilibrium constant for the reaction

$$UO_2Cl_2 + UO_2 = 2\ UO_2Cl$$

in LiCl-KCl eutectic has been determined to be 0.0016 at 500 C and 0.0050 at 600 C (concentrations expressed in molarity and the activity

of UO_2 taken as unity). These values were obtained in a series of experiments involving the reduction of uranyl(VI), initially at a concentration of 0.017 molar, by H_2-sparging the molten salt solution until solid UO_2 was present and then allowing the species to adjust to the equilibrium condition (equilibrium was approached from both directions). The concentrations of uranyl(V) were calculated from the absorbance of its peak at 1534 mμ (where uranyl(VI) does not absorb) and those of uranyl(VI) from the absorbance values in the range 490 to 470 mμ (where uranyl(V) does not absorb).

The molar absorptivity of uranyl(V) at the major peak at 1534 mμ was determined to be 25 at 500 C and 19.5 at 600 C. These determinations were made by reducing a small enough portion of the uranyl(VI) that no solid UO_2 was present and assuming the concentration of uranyl(V) to be equal to the decrease in the concentration of uranyl(VI). The data indicate that the solubility of UO_2 as such is lower than about 2×10^{-4} molar, which is an insignificant amount in these experiments.

Aqueous Processing of PRTR UO_2-PuO_2 Fuels

A two-step procedure for dissolution of PRTR UO_2-PuO_2 fuels is being investigated. Dilute nitric acid would be used to dissolve the uranium dioxide. The more refractory plutonium oxide would be dissolved in HNO_3-NH_4F or HNO_3-NH_4F-$Al(NO_3)_3$ solutions. Dissolution rate studies were made using PuO_2 powder (-325 mesh, 95 percent theoretical density, prepared by calcining plutonium oxalate two to three hours at 950 C) as the refractory oxide. At 80 C, complete dissolution of the oxide was obtained at 1 - 1.5, 3.5 - 4.5 and 5 - 6 hours in 15.5 M HNO_3 containing 0.1, 0.05 and 0.025 M NH_4F, respectively. The dissolution times were not greatly increased by reducing nitric acid molarity to 12.7. At 60 C, 5 - 6 hours were required to attain complete dissolution in 15.5 M HNO_3 - 0.1 M NH_4F. Corrosion rates for 304-L stainless steel under these conditions are being determined.

RADIOACTIVE RESIDUE PROCESSING DEVELOPMENT

Calcination Studies

Two full-level runs were made during the month in the A-cell spray calciner, one with Redox waste to which glass-making additives had been added and the other with sugar-denitrated current Purex waste. A series of runs in the "cold" laboratory calciner showed that use of borate and silica with Redox waste in ratios of Σ metal oxides/B_2O_3/SiO_2 = 1/1.5/1.5 would result in a good glass melting at ≤ 900 C.

The hot-cell run was uneventful, except for pre-run difficulties (plugged lines) in introducing saturated boric acid solutions and silica slurries into the cell. Ruthenium evolution was less than 2 percent, and a good, glassy-appearing melt was obtained. The run with the low-acid Purex waste was terminated prematurely (due to an electrical short circuit in the power system) but preliminary indication from samples taken early in the run is that ruthenium evolution was very low with this feed, also implying that low acidity was probably responsible for the favorable ruthenium behavior in both cases.

Continuous Glass Making

Laboratory study continued of (1) sulfate segregation in ORNL-type lithium sulfate glass, and (2) phosphate glasses derived from Purex-type waste containing fission products equivalent to high-burnup power reactor fuels. Meanwhile a "cold" mockup of equipment for hot-cell testing of the Brookhaven glass process was assembled and testing initiated.

Observation of a sulfate-rich second phase in ORNL lithium glass was reported last month. Further experiments have shown that formation of the second phase is critically related to the sulfate to phosphate ratio: $[(SO_4^{=})/(SO_4^{=}) + (PO_4^{\equiv})]$ (where the parentheses refer to chemical equivalents rather than molar concentrations). For values of the ratio ≤ 0.12, a single phase was obtained. Above a ratio of 0.18, two phases were formed. The drip temperature increased markedly (from 744 to 926 C) between a ratio of 0.18 and 0.24.

In other laboratory experiments, incorporation of mixed fission products equivalent to 0.5, 1, 2, 3 and 4 x 10^4 MWD/T in a Purex phosphate glass resulted in all cases in a polyphase melt at 1000 C. Increasing the temperature to 1275 C still did not produce homogeneous systems. However, the resultant solids were hard and strong and may represent a satisfactory storage media, even though inhomogeneous.

A mockup of equipment for the proposed hot-cell continuous glass test (HCCGT) was operated at month's end and certain design deficiencies revealed. Following replacement of defective or failed equipment pieces, testing will resume.

Cold Semiworks Spray Calciner

In continuing studies to define the design and operational parameters of a spray calciner, recirculation rates for the 8.5-ft. and 4.75-ft. long by 18-in. diameter reactors, using a 14-in. diameter draft tube,

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were independent of reactor length within an experimental error of about 20 percent. Recirculation rates for the 4.75-ft long reactor using a 16-in. diameter draft tube were only 1/2 to 1/3 as great as the rates for the 14-in. diameter draft tube.

Water capacities were measured for the short reactor using a 16-in. draft tube with and without an extended heat transfer surface (fins). The fins had an insignificant effect on capacity.

Intermediate-Level Waste Treatment

A pilot plant run (No. 4) was performed to demonstrate the efficiency of a three-bed demineralizer for removing radionuclides from steam-stripped Purex tank farm condensate. The run was terminated after 450 gallons of feed had been processed due to excessive pressure drop across the thin bed exchanger. The increased pressure drop was due to an unidentified solid which appeared in the feed 2 to 3 days after receiving the condensate. At the time the run was stopped the cesium DF across the thin bed of clinoptilolite was 1.7, and the ruthenium DF across the strong base anion exchanger was about 4. These DF's are much lower than those obtained in earlier runs with solids-free condensate which had a much higher radionuclide concentration.

Steam stripping the condensate within 2 days appears to reduce solids formation. Run No. 5 was started after modifications were made to permit steam stripping within 2 days after receiving the condensate. This run had to be stopped after 2100 gallons of feed had been processed, due to excessive pressure drop across the thin bed and anion exchanger.

Studies are continuing to find the source of the solids and to modify the process to handle solids, if present.

CONTAINMENT SYSTEMS EXPERIMENT

Upon authorization for the Containment Systems Experiment, immediate action was taken to form an organization which has the overall responsibility for the program.

The location of the experiment will be in T-Plant canyon head-end section rather than U-Plant or 200N as originally proposed. The T-Plant Galleries will be used for control stations, offices, shops and other necessary facilities.

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Preliminary analysis of the design and fabrication schedule shows that the containment shell and reactor simulator are critical items in regard to procurement time required. All other work is of shorter duration and can be fitted to the schedules of the critical items.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

The test model, 662-node electrical analog network was constructed, and all cabling connections were made. Testing and final readout of the model is in progress. A program was prepared to get the analytical solution for potentials at each node point on the analog. Comparison of these results with the electrical potential readout by the scanning equipment will permit evaluation of the accuracy of resistance network duplication. Also, experience related to operating characteristics (noise problems), assembly techniques and equipment adaptability is being obtained from construction and operation of the test model.

Careful consideration of transient flow in heterogeneous porous media turned up a condition related to the permeability, K, (a function of spatial location but independent of time) which may result in reducing the field measurement efforts needed to obtain the boundary permeability conditions required for the ground-water analog. Further pursuit of this solution method is being made through derivation and analysis of the "streak function" which defines the locus of particles which have passed through a common point. The utility of the streak function in obtaining boundary values appears encouraging, but considerable more work and study are needed before the existence and application of such an additional mathematical tool can be evaluated.

Seismic refraction line 1 of the geophysical program crossed a low point on the basalt surface about a mile southeast of the Hanford Wye Barricade and well 699-15-15. Basalt was 100 feet deeper than at the well and 44 feet deeper than the previously known deepest occurrence of the basalt surface, beneath 100-B Area. Airborne magnetometer survey data and the seismic data suggest that the depression extends northeastward a mile or more, possibly to a lower point. Improved knowledge of the basalt surface there, and accordingly of the basalt structure and the overlying, conformable Ringold Formation, will help better define the probable direction of waste movement to and at depth from 200-East Area toward the Columbia River.

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RADIOLOGICAL AND HEALTH CHEMISTRY

Bioassay Techniques

Initial application of multidimensional gamma spectrometry to the determination of radionuclides in urine showed that Na-24, K-40, Cu-64, Cs-137 and an as-yet unidentified radionuclide with gamma activities at 0.5 and 1.8 mev could be readily determined in an overnight count on a 24-hour sample. After decay of the Na-24, the radionuclides Cr-51, Zn-65, Sc-46, Co-60 and perhaps Np-239 can be measured. These samples were obtained from persons whose drinking water is derived from the Columbia River (in this case, Kennewick residents). The excretion rates of these nuclides will be compared with their concentrations in the drinking water and foods.

Tracers for Meteorological Studies

Electronic pulse counting techniques and cooling of the photomultiplier to reduce thermionic emission were applied to fluorescence counting to provide increased sensitivity as was obtained in phosphorescence counting. With the Farrand Spectrofluorometer used, fluorescein could be determined down to 5×10^{-11} g (improvement by a factor of 20). Further improvement is expected when the IP21 photomultiplier tubes are replaced with selected DuMont 6292 tubes. Some preliminary dual tracer analytical techniques using fluorescein and aspirin as the tracers showed promise.

Radiation Chemistry

The protection index of phenol was measured as a function of temperature. The data show the same pattern of increase of protection index with decreasing temperature as has been found with all the other solutes examined. The increase was not very great however, perhaps indicating that phenol is rather similar to erioglaucine in its ability to organize a water sheath about itself. Further work will be directed toward obtaining experimental evidence bearing on the postulate of water structure about solutes.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Particle Sampling Studies

Studies of errors resulting from sampling air streams at subisokinetic flow rates continued. Tests for a windspeed of 2.7 mph and a subisokinetic sampling ratio of 11.4 were performed. The 2.7 mph data show sampling errors of two to four for 18 μ particles and an error of six

for 6 μ particles. This error is much larger than expected for the smaller particles and for the lower windspeeds.

A significantly non-uniform pupulation density for 18 to 25 μ particles was noted in one of the 2.7 mph tests. The non-uniformity corresponded to the location of the central solid dowel in the filter packing, more particles being collected on the lower half of the region. Tests at no flow through the filter showed the larger particles were collected on the lower half of the filter due to the action of gravity. These gravity forces have not been noted for smaller particles.

ISOTOPES DEVELOPMENT - 08 PROGRAM

Ion Exchange Purification of Promethium

A program of column elution experiments aimed at the development of a high capacity, acid-side, chromatographic process for promethium purification was reported last month, as was the finding that HEDTA possesses certain apparent advantages over either EDTA or DTPA. In a run made this month (the third with HEDTA) a concentration of 0.030 M was successfully used. This is twice the concentration that can be employed with either DTPA or EDTA (because of the limited solubility of the EDTA- and DTPA-rare earth complexes). Because of the higher eluate concentration, rate of band movement was approximately doubled. Purity achieved was apparently not compromised by the shorter time cycle. Additional runs will be made with HEDTA to determine the maximum HEDTA concentration that can be used, the optimum pH, and the minimum elution distance consistent with good separation. Additional columns are being installed to expedite these studies.

Pneumatic Compaction of Fission Product Heat Source Materials

A criticism of the high-energy-compaction process for isotopic fuel encapsulation has been the relatively thick can walls so far employed (to avoid wrinkling). Because of the incentives, particularly in aerospace applications, for minimizing weight, ways are being sought to produce acceptable thin-walled compactions. Several schemes are being tried. The most direct approach is to use a thick walled can, designed to give uniform wall thickness, and to then machine off unwanted surplus metal. This was done in a trial strontium titanate compaction. Following compaction, cladding thickness was measured by ultrasonic techniques (Non-Destructive Testing Operation) and the can walls then machined to 3/32-in. thickness. Following retesting, the compact will be sectioned and examined. A second approach used a thin-walled stainless steel can surrounded by a tight-fitting mild

steel jacket, which can bs subsequently dissolved in acid. This compaction also looked very good, but has not yet been sectioned. Other techniques are also under investigation.

Supporting laboratory work is being directed at the preparation of fission product compounds (of Cs, Sr, Ce and Pm) suitable for high energy compaction and at the development of promethium forms suitable for aerospace application. Plans are also being made for hot-cell preparation of certain of these materials to evaluate the (possibly deleterious) effects of radiation and isotopic decay.

E. R. Irish

for Manager
Chemical Laboratory

WH Reas:cf

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BIOLOGY LABORATORY

A. ORGANIZATION AND PERSONNEL

Dr. Jacob Eapen of the Biology Division, Atomic Energy Establishment, Bombay, India, began a one-year assignment with this Laboratory on November 25, 1963. He is sponsored by the International Atomic Energy Agency

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Hydrazine Hydrate

The possibility of the use of hydrazine-hydrate solutions to clean some NPR pipe systems initiated a study of the toxicity of this material to fish. Bioassay continues, but early work indicates the material can be highly toxic to fish. Concentration of 6 ppm can be 100% lethal to young trout.

Vertan 675

Another descaling composition, Vertan 675, a similar but different formula from Vertan 690 is proposed for use in the primary loop of NPR. Further tests are needed to provide reliable estimates of critical lethal concentration to fish, but preliminary tests clearly show the greater toxicity of Vertan 675 compared to Vertan 690. Previous estimate of 1500 ppm Vertan 690 for LD_{10} compares to a 100% kill at 560 ppm Vertan 675 in our early test.

Columnaris

The necessary contacts and arrangements with Washington State Department of Fisheries to sample fish for columnaris at the artificial spawning channel project at Priest Rapids Dam were made. The strategic site of Priest Rapids Dam just upstream from Hanford permits a promising study of the significance of columnaris on the local fall-run chinook salmon spawners. A number of adult chinooks have already been sampled. Water from the spawning channel sampled on October 31 provided an estimate of 1.25 organisms per ml water. Despite the presence of micro-organisms the chinooks are spawning with a high degree of success as indicated by little egg retention by the females and successful deposition of eggs in the spawning channel.

Observations continue for columnaris in groups of trout reared under three different temperature conditions, normal river water, 4F warmer and 4F cooler than river water. For all practical purposes mortalities have

now ceased during our cooler winter temperatures. This study at 22 weeks clearly shows the adverse effect of 4F elevated temperatures which was expressed by higher mortalities and growth depression. No marked difference in mortalities occurred between the normal river water group and the -4F group, but growth was considerably better in cooler water.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Zinc

Serially killing of trout fed single oral doses of 230 μc Zn^{65} to observe distribution and retention continues. At 141 days post-administration the average body burden is estimated at about 6% of original dose or 14 μc of which more than 50% is found in the gastrointestinal tract.

Strontium

In miniature swine ingesting Sr^{90} daily hematopoietic tissue is one of the two tissues which receives by far the highest radiation doses (bone receives the highest dose). We have, therefore, given some attention to it with regard to developing sensitive techniques for determining radiation damage to it. Under our experimental conditions animals ingesting up to 125 μc Sr^{90}/day are able to produce erythrocytes in adequate numbers to maintain a normal level of circulating erythrocytes. Since it is well established that in the normal individual there exists a considerable reserve of hematopoietic tissue, it would seem possible in these animals where the majority of the bone marrow is receiving up to 15 rads/day that a portion of this reserve is being used to maintain normal blood cell numbers. If this is the case, then, it may be anticipated that the animal will have a lesser reserve to utilize in the event of additional stress.

To determine if the "reserve capacity" of miniature swine ingesting Sr^{90} and their controls is altered, some are being placed in a hypoxic environment (a large vacuum chamber) and their ability to respond to this stress determined. The hypoxia in this case serves as a stimulus for increased levels of erythropoietin to be elaborated, which is then a stimulus for increased erythrocyte production. The erythrocyte production of the animals is then estimated by measuring the uptake of an intravenously administered dose of Fe^{59} by the developing erythrocytes. Under these conditions where there is maximum stimulation of erythropoiesis, the Fe^{59} uptake is an indicator of the "reserve capacity" for erythrocyte production.

Analytical results are not yet available. Initial experiments indicated the desirability of modifying the existing vacuum chamber to permit the animals to remain in the hypoxic environment for extended periods of time.

Very preliminary data were obtained on the neutrophil reserve of miniature swine ingesting 25 μc Sr^{90}/day using single injections of bacterial endotoxin. These results indicate that the neutrophil reserve of these animals whose bone marrow is receiving 3-4 rads/day is similar to that of control animals.

Neptunium

The incorporation of C^{14}-labeled acetate into liver fats was stimulated threefold by Np^{237} as early as three hours after administration to female rats. Male rats which exhibit a much lower toxicity to Np^{237} show decreased $C^{14}O_2$ production from C^{14}-labeled palmitic acid. The decrease, about 25% less than controls, was smaller than that previously observed in female rats.

Plutonium

Experiments involving perfusion of various segments of the intestinal tract and cannulation of the bile duct are being performed to study secretion of plutonium into the intestinal tract following administration of various chemical forms of plutonium by various routes (intravenous, intratracheal and subcutaneous). Following intravenous plutonium citrate administration, the largest secretion occurs in the duodenum followed in decreasing order by the jejunum, ileum, large intestine and cecum. Plutonium-burdened adrenals have been transplanted to plutonium-free rats with erratic success. A plutonium burden of 0.01 μc/g of adrenal does not appear to affect early regeneration of the transplant. Difficulties are concerned principally with graft-host response.

Rats were splenectomized to study the effect of Pu^{238} and Pu^{239} on animals lacking the reserve hemopoietic capacity of the spleen. No results are as yet available.

Earlier work on the effect of Na_3Ca DTPA on Pu^{239} retention following a subcutaneous injection of Pu^{239} (IV) nitrate suggested that increasing the amount of DTPA injected intravenously following contamination reduced the amount of Pu^{239} retained by the liver and skeleton. Additional experiments indicate that when DTPA was injected at a level of 0.0014 g/kg (one-tenth the recommended human therapeutic dose) a reduction to about 75% of that in controls occurred 16 days after Pu^{239} injection. Increasing the quantity of DTPA injected to 0.014 g/kg or 0.14 g/kg effected a further reduction to about 50% of the control value. There was no appreciable difference between these latter two treatments. (Earlier work showed suffusion of the Pu^{239} injection site with 0.017 g Na Ca DTPA either immediately following Pu^{239} injection or four hours later did not appear to cause gross movement of the Pu^{239} from the injection site and produced only a slight reduction in the liver and skeletal Pu^{239} content.)

Inhalation Studies

In rats DTPA and TTHA removed 25% of inhaled plutonium nitrate, while diethyldithiocarbamate was ineffective 30 days after exposure. After DTPA treatment the plutonium content of all tissues other than lung was about 7% of the lung burden compared to about 25% for the other treated animals and the controls.

In dogs three months after exposure, the biological half-life for retention of Ce^{144}-Pr^{144}, inhaled as CeO_2 calcined at 300°+ is 550 days. In earlier experiments the biological half-life for Ce^{144}-Pr^{144} inhaled as $Ce^{144}O_2$ prepared by chemical oxidation was less than 400 days.

Recent examination of all dogs in our colony showed an unusually high incidence of retinal degeneration which was seen in only one dog last year. Although this finding is not confined to the plutonium-treated animals, the incidence is higher in this group. A number of dogs also have lens opacities. The results of the examination require further study before conclusions can be drawn. It is thought to be hereditary in some cases and in others it is associated with previous conditions of distemper, uveitis, chorioretinitis, and nutritional deficiencies.

Gastrointestinal Radiation Injury

About fifty rats were employed in an experiment measuring the leakage into the intestine of I^{131}-labeled polyvinylpyrrolidone (PVP) as an indicator of intestinal radiation damage. Comparisons were made on effect of radiation dose, pretreatment with cysteine or AET, and bile removal or replacement. Results are not yet available.

Irradiation of Intraperitoneal Fluid

Experiments were performed in preparation for studying the effect of in vivo irradiation by alpha emitters enclosed within dialysis tubing implanted intraperitoneally. No differences were noted in the electronic spectra of peritoneal washes from rats receiving 0, 400 and 1,000 r X-ray.

Secondary Disease Studies

Liver cytochemical studies of secondary disease were conducted in xenogeneic chimeras (mice injected with rat cells after 950 r X ray). Leukocytic infiltration in the periportal sinuses was not evident during the first two weeks post-treatment, but increased significantly at three weeks. The number of littoral cells per high power field increased significantly at two and three weeks post treatment.

Encouraging results are being obtained in the use of the Coulter counter for cytotoxic assay. Spleen and bone marrow cells have been successfully employed as target cells.

Microbiology

In preparation for studies of metabolic channeling and feedback mechanisms, two enzymes (tryptophan synthetase and tryptophanase) were isolated and partially purified. These enzymes are required for preparation of radioactive substrates to be utilized in experiments dealing with channeling of tryptophan intermediates in Neurospora.

More precise identification of amino acids leaking from irradiated and non-irradiated yeast cells was attained using two dimensional chromatography with carboxymethyl cellulose than has previously been attained. Medium from non-irradiated cells contained small amounts of glutamic acid, isoleucine, histadine, lysine, and serine. Higher concentrations of these same amino acids plus threonine were present in media from irradiated cells.

Radiation Effects on Insects

In Tribolium studies, upon obtaining an adequate balance, it was found that weighing of flour beetles was a satisfactory and more rapid method for counting adults.

Studies of temperature and radiation effects showed that reproductive ability increased with temperature rise between 25 and 32 C but decreased with increasing X-ray doses up to 2625 r. In mixed species tests, T. castaneum was more prolific than was T. confusum.

Three-week-old virgin beetles were exposed to 3-4 Mev fast neutrons from the Van de Graaf generator, and single pair matings made to compare with results from X-ray tests. Results are not yet available.

Plant Nutrition

Alfalfa from plots contaminated in 1954 with Sr^{90} and in 1963 with Cs^{137} was harvested at the end of its first year of growth. The Sr^{90} concentration was about twofold lower than found for beans in 1962, and about the same as for barley in 1957 and 1958. Concentrations of Cs^{137} in the alfalfa were about 50-fold lower than concentrations of Sr^{90} when the same amounts of the two isotopes were present in the soil.

Plant Ecology

Greasewood leaves were observed to accumulate sodium during the spring and summer, with a maximum of 90 mg Na/g dry leaf in early September. Some loss thereafter might be due to movement back into woody tissues, to differential leaf drop, or to leaching of sodium from leaves.

Columbia River Limnology

An error in counting was found which renders previously reported Zn^{65} values in error. More proper values are now being obtained.

Periphyton, as determined by dry weight and by chlorophyll a, decreased markedly during the month. On October 28 there were 400 nc P^{32}/g organic matter in periphyton samples.

Average radionuclide content of plankton on October 23 was 82 nc P^{32}, 820 nc Cr^{51}, 190 nc Zn^{65}--all per gram organic matter. This is a substantial increase of Zn^{65} and Cr^{51}, over last month's averages.

Techniques

Commercially prepared paper chromatography strips (Urograph, Warner-Chilcott) for blood urea nitrogen determinations are being evaluated. Preliminary results compare favorably with evidence of renal damage assessed by Hippuran I^{131} renograms. Further comparisons are in progress using another method of BUN determinations and correlating the results with the degree of renal histopathology.

Hippuran I^{131} renograms of miniature swine given various dose levels of $HgCl_2$ intravenously reveal complete loss of kidney function almost immediately which persists for several days and is followed by a return to a near normal excretion pattern.

Because crayfish have a structure, the gastrolith, which functions as a calcium pool for the hardening of the integument during molt, interest centers on studying the growth of the gastrolith to increase knowledge of the metabolism of radioactive strontium and calcium. The technique is under development to induce molting by eyestalk extirpation and to follow the growth of the gastrolith by an X-ray procedure.

HA Kornberg

Manager
BIOLOGY LABORATORY

HA Kornberg:es

TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORY

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposia

Hanson, W. C. Project Chariot. Am. Soc. of Civil Engineers, Tri-Cities Branch, Kennewick, Wash. November 14, 1963.

b. Seminars (Off-Site and Local)

Eberhardt, L. L. Sampling ecosystems for fallout radionuclides. OSU, 11/6.

Bustad, L. K. Physio-pathological effects of ionizing radiation. Toxicology Course, College of Veterinary Medicine, Washington State University, Pullman, Wash. November 11, 1963.

Hungate, F. P. and H. A. Kornberg. Members of a panel discussion for Health Physics Society Meeting, Richland, Washington. November 19, 1963.

c. Seminars (Biology)

Hanson, W. C. Radioactivity in northern Alaskan eskimos. Nov. 6, 1963.

Stuart, B. O. Physiological behavior of inhaled isotopes. Nov. 13, 1963.

Thompson, R. C. Present status of the Pu problem. Nov. 20, 1963.

Rickard, W. H. Ecological studies in sagebrush and greasewood communities. Nov. 27, 1963.

d. Miscellaneous

Hanson, W. C. Radioecological studies of Northern Alaska. Richland Rod & Gun Club, Richland. November 4, 1963.

Hanson, W. C. Hanford Radioecological studies. Burbank Mens' Fellowship Group, Burbank, Wash. Nov. 5, 1963.

Hanson, W. C. Radioactivity in northern Alaskan Eskimos and their foods. Pasco Methodist Church Mens' Group, Pasco, Wash. Nov. 12, 1963.

II. Articles Published

a. HW Documents

None

b. Open Literature

Casey, H.W., R.O. McClellan, W.J. Clarke, and L.K. Bustad. 1963. Iodine-131-labeled rose bengal blood clearance as a liver function test in sheep. Am. J. Vet. Res. 24: 1189-1193.

b. Open Literature (continued)

Bustad, L. K. 1963. Biology of radioiodine. (Review of Symposium). Nature 199: 1142-1144.

III. Visits and Visitors

a. Visits to Hanford

11/5/63 - W. P. Bebbington, Savannah River Operations, Aiken, S.C. Discuss research with HA Kornberg and staff.

11/5/63 - Robert Rucker, Western Disease Lab, Seattle, Wash. Discuss research with RE Nakatani.

11/5/63 - M. E. Ensminger, Consultants, Agri-Services, Clovis, Calif. Discuss research.

11/11/63 - Dr. R. A. Neve' and 8 students from Biology Department, Seattle University, Seattle, Washington. Tour and discuss radiation research.

11/11/63 - Stuart P. Davey and wife. Formerly with US Fish and Wildlife Service, St. Paul Island, Alaska. Consult on Alaskan research with FP Hungate and staff.

11/12/63 - Drs. Harvey and Berkut, AEC, Division of Biology and Medicine, Washington, D.C. Consult on research with Dr. Kornberg and staff.

11/13/63 - Frank Howell and associates (3) from the Entomology Research Laboratory, Yakima, Washington, consulted with Drs. Hungate and Erdman on insect sterilization.

11/13/63 - Gary Finger, Department of Fisheries and Game, Juneau, Alaska. Discuss Alaska fishery problems with D.G. Watson and W.C. Hanson.

b. Visits Off-Site

11/3/63 - Glenda S. Vogt conferred on fish hematology at the Western Fish Disease Laboratory in Seattle with Drs. Yasutake and Klontz.

11/4 & 26 - B. O. Stuart obtained air samples of mine atmosphere at the Dawn Mines, Ford, Wash.

11/4-8 - W. M. Matchett presented a paper at the American Society of Cell Biologists in New York.

11/4-8 - E. M. Uyeki consulted with Dr. Ornstein on a microspectrophotometer at Mt. Sinai Hospital and attended the meeting of the American Society of Cell Biologists in New York.

11/6-8 - R. E. Nakatani, M. P. Fujihara, L. L. Eberhardt, and C. E. Cushing attended a symposium at Oregon State University in Corvallis sponsored by the Department of Statistics.

11/8 - W. H. Rickard and K. Price discussed ecology with Dr. Daubenmire at the Washington State University, Pullman.

b. Visits Off-Site (continued)

11/11/63 - L. K. Bustad presented a lecture to the Toxicology Course at the Veterinary School of Medicine, Washington State University, Pullman, Washington.

11/11/63 - V. G. Horstman inspected feed in Hermiston, Oregon.
11/14/63 - J. M. Dean and P. A. Olson obtained salmon eggs from the fish hatchery in Underwood, Washington.
11/14/63 - W. J. Clarke discussed pathology with Dr. Spencer at Washington State University, Pullman.
11/14-15 - W. J. Bair presented results of plutonium biological studies at the Nuclear Safety Working Group Meeting in Washington, D.C.
11/15 & 20 - L. L. Eberhardt, C. E. Cushing, and W. H. Rickard collected fallout samples at the Wooten Game Range.
11/20-22 - F. P. Hungate and M. F. Sullivan attended a meeting on radiation biology applied to space at the General Electric Valley Forge Space Technology Center in Philadelphia. Contact was with Dr. Price.
11/22 - L. K. Bustad discussed research with Drs. Gorbman and McKenzie at the University of Washington in Seattle.

APPLIED MATHEMATICS OPERATION

MONTHLY REPORT - NOVEMBER, 1963

ORGANIZATION AND PERSONNEL

J. L. Westra, Technical Graduate, joined the group on November 1, 1963, for a three month assignment.

OPERATIONS RESEARCH ACTIVITIES

Development of a Reactor Outage Information System continues. The outage accountability data file using 9 PAC has been completed and debugged. The routine quarterly report previously requested has been updated to handle final quarter 1963 data.

For H-Reactor, a library of job cards has been devised for the outage planning, recording, and file aspects of the information system. These cards are to be used in connection with the planned major outage of January, 1964. Among other things, the use of these cards is expected to help delineate operation tasks from maintenance tasks.

Final development of a Maintenance Information System for 100-B and 100-C has been held up pending the clarification of areas of responsibility.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

N-Reactor Department

The analysis was completed and a report issued of test data relative to the precision inherent in a proposed method of measuring deformation of metals under stress.

A study was initiated to assess the feasibility of predicting variation in the thickness of fuel cladding, employing pre-extrusion measurements of the uranium billet and the zirconium shell.

A tentative approach to the problem of assessing the integrity of the underground piping material through which water is currently supplied to the reactor areas was outlined.

Continued assistance was given in the development of a specific proposal for allocating over-all reactor production to the individual producing elements.

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Empirical formulae, suitable for routine data process handling, were developed relating enthalpy and specific density to temperature.

Chemical Processing Department

A suggestion to increase the minimum plutonium content of alpha components to XX.70 and the concomitant use of $\bar{R}$, $\bar{X}$, charts to demonstrate compliance was discussed with representatives of the sites involved. CPD will present a summary of all data for this year and a counter proposal to retain the use of a statistical tolerance statement for demonstration of compliance with minimum plutonium content specification.

A review was made of available replicate gaugings to provide an estimate of random measurement error in determining compliance of weapon components with dimensional specifications.

Work on an outline delineating a proposed Z-Plant Accountability Model was continued.

Irradiation Processing Department

An analysis is being made of data from an experiment to assess the effects of various heat treatment variables in the preparation of uranium ingots and dingots on the preirradiation warp of fuel element cores.

The batch, sample, and measurement errors in the X-ray and spectrographic measurement of the percent Si, Ni, Fe, and Pb in canning pots were estimated. Comparisons were also made between the X-ray and spectrographic measurements.

The relative worth values are measures of the poisoning effects of control rod sections compared to a standard. An analysis was made to see if significant differences exist between these values for different heats (batches) of material. One-sided tolerance statements were also determined and variance components estimated.

Further work was done in comparing the preirradiation dimensional and weight characteristics of fuel elements manufactured by the AlSi and hot-die-sizing processes. The average, standard deviation, 95 percent confidence limits on the mean, and tolerance statements were determined for each characteristic. The variances of each characteristic for the two processes were compared as an indication of process reproducibility capabilities.

An empirical formula for predicting the melting point of an alloy as a function of the percentages of the three constituent elements Al, Sn, and Zn was determined. The formula is restricted to a certain range of the three constituent elements.

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Stud pull data on fuel elements canned by the hot-die-sizing process were analyzed. The results indicated that the average pull strengths along the fuel element are greatest at the base and cap ends. Results were consistent with previous tests.

Work is continuing on the analysis of bonding integrity measurements of HDS fuel elements in order to determine what the optimum preheat times and furnace temperatures are.

An analysis is being made to determine the nature of the temperature distribution in HDS fuel elements during the preheat, sizing, and cooling phases of the operation.

Data from an alpha counter are being analyzed in order to determine if the data are of a homogeneous Poisson nature, to detect whether drifts occur and to assess the accuracy of different methods of calibration. The probability of an alpha particle's being emitted from the fuel element was also calculated for particles issuing within the mean length's distance from the surface of the fuel element.

Work is being done on the statistical analysis of PT-572A which has as its objective the characterization of the irradiation behavior of self-supported fuel elements in smooth process tubes.

A revised system for systematically probologging and replacing tubes is being reviewed for its statistical correctness.

A considerable number of linear models have been fitted to calibrate probolog measurements with wall thickness values. In the calibration, the wall thickness is the known independent variable but in the application of the formula it becomes the predicted value. A data processing program is being developed which will use the regression line in reverse and put confidence limits on the predicted wall thicknesses.

Extensive amounts of data on process tests to evaluate the performance of Diversey 514 while extending its solution life by pH control are being analyzed. The analysis consists of determining whether there are adverse drifts in such bond integrity measurements as internal and external bond count, total count, and total bad discs.

An analysis is being made of the errors in the preirradiation dimensional measurements of hot-die-sizing fuel elements.

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STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HL

2000 Program

Checking and programming of the boundary value approach to the solution of the system of simultaneous nonlinear differential equations expressing mass transfer in the column as a function of the column position continued. Numerical techniques for performing eigenvalue and eigenvector calculations on nonsymmetric matrices were studied in connection with the final step in the mathematical solution of the boundary value problem.

Data on the formation and release of radioactive film from the surface of process tubes into the effluent stream is being studied in an effort to construct a mathematical model of this complicated process.

3000 Program

The EDPM program which prepares the magnetic tape input to the prototype Sheffield rotary contour gauge has been completed and debugged. Documentation will begin as soon as trial runs can be made on the actual equipment once it arrives on plant.

Work has begun to design metal blanks for the power roll forming process which can be cast in a double cavity mold. Because of the limited size of the casting furnaces, the modified uniform shear concept is no longer applicable.

4000 Program

Closed form solutions were obtained to two problems which arose in connection with instrument development work. One problem was concerned with determining the volume of a complex geometrical body, and the second involved ascertaining the critical values of a set of parameters so as to maximize a particular analytic expression.

Additional computer runs were made with the earlier-developed EDPM program which analyzes the heat transfer characteristics in a waste disposal concentration tank.

Analysis continued on the theory of sonic transmission in layered media which possesses linear visco-elastic properties.

Work continued on a formal Hanford report describing the quantitative metallographic techniques developed during the past several years for estimating the properties of a second phase particle embedded in a matrix. These techniques are being used to estimate fission gas pore distributions in radiated uranium

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fuel. Weekly meetings are being held concerning the analysis of the large body of metallographic data collected during the last several years on the swelling studies program. To determine the magnitude of reader error and bias, seven different persons are reading a number of photomicrographs on the Zeiss particle size analyzer.

A preliminary definition of the REM retrieval pass has been completed and submitted to the customer for his approval.

5000 Program

Calculation of the initial set of channel group probabilities for inclusion in the IRA system has been completed and comparison is being made with the empirical values previously used. A study of the variability of these probabilities and the effect on the goodness of fit criterion and the standard error estimates in the GEM program is in progress.

Work was started on the calculation of the power function of the Poisson index statistic used to check stability of radioactive counters. Current efforts are directed toward writing several subroutines for the calculation of the cumulative distribution function of a linear combination of non-central chi-square variables. The power function can be represented as a convolution of this distribution with that of a normally distributed variable.

Additional analysis and EDPM programming were completed on the crystal indexing problem.

6000 Program

Closed form solutions were obtained to a set of four simultaneous, non-homogeneous, linear, ordinary differential equations which represent the first attempt at a mathematical model of a biological uptake and retention process.

The statistical analysis of data from a study to determine what effect the photoperiod and time of day a blood sample is taken has on the amount of blood glucose measured in a crab has been concluded.

A statistical analysis of data from an experiment to investigate the effect of incubation and rearing temperatures on the growth of cichlids was completed.

Data on adult progeny from neutron exposed Tribolium castaneum were statistically analyzed. One purpose of this study was to investigate the effect that neutron exposure of one member or both members of a mating pair and temperature have on age at onset of reproduction. Another purpose was to investigate the

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effect of these same two factors on the average number of adult progeny produced per mating pair.

OTHER

Equations of an empirical nature, which predict the amount of Pu-239, 240, 241, and 242 as a function of the percent burnup have been fitted.

An estimate of the total amount of Pu in the recuplex crib was estimated on the basis of earth samples taken from different crib locations.

Assistance was given in evaluating the measurement variation in optical and micrometer measurements of graphite expansion.

Carl A. Bennett

Manager
Applied Mathematics

CAB:dgl

[redacted]

[redacted]

REACTOR PROGRAM - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Reduced Spatial Concentration Study

Fuel costs were calculated for thorium enriched with plutonium of two compositions at various densities for very hard spectrum thermal reactors having a moderator-to-fuel-volume-ratio of 1.0 and 1.5, using 1/2-inch uniformly spaced rods. The results should be considered tentative because the MELEAGER model has not been closely calibrated for hard spectrum, plutonium enriched thorium cases. However, the trends indicated are reasonable and of sufficient interest to warrant reporting. Employing plutonium to enrich thorium, the results indicate less sensitivity of fuel costs to variations in spatial concentration than is the case for enrichment of U^{238} with plutonium. Curve A on Figure 1 shows minimized fuel costs for thorium enriched with plutonium (of composition 95% Pu^{239}, 5% Pu^{240}). The data represented on Figure 1 are based on a batch cycle. The same trends exist for graded cycles although the fuel cost is usually less. The small change in fuel cost for the density range 6 to 9 grams per cubic centimeter effective fuel density* is partly due to the very large effective Pu^{240} cross section that resulted. In this range, the reduction in thorium density did not reduce fertility of the fuel to the point where it was uneconomical. In Curve B of Figure 1 overfertility (from the standpoint of having sufficient reactivity for reactor operation) did not occur at any density. However, the most economical density (and corresponding optimum fertility for this reactor) appears to be about 6 grams per cubic centimeter effective fuel density.

Figure 2 shows the effect of changed fuel use charge (AEC interest rate) on the optimum density, again based on a batch cycle. At higher fuel use charge rates, it becomes increasingly important to achieve the optimum fertility.

Seed Blanket Studies

Calculations for seed blanket reactors using the "hot" control rod design were begun. This design visualizes part of the seed (the "hot" rod) as moveable such that the reactor is subcritical with the rod removed. Maximum neutron economy and minimum power peaking in the seed can be achieved with this design. The calculations are based on a cylindrical representation of the seed blanket module which, in practice, would probably be square or hexagonal. Modules of 10, 20, and 30 centimeter radii were studied with seed volumes of 5, 10, 20, 30, and 50 percent. At month end doubts were cast on

*Effective fuel density is defined to be the density in grams per cubic centimeter of the actinides (this word includes the fuel isotopes of interest) in the fuel.

Plutonium Compositions:

Curve A - 95% Pu^{239}, 5% Pu^{240}

Curve B - 65% Pu^{239}, 25% Pu^{240}, 8% Pu^{241}, 2% Pu^{242}

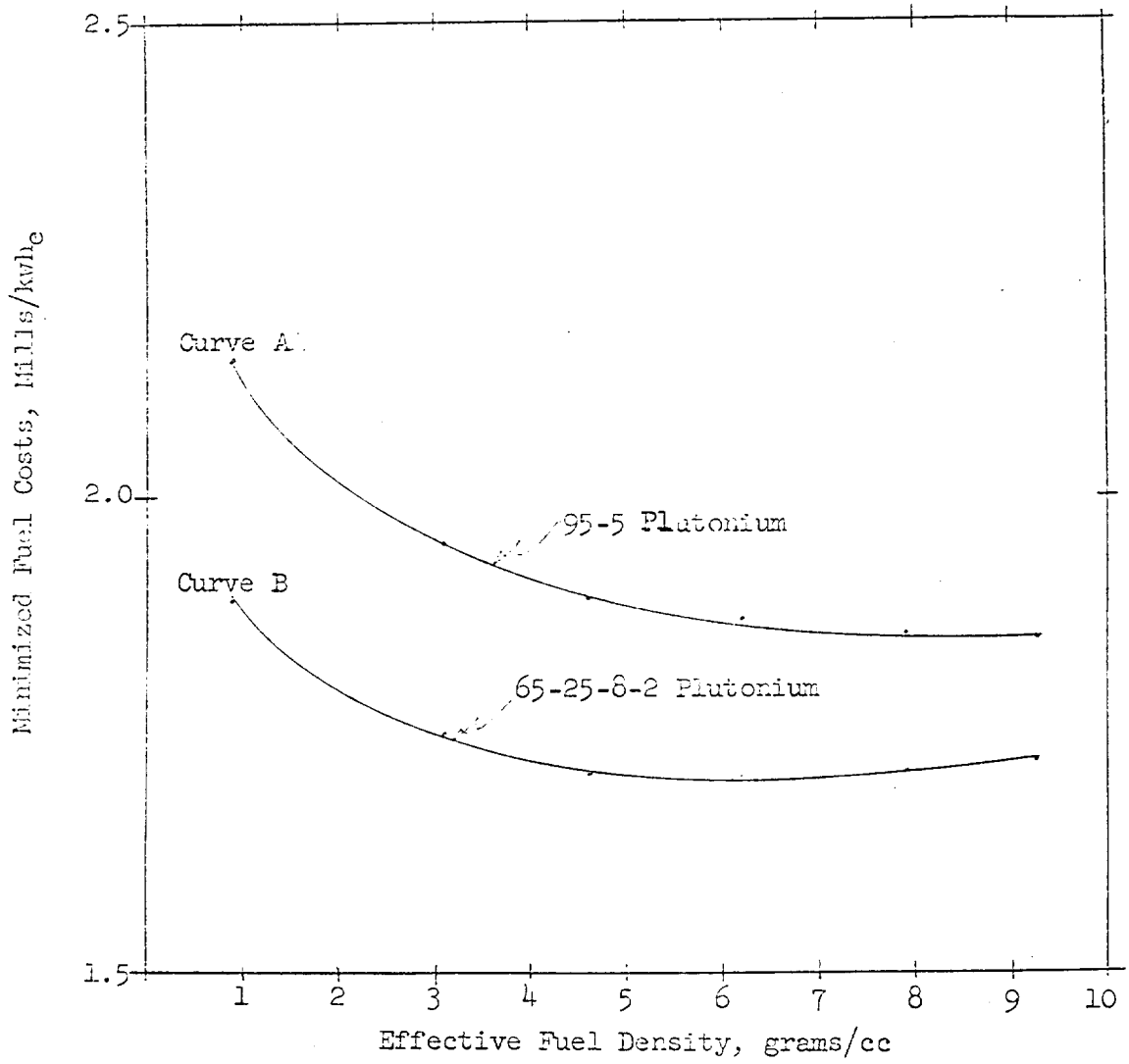


FIGURE 1

MINIMIZED FUEL COSTS AS A FUNCTION OF EFFECTIVE FUEL DENSITY FOR THORIUM FUELS ENRICHED WITH TWO PLUTONIUM COMPOSITIONS

AEC Interest Rate Parameterized

Plutonium Composition: 65% Pu-239 25% Pu-240
8% Pu-241 2% Pu-242

Water/Fuel Volume Ratio = 1.5
Small Rods, Increased Scattering, 165 w/cc
Specific Power

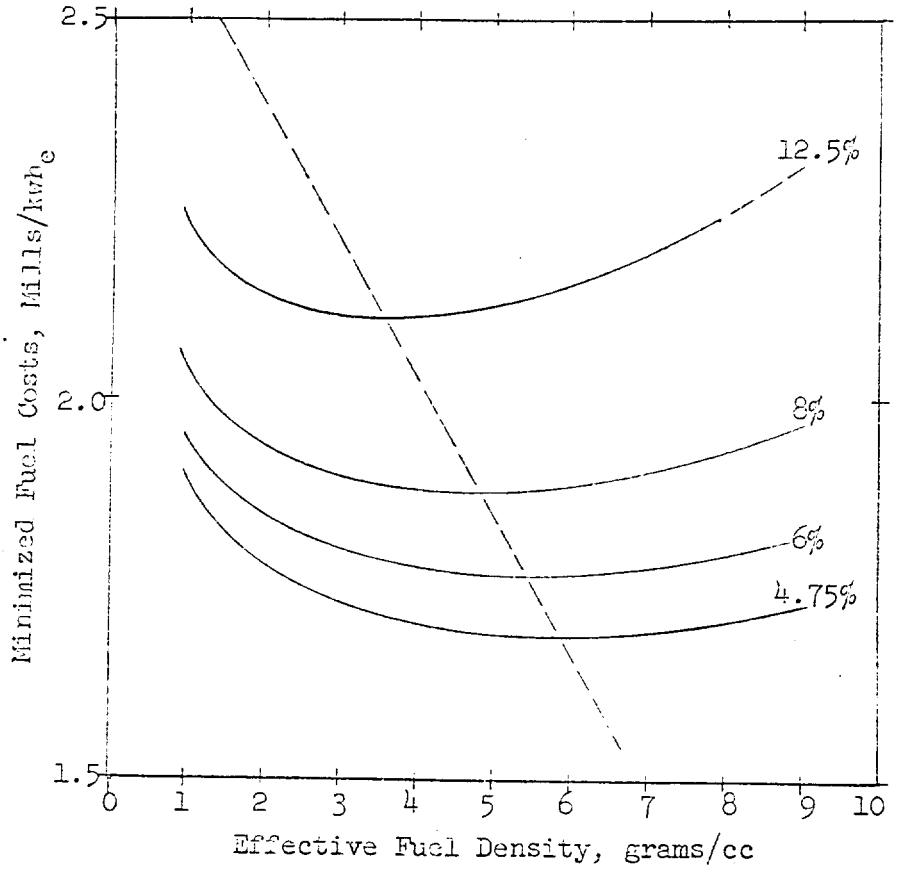


FIGURE 2

MINIMIZED FUEL COSTS AS A FUNCTION OF EFFECTIVE FUEL DENSITY FOR A PLUTONIUM ENRICHED THORIUM FUEL

these results because of the method used to account for competition for neutrons by the various isotopes. The difficulty occurs with the resonances near the thermal range, including the 0.3 ev Pu^{239} resonance. Consequently, the calculations are being repeated with less cross shielding among the overlapping resonances of the various isotopes involved.

Doubling Times and Breeding Ratios

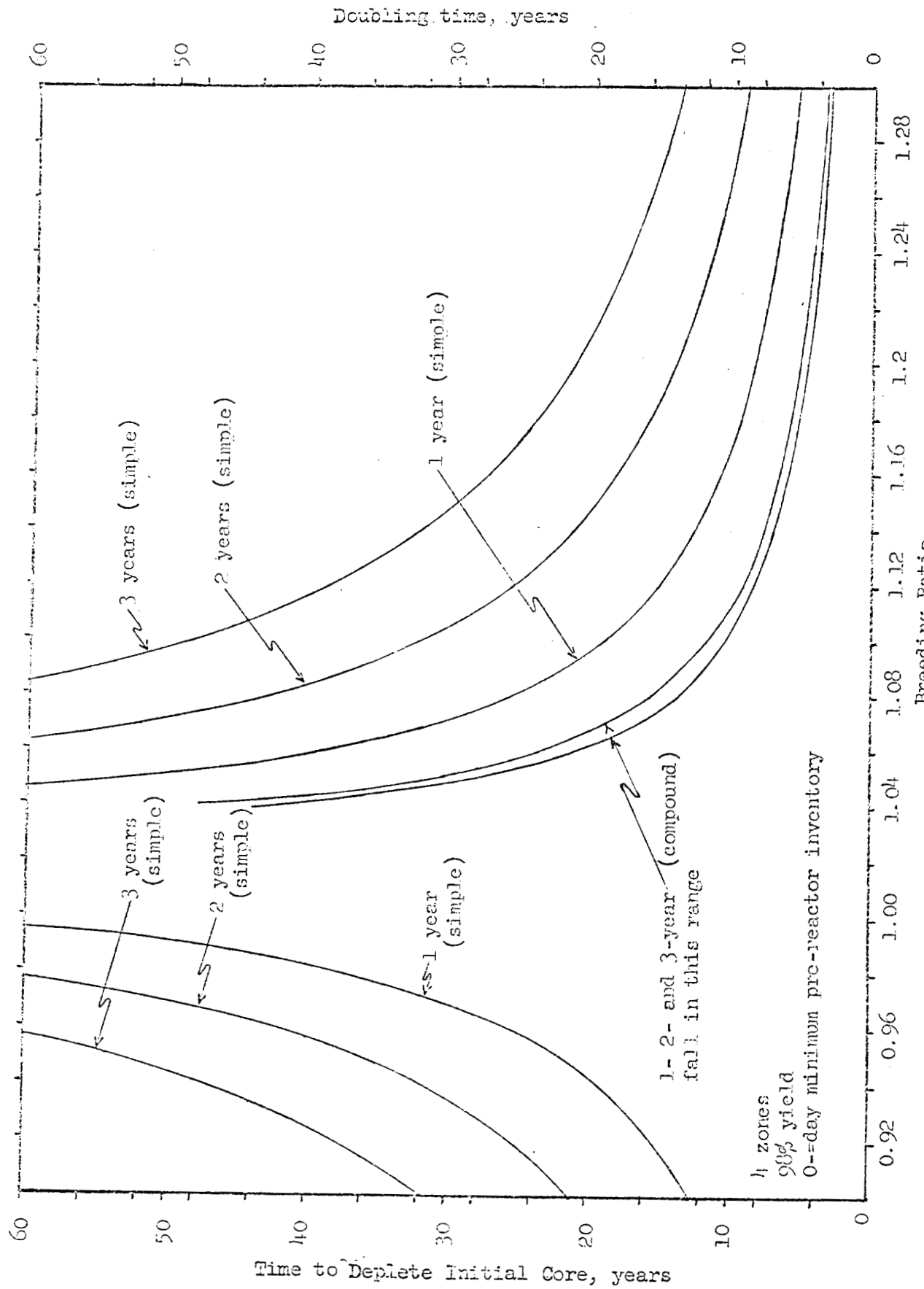
In connection with the conjectured "need to breed" by the year 2000-2100 (depending upon the optimism of the analyst), the question should be raised with respect to, "How much breeding is desirable?" Often relatively short doubling times are inferred as being necessary which tends to eliminate consideration of the use of thorium and U^{238} thermal breeders whose doubling times may be too long. To examine the credulity of this assumption, a study was conducted to investigate the fuel breeder characteristics of fuel-breeder and near-breeder complexes. For those complexes which would breed, both simple and compound doubling times (time necessary to generate enough fuel to make up 1.2 cores) were determined. For near-breeder reactors, doubling time is negative and becomes the time a near-breeder can operate before it consumes an amount of fuel equal to 1.2 cores. These calculations are made on the basis that a 90-day minimum turn around is possible which results in approximately 1.2 cores for most cases. However, an arbitrary selection decision is made by the computer which cannot acknowledge "shades of gray" rather than black or white and, as a consequence, the out-of-reactor inventory varies stepwise so that for a given net breeding ratio, the simple doubling time for a 2-year irradiation time is not precisely twice the simple doubling time for a 1-year irradiation time.

Figure 3 shows the doubling time versus breeding ratio for various exposure times. The simple doubling time considers that the fuel must be recycled in a single reactor; thus, the excess fuel bred in each cycle must be accumulated outside of the reactor until fuel for an entire new reactor is formed. The compound system assumes there are other reactors with the same breeding ratio being built which can use the bred fuel to breed still more fuel. These significant results are much like compound and simple interest in banking.

From these results and further analyses being made, it is suspected that the effective uranium ore reserves are really much greater than some analysts indicate; and that the near- or slight-breeder may provide adequate extension of the U^{238} reserves by use of plutonium recycle and in conjunction with thorium systems.

Plutonium Values As a Function of Composition

A study of possible methods of valuing plutonium in various reactor types and fueling schemes as a function of the isotopic plutonium compositions has been undertaken. The plutonium values and reactor



DOUBLING TIME ON SIMPLE AND COMPOUND BASES VERSUS BREEDING RATIOS FOR VARIOUS EXPOSURE TIMES

FIGURE 3

1235480

descriptions were taken from the study titled Plutonium and Uranium-233 Values Computed for Successive Recycle in Five Simulated Reactor Types, HW-76195. With the assistance of Applied Mathematics, these values were submitted to analysis in the GEORGE code to determine the reactor parameters, or other variables, on which these values might be dependent. Initially, each plutonium value was fitted against the respective compositions of the isotopes for each reactor. The coefficients, as calculated by the GEORGE code, were averaged for a given fueling scheme. Initial calculations indicate that these equations will predict plutonium values in terms of the isotopic plutonium composition for a given fueling scheme (that is, batch or graded) within $0.75 per gram, or less.

The major reactor physics parameters were examined to see if improvements could be made on predicted values. The Sigma Nonfuel term (SNF), as utilized by the MELEAGER code, shows some promise of indicating plutonium values as a function of reactor parameters, but results at this point are inconclusive. The SNF term is indicative of the parasitic absorbers in the reactor; for example, SNF is greater for zirconium jacketed fuel that is H_2O cooled and moderated than for zirconium jacketed fuel that is D_2O cooled and moderated.

Yankee Comparison

An attempt to simulate the Yankee Pressurized Water Reactor was made using the JASON-MELEAGER CHAIN; the plutonium composition as a function of U^{235} depletion was plotted and compared with experimental and theoretical results appearing in WCAP-6056. The MELEAGER results compared favorably with the experimental curves and, in some instances, matched the experimental curves more closely than those calculated in WCAP-6056. Thus, it appears that JASON-MELEAGER adequately describes a given reactor system of this type, if sufficient input data are available as in this case; namely, WCAP-6056 and the data sheets in the March 1961 issue of Nucleonics.

Technical Interchange Activities

Three fuel cycle analysis papers were presented by Programming people at the American Nuclear Society Winter Meeting in New York City, November 18 through 21, 1963.

The paper titled Exploiting Plutonium-240 by Reducing Uranium-238 Concentrations discussed possible advantages of reducing the U^{238} concentration of fuel when plutonium is used for enrichment in a reactor lattice optimized for U^{235}-U^{238} operation.

A complementary paper was presented by Applied Physics reporting work done in cooperation with Programming. This paper was titled Some Economic Consequences of Plutonium Interchange in Thermal Reactors As Affected By Isotope Composition, Moderator Type, Fueling Mode, and Clad Material.

The paper described considerations of altering the lattice spacing while maintaining the U^{238} concentration constant. This latter approach can be used when new reactors are being designed or when considerable renovation is possible to an existing reactor.

The second paper presented by Programming was titled Calculation of Production and Pricing of Transuranium Isotopic Heat Sources. This paper discussed results of an analysis of Cm^{244} and Pu^{238} production in central station power plants. Included was an analysis of the alteration to plutonium values if Cm^{244} would indeed command a high market price.

The third paper presented by Programming personnel titled Some Observations on Fissile Conservation in Thermal Reactors discussed the role that plutonium recycle and proper use of the diffusion cascades can make in view of an expanded nuclear economy with limited uranium reserves. The discussion showed that, with the diffusion plant operated with optimum tails composition, the utilization of uranium automatically goes up as the price of natural uranium is increased. Thus, operation of the diffusion cascade at minimum costs for which the tails composition goes down as uranium ore prices are increased tends to ensure conservation of U^{235}; and analyses which ignore this response of the cascade are misleading in this respect.

The paper also included comparisons of the conservation role of plutonium recycle with unimproved, and improved, neutron economy thermal reactors. With unimproved reactors, plutonium recycle may appear to extend the utilization of U^{235} only 70 to 100 percent; but, with improved thermal reactor types, plutonium recycle improves the utilization by 200 percent and by 300 to 400 percent if the price of natural uranium is doubled. This paper is the first of a planned program to fully evaluate the role of plutonium recycle and uranium conservation, as indicated elsewhere in this monthly report.

All four papers constitute synopses of formal Hanford documents now in preparation.

Nuclear Safety Activities

Work is continuing on more reliable methods for estimating radiation exposure as a function of distance, for cases such as a reactor accident which releases fission products to the environs. A current Hanford analysis, employing local wind variability data, shows that changes in wind speed and direction during the period of release of fission products tend to reduce the estimated distance at which a given exposure would be expected, as compared to previous conservative assumptions of constant wind speed and direction. The analysis of wind variability information provided by Atmospheric Physics was the basis for the calculations. Indications are that local variability in wind speed and direction may overshadow the importance of turbulent diffusion in the lateral dilution of a plume of contaminants. This new approach to reactor accident evaluation could have considerable impact on reactor siting questions.

The AEC approved the fuel element design limits set forth in PRTR Final Safeguards Analysis Supplement 6 - Revised Limits. All other provisions of Supplement 6 had been approved previously by the AEC.

J. W. Woodfield

FW Woodfield:pc

Manager
Programming

RADIATION PROTECTION OPERATION REPORT FOR THE MONTH OF NOVEMBER 1963

A. ORGANIZATION AND PERSONNEL

Frank H. Sanders terminated this month to accept a position with Reynolds Electric and Engineering Company. Fran S. Glick transferred to NRD Department as of 11/13/63. Myrtle O. Wendland has been removed from the active roll because of illness. Emily H. Szymanski transferred from IDO to CDS&RO on 11/11/63. W. V. Baumgartner joined RD&C transferring from EDO. Grace P. Gibson transferred from CDS&RO to EDO on 11/18/63. G. T. Hewitt joined Radiation Monitoring on 11/18/63.

B. ACTIVITIES

Occupational Exposure Experience

There were no new plutonium deposition cases confirmed during the month. A re-evaluation of a deposition case previously estimated to be less than 10 percent of the MPBB resulted in its removal from the list of confirmed deposition cases. The total number of individuals who have internal plutonium deposition at Hanford is 324 of which 234 are currently employed.

During the month there were seven incidents involving 23 employees which required special bioassay sampling for plutonium analyses. The following is a brief description of these incidents.

A fire occurred at the Weapons Manufacturing Building (234-5) on November 4, 1963 in the leaching hood located in Room 227. A fireman pressurized the hood when he sprayed dry chemicals into it. Contamination on the clothing of the two operators working in the hood was in excess of 40,000 d/m and contamination on the skin ranged from 2,000 to 20,000 d/m. Nasal smears from the two operators were negative.

A fire occurred in the Redox Final Concentration Building (233-S) on November 6, 1963. Ten firemen involved in fighting the fire received skin contamination ranging from a few thousand d/m to greater than 40,000 d/m after showering. One individual who was surveyed prior to his shower had contamination in levels measurable with a Juno. Positive nasal

contamination was found only on two individuals. Three operators who were working in the 233-S Building when the fire occurred left immediately.

Evaluation of gold foils from a Hanford criticality dosimeter located approximately 20 feet from the source of the fire on the other side of the concrete wall indicated no criticality had occurred in conjunction with the fire. Surveys showed that plutonium contamination had spread throughout a major portion of the building and ranged from a few hundred d/m to several hundred thousand d/m. Some contamination was found on the roof of the building around three ventilation ducts and the flashing around the edge of the roof where smoke had been vented. The contamination levels have been reduced in most of the building with the exception of the stairwell and process area #2.

The spot analysis results of urine samples received from the individuals involved in the fire indicate that in most cases a slight amount of plutonium is present in the urine. Based on these early results, which are inconclusive, it appears that the amount of internal deposition of plutonium will be small in each case.

A CPD millwright received nasal contamination ranging from 100 d/m to 2,000 d/m at the Weapons Manufacturing Building (234-5) on November 13, 1963 while changing the 19-B airlock door between the conveyor and hood. Upon completion of the work, a survey revealed skin and clothing contamination up to 20,000 d/m. After the respiratory protection equipment was removed contamination was found on his face and nose which apparently had been transferred from his hands. A survey of the work area revealed the presence of contamination on the horizontal surfaces of an adjacent hood from which the person's contamination evidently came.

Three CPD technologists received nasal contamination ranging from 10 d/m to 300 d/m while performing work in the emission spectrometer hood in the Weapons Manufacturing Building (234-5) on November 19, 1963. A plutonium oxide pellet caused some cotton gauze to catch on fire. Before the fire was put out, a hood glove was burned which caused the contamination to become airborne.

A CPD operator received a spot of plutonium nitrate contamination on his forehead reading 20,000 d/m at the Purex Building (202-A) on November 19, 1963. The employee was using a Wand to transfer

plutonium nitrate solution from a PR can. Normally when a screen, which is located on the end of the wand, becomes clogged with solid materials the side of the PR can is struck with the Wand to dislodge the materials. The employee looked into the PR can while striking it with the Wand and some liquid was splashed onto his forehead. The contamination was initially reduced to 3,000 d/m and the employee was released. He was examined by Industrial Medical the next day and again released without further decontamination since the skin was sensitive from the acid burn. The contamination was further reduced to 1,000 d/m on November 26.

There were no plutonium contaminated injuries during the month, and the total number for the year remains at 20 with 14 of these requiring excision.

Several other occurrences of significance which occurred during the month are summarized below:

Special evaluation of the film badge dosimeters worn by two IPD maintenance men on November 18, 1963 showed that each employee received a dose of 1.45 rems. The established dose rates for the work these employees were performing on the Ball 3X system in the ball room of the 105-C Reactor Building were up to 150 mr/hr. Upon completion of the work their pocket dosimeters read off-scale. Since a resurvey of the work area indicated that the employees would have received as much as 300 mrem during their work, the source of the dose described remains unknown.

A review of the employees' total exposure for the year to date including this incident indicates one has 3.2 rems, and the other has 4.1 rems. One employee has a total of 2.4 rems for the last 13 weeks. This incident is to be documented as Hanford Class B. Incident #191.

False critical radiation alarms occurred at the 105-C Reactor Building on four different occasions during the month. These alarms resulted from equipment malfunction. The equipment was removed from service for repairs.

A model "D" manipulator was removed from "A" Cell of the Radiometallurgy Building (327) on November 18, 1963 for repair. The manipulator was removed directly into the air since there are no provisions for decontaminating and repairing the model "D" in any of the cells. During this operation, contamination

in levels up to 60,000 c/m was spread around and on the cell. In addition, four personnel involved in the work received clothing contamination up to 80,000 c/m; however, skin contamination did not occur.

A false critical radiation alarm occurred in the storage area of the 105-DR reactor building on November 26, 1963. The alarm occurred when water from a spray hose hit the detector causing it to short out. The four personnel in the area immediately evacuated. Re-entry was performed according to standard procedures.

A PRTR primary system heavy water level indicator was installed and tested during the month. Dose rates to the hands and body during testing of the equipment were 30 R/hour and 150 mrem/hour, respectively. Personnel exposures were controlled to 500 mrem to the hands and 100 mrem to the body.

A number of leaks in heat exchanger HX-5 resulted in a small release of heavy water to the liquid effluent system. The leaks resulted in an unscheduled shutdown of the PRTR and extensive inspection and maintenance were required to detect and repair the leaks. Personnel were exposed to dose rates from external radiation varying from 30 to 300 mrem/hour, and also came in contact with heavy water containing low concentrations of tritium. The maximum exposure estimates to personnel are on the order of 600 to 700 mrem including exposures from all sources.

Environmental Experience

Concentrations of fallout materials in the air of the Pacific Northwest have slowly decreased since August, 1963. The average value for November, 1 pc β/m^3, was the lowest monthly average observed since the USSR resumed testing in the fall of 1961.

About 0.2 curie of I^{131} was emitted from the Purex stack on November 6, 1963, during dissolution of the remainder of the metal in C-cell dissolver. This metal had been held for decay of the I^{131} content since September 2, 1963, when it was inadvertently charged after only eighteen days cooling. The fraction of the metal actually dissolved on September 2, 1963, had also been held for I^{131} decay (as a solution in D-cell). Blending of this later solution with current process solutions had begun on November 5, 1963, with no noticeable increase in I^{131} emission from the Purex stack.

Ground surveys performed in the vicinity of 200 West Area following the fire in the Redox Final Concentration Facility (233-S) on November 6, 1963, revealed no detectable contamination (<500 d/m on portable poppy survey meters). The air filter samples operated at the three atmospheric monitoring stations (614 Buildings) in 200 West Area indicated increased concentrations of both alpha and beta emitting radionuclides. The highest of these was at the Redox station and amounted to about three times the upper limit of usual results. Nearly all other air filter samples changed on November 6, 1963, contained normal amounts of radioactive materials.

Approximately 35 millicuries of I^{131} and I^{132} were emitted from the High Level Radiochemical Facility (325-B) stack during day shift on Friday, November 15, 1963. Ground surveys performed around the 325 Building revealed no contamination attributable to this emission. The work responsible for the stack emission was the ignition at 200° C of an irradiated PuO fuel sample from PRTR containing a total of ∿2 curies of I^{131}. A saturated BaOH scrubber had been installed in the off-gas line within the cell and apparently had prevented all but ∿5% of the inventory from escaping to the ventilation duct and filters and stack. Radioiodine emission from this stack continued over the weekend. A total of 64 millicuries was emitted between 1730, November 15, and 0930, November 18, 1963.

The ventilation filters were changed on November 18 and 19, 1963, in an attempt to reduce the stack emission. Samples of these filters submitted to the laboratory for gamma spectrometric analysis indicated that the filters had trapped a total of 20 mc of I^{131}. Environmental Monitoring revised its stack sampling facilities on November 18, 1963, to include a caustic scrubber radioiodine sampler. Charcoal filters will be installed in the 325-B Building ventilation system before any more work of this nature is undertaken.

A cooling coil leak discovered November 7, 1963, in the "rare earth fraction" storage tank (6-1) in B-Plant allowed ∿500 curies of Ce^{144} and ∿100 curies of strontium, primarily Sr^{90}, to escape to the B-Plant retention basin. To stabilize the contamination, the ditch leading from the basin to the B swamp was diked off and the contaminated liquid discharged to the ditch. While the ditch was open, a high wind blew tumbleweeds through the contaminated liquid and against the 200-E fence northeast of the retention basin. About one mile of the fence line was contaminated. The weeds, contaminated up to 100 rads/hr at surface, were immediately picked up by CPD personnel. Low level contamination,

up to 50,000 c/m, was located outside the fence line during ground surveys performed by EMO personnel. Efforts to stabilize the contamination were in progress at month's end. The 6-1 storage tank has been replaced in the process.

One aerial monitoring flight was made. The route taken was the Columbia River from Richland to Longview. Poor flying conditions prevented the flight from continuing to the ocean. No unusual readings were noted.

A total of 367 biological, produce, and food samples were obtained for radiochemical analyses. They were:

Milk	168 gallons	60 samples
Pasture grass		20 samples
Beef thyroids		25 samples
Hay		5 samples
Ground Round	6 pounds	3 samples
Oysters	2 pounds	1 sample
Fish		80 samples
Pheasant		1 sample
Duck Muscle		57 samples
Duck Heads		113 samples
Goose Muscle		1 sample
Goose Head		1 sample

Milk and pasture grass sampling schedules were reduced this month because of the removal of most dairy herds from fresh pasture.

Studies and Improvements

The preliminary sketch of the proposed Fast Reactor Critical Facility was reviewed for compliance to established radiological design criteria. Although several earlier recommendations on radiological design improvements were incorporated into this drawing, the following additional comments were offered: (1) A change in site location from one near the Low Level Chemistry Building to a location in the general area south of PRTR, (2) establishing an outside entrance to the shop to relieve traffic from traversing a radiation zone, (3) relocation of the counting room to a location having a lower background, and (4) installation of a door between the RMO office and change room which would expedite communication and service.

The preliminary design of the proposed Radio-Surgery Facility was reviewed for comment on the radiation protection aspects of the building and facility layout. Design criteria resulting from this review included the need for establishing two air locks, changing

two shielded swinging doors to one-way doors for purposes of regulating traffic patterns from contaminated areas, nomenclature changes on rooms, and removing several deficiencies in the present shielding design.

The safety interlock system for the Flash X-ray was installed and testing was carried out to insure safe operation. The unit cannot now be charged to operating voltage or triggered with the door to the exposure room open. Similarly, the door to the exposure room cannot be opened without discharging the X-ray if it has been left in a charged condition. Flashing red lights operate inside and around the building whenever the Flash X-ray is in a chargeable attitude.

Several exposures to both neutrons and gamma radiation fields were completed with LiF phosphor obtained from Controls for Radiation (ConRad).

The results of the neutron exposures showed that the minimum detectable dose is about 1 rad for ConRad phosphor under the present operating conditions. The minimum detectable dose is also about one rad for Harshaw LiF phosphor. A comparison of the gamma sensitivity of the Harshaw and ConRad phosphors shows that on the average Harshaw's phosphor is 1.9 times as sensitive to radium gamma radiation. It was also found that the presence of alcohol in the dosimeter vial reduced the gamma sensitivity by about 8% for ConRad phosphor and about 12% for the Harshaw phosphor. The alcohol is added to the dosimeter vials to provide fast neutron sensitivity.

Some modifications of the electronic reading device were recently developed by ConRad to increase the sensitivity of the reader. ConRad claims the sensitivity can be increased by a factor of ten which would yield a minimum detectable dose of about 10 mr of gamma radiation.

All of the plutonium particle size distributions reported previously were based on the assumption that the aerosol sampled contained only Pu-239. While in the production of Pu-239, small amounts of other plutonium isotopes with a shorter radiological half life and higher specific activity are produced. It appears that other possible errors in the filming of the particle, location of the particle within the filter paper and actual density of the particle are probably as substantial as the error made in assuming 100% Pu-239.

A total of 46 Audio Annunciators were received from the vendor. Assembly of the remaining units will be completed upon arrival of potentiometers.

A prototype circuit board for the new Scintran mixer-gate circuit and negative was received from Nucleonics Instrumentation. The

board represents an attractive improvement to the Model II Scintran performance; it essentially eliminates all objectionable background hum, and increases the volume of each audio signal. This circuit will be incorporated in the next purchase of Scintrans. These circuit boards can be added to all on-plant Scintrans by simply interchanging with one of the current circuit boards.

A new film badge dosimeter station was established in the 105 Building in 100-N Area. Additional changes in the film badge dosimeter stations in 100-N will have to be made after the perimeter fence and badge house have been established.

Preparations were completed to accommodate on-plant plutonium bioassay sampling in the 231-Z Building beginning in December.

A brochure describing both the iron room of the Hanford Whole Body Counter and the Mobile Whole Body Counter was published.

Toxicity testing by Aquatic Biology of two more chemicals for use at the N reactor was arranged. The 109-N portion of the primary coolant loop was laid up with a hydrazine solution after cleaning. Discharge of this solution to the river is planned before operation. A new chemical cleaner, Vertan 675, is expected to be used for supplemental cleaning of the 105-N portion of the primary loop.

The results of a comparative sample of reactor effluent water submitted to three laboratories were reported. Agreement between the laboratories was much better than in the past. Phosphorus-32 and iodine-131 results reported by Purex Laboratory were slightly lower than either the Radiological Chemistry Laboratory or the Radiological Chemical Analysis Laboratory. There was no significant difference in the zinc-65, chromium-51, arsenic-76, and neptunium-239 determinations.

A revision to the Columbia River Emergency Plan was prepared which better defines action levels and preferable lines of notification.

A ten-foot dry well was dug on the east end of the 108-FC Building for the purpose of monitoring effluent gas at different underground depths during 105-F operations, and also during shutdown periods. This has been a daily routine for at least two weeks. It now appears certain that the effluent-caused background adjacent to the 108-F Biology Building will not be corrected until all of the joints on the underground effluent pipe have been repaired. A start was made on this major repair operation during the last reactor shutdown.

Research Studies

Effect of Reactor Effluent on the Quality of Columbia River Water (02)

A study of the effects of reactor effluent on a river water quality, with emphasis on temperature, continued. Dye studies continued with spot releases from the DR and K reactor outfall lines for measurement of time of travel, rate of dispersion, and temperature changes in the tagged water masses.

Assistance of the Applied Mathematics Section was obtained for the determination of correlation coefficients between pertinent meteorological parameters and observed temperature changes through the Hanford section of the Columbia River.

Mechanisms of Environmental Exposure

Collection of environmental sampling data has been initiated preparatory to correlation studies for P^{32}, Zn^{65}, K^{40}, and Cs^{137}. The objective of the correlation study is to determine the relationships between P^{32} and some gamma emitting isotope at each stage of the food chain.

One of the gamma emitting isotopes commonly measured in environmental samples is K^{40}. It is not measured in our water samples, however, and steps have been taken to determine the feasibility of reporting K^{40} for water samples collected. The primary purpose for this is that K^{40} makes a good relative index. Anomalies in the K^{40} content often indicate discrepancies in other measurements. As a matter of interest, the annual dose, total body, due to K^{40} deposition was calculated to be 12 mrems/year per 100 grams of potassium. For a person of standard man proportions, this amounts to 17 mrems/year.

C. RELATIONS

Safety meetings were held throughout the Section during the month. There were no security violations.

There were no suggestions submitted for evaluation, and no suggestion awards made during the month.

D. SIGNIFICANT REPORTS

HW-76525 10 - "Radiological Status of the Hanford Environs for October, 1963", by R. F. Foster.

HW-79605 - "Radioactive Liquid Waste Disposal" by R. H. Wilson.

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

	November	1963
External Exposure Above Permissible Limits		
Whole Body Penetrating	2	3
Whole Body Skin	0	0
Extremity	0	1
Hanford Pocket Dosimeters		
Dosimeters Processed	10,720	74,559
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	20,417	116,911
Results - 100-300 mrads	358	2,064
Results - 300-500 mrads	43	199
Results - Over 500 mrads	6	43
Lost Results	20	255
Average Dose Per Film Packet - mrad (ow)	7.2	7.6
- mr (s)	24.6	30.3
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	3,516	20,481
Results - 50-100 mrem	0	18
Results - 100-300 mrem	0	2
Results - Over 300 mrem	0	0
Lost Results	7	109
Fast Neutron		
Film Read	1,023	5,620
Results - 50-100 mrem	14	282
Results - 100-300 mrem	142	629
Results - Over 300 mrem	2	8
Lost Results	7	88
Hand Checks		
Checks Taken - Alpha	27,151	405,910
- Beta-Gamma	45,488	622,432
Skin Contamination		
Plutonium	26	274
Fission Products	38	420
Uranium	0	5
Tritium	0	0
Thorium	0	1

Whole Body Counter

Subject	Number of Examinations 747-A WBC	1963	Mobile WBC	1963
GE Employees				
Regular	32	632	175	497
Incident Cases	1	145		1
Terminations	5	129		1
New Hires	31	522		1
Special Studies	4	480		
Non-Employees				
Children	3	24		
Visitors	1	45		
Environmental Studies	1	20	—	—
	78	1,997	175	500

Bioassay

Analysis	Current Reporting Limit	Results Above Reporting Limit Nov.	1963	Samples Assayed Nov.	1963
Plutonium	2.2×10^{-8} µc/sample	37	893	493	6,388
Fission Prod.	3.1×10^{-5} µc/sample	9	107	419	5,741
Strontium	3.1×10^{-5} µc/sample	-	45	-	45
Tritium	5.0 µc/l	223	1,538	337	2,612
Uranium	0.14 ugm/l	-	-	419	1,688
Special Studies		-	-	46	404

Calibrations

	Number of Units Calibrated November	1963
Portable Instruments		
CP Meter	1,118	11,760
Juno	263	2,847
GM	565	6,303
Other	225	2,131
Audits	103	1,174
	2,274	24,215
Personnel Meters		
Badge Film	1,140	9,152
Pencils	105	1,170
Other	312	3,350
	1,557	13,672

	Number of Units Calibrated	
	November	1963
Miscellaneous Special Services	126	14,375
Total Number of Calibrations	3,957	52,262

A R Keene
Manager
RADIATION PROTECTION

AR Keene:ICN:ald

FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

During the month, the Hanford Laboratories' operating cost control budget was adjusted to incorporate changes in program planning as follows:

	Increase (Decrease)
04 Program	
Reactor Fuels and Materials	$ 200 000
Plutonium Ceramics Research	(35 000)
Reactor Components	26 000
Containment Systems Experiment	50 000
02 Program	
IPD Sponsored Reactor Program	(170 000)
NRD Sponsored Reactor Program	60 000
NRD Sponsored Metallurgy Program	(23 000)
Other Services	
Fabrication of Thorium Oxide Fuel for IPD	70 000

RLOO-AEC transmitted to Hanford Laboratories an authorization of $20,000 from the Air Force Cambridge Research Laboratories for continuation of the Atmospheric Diffusion Studies for the period April 1, 1964 through March 31, 1965.

During November, Hanford Laboratories was allocated an additional $30,000 for capital work order costs, bringing the authorization for FY 1964 to $160,000. Hanford Laboratories was not allocated any additional equipment obligational authority during the month.

Priority requests were transmitted to RLOO-AEC for additional equipment obligational authority totaling $117,200 for the 04 Program. Further delay in ordering several items of necessary equipment will hinder or delay certain aspects of the research and development work.

A midyear review of irradiation unit requirements for FY 1964 was prepared and submitted at the request of RLOO-AEC. A summary by program follows:

	FY 1964
02 Program	
Metallurgy Program	$ 332 000
Reactor Program	693 000
04 Program	
Gas Cooled Reactor Program	856 000
Reactor Fuels and Materials	2 652 000
Plutonium Recycle Program	138 000
Plutonium Recycle Reactors - Other	272 000
Total	$4 943 000

A special accounting code was established for the activity described below:

.8P Consultation with Phillips Petroleum by M. R. Egan on the development of Monte Carlo computer program techniques. Billing will be for travel and subsistence.

The following organization and program code changes were made during the month:

Codes Established

Organization Codes

7410 Critical Mass Operation
7420 Experimental Physics Research Operation
7464 Experimental Reactor Operation
7465 Reactor Lattice Physics Operation

Program Codes

.13 Containment Systems Experiment R & D (04 Program)
.14 Containment Systems Experiment Facility (04 Program)
.52 IPD - U-233 R & D (02 Program)
.63 CPD - 1807 Process Technology (03 Program)

Codes Canceled

Organization Codes

7420 Nuclear Physics Research - General
7422 Experimental Nuclear Physics Operation
7423 Reactor Lattice Physics Operation
7424 Experimental Reactors Operation
7425 Critical Mass Physics Operation

General Accounting

Following is the status of letters requesting AEC concurrence in proposed actions:

AT-313	Professional Research and Teaching Leave Assignment	Approved 11-8-63
AT-316	AEC Monograph on Iodine 131 - L. K. Bustad and J. K. Soldat	In process

The following new and revised OPGs were issued during November 1963:

OPG No.	New	Revised	Title
2.3.7		x	Contract and Accounting Operation Manager Position Guide
11.2		x	HL Numerical Index
11.2 App.	x		HL Subject Index
3.1.4 & App.		x	Absence Payment Plans
3.1.8		x	Suggestion Plan
3.1.3 p. 2		x	GE Educational Assistance Program
8.5		x	Property Management of Protective Clothing and Equipment
8.10		x	Control of Research and Development (RDX) Equipment
8.15 p. 3		x	Property of Others
8.17		x	Control of Liquor, Grain Alcohol, Narcotics
8.18		x	Assistance to Hanford - Property Controls
8.7		x	Control of Materials and Supplies
8.12		x	Control of Laid-up Facilities

Hanford Laboratories' net material investment at November 1, 1963 totaled $23.2 million as detailed below:

		(In thousands)
SS Material		$21 659
Reactor and Other Special Material		1 251
Spare Parts		337
Yttrium		26
Subtotal		23 273
Reserve: Spare Parts	$79	
Yttrium	26	(105)
Net Inventory Investment		$23 168

The cumulative value of nuclear material consumed in research by Hanford Laboratories during FY 1964 (at November 1, 1963) is $246,580 comprised as follows:

02 Program	$ 16 877
03 Program	139 972
04 Program	89 731
	$246 580

A report of the audit of blank One Trip Property Passes in the possession of 57 authorized holders reveals four missing passes out of a total of 819 passes issued for the year compared to two lost passes out of a total of 2,197 issued the preceding year.

Reconciliation of the annual HAPO physical inventory of reactor and other special materials taken in September was completed and a report issued by C&AO on November 19, 1963. The inventory disclosed a deficit of $138.09 for Hanford Laboratories due to failure of holders to indicate documentation for materials consumed in experiments and/or tests. The deficit is relatively minor when compared with a total HL material investment in excess of one million dollars at time of inventory. All shortages were reported to and investigated by HAPO Security.

Unitization reports were completed and issued during the month on the following projects:

CAH-822	Pressurized Gas Cooled Loop Facility, 309 Building	$1 163 500
CAH-936	Coolant Systems Development Laboratory 1706-KE Building	129 071
		$1 292 571

The heavy water inventory at the end of November 1963 showed an estimated loss of 1,036 pounds ($14,276) for the PRTR and a loss of 37 pounds ($510) for the PRCF. Heavy water scrap generated during the month amounted to an estimated 6,015 pounds, resulting in a $10,045 charge to operating cost. A billing was submitted to SROO for the November 4 shipment of heavy water consisting of 19,091.23 pounds with a fund value of $240,940.36 and a non-fund value of $116,124.67. A requisition was placed November 19, 1963 for 27,500 pounds (55 drums) of heavy water from SROO with a delivery date of January 15, 1964. The inventory fund value of this material will be approximately $374,000.

Laboratory Storage Pool activity is summarized as follows:

	Current Month		FY to Date	
	Quantity	Value	Quantity	Value
Beginning Balance	1 792	$813 306	1 480	$ 811 520
Items Received	57	146 930	948	388 346
Items Reclaimed by Custodians	(9)	(4 774)	(131)	(46 115)
Equipment Transfers	(16)	(3 577)	(159)	(52 797)
Items Disposed by PDR			(140)	(11 290)
Items Disposed by Excess	(77)	(1 605)	(251)	(139 384)
	1 747	$950 280	1 747	$ 950 280-1)

(1- Includes 166 items valued at $117,387 on loan at November 30, 1963.

During the month, 98 items valued at $33,488 were loaned and/or transferred in lieu of purchases. A total of 390 items valued at $143,640 has been redirected to useful purposes this fiscal year. Operating costs for October were $1,944 and for FY 1964, $5,946.

Total value of equipment and material in custody of the Laboratory Storage Pool at November 30, 1963, was $1.9 million, including Reactor and Other Special Materials of $291,912, SS Materials of $154,800, and other materials valued at $466,097.

During the month, aluminum (tubes, rods, and plate), valued at $27,035, was transferred to the Pool from the basement of the 326 Building; zirconium (R & D) tubes, valued at $13,420, were transferred for storage from the 314 Building; 2,879 pounds of zirconium scrap, valued at $5,758, were excessed; and UO_2, valued at $9,000, was issued for use by Fuels Testing and Analysis and Fabrication Development.

Action on projects during the month:

Physical Completion Notice Issued

CAH-995 309 Building Air Conditioning Modifications

The following contracts were processed during the month:

CAH-417	Henry Eyring	SA-314	Scientia Laboratory
CA-421	Kermit L. Garlid	SA-315	W. E. Welch
CA-422	Robert W. Albrecht	SA-316	W. H. Harris
SA-283	Pulp and Paper Research Institute of Canada	SA-319	B. C. Christopher
		SA-320	P. M. Aldrich

Personnel Accounting

All exempt employees except two have completed Acknowledgment of Directive Policy - General No. 20.5, dated January 4, 1963. The two employees who have not signed are in Idaho Falls and will be personally contacted by their supervisor.

J. A. Allen retired effective December 1, 1963.

Rate of record of all nonexempt non-unit employees was checked with rate of record in the Personnel Source File and no discrepancies were found.

Personnel statistics follow:

Employee Changes	Total	Exempt	Nonexempt
Employees at beginning of month	1 786	788	998
Additions and transfers in	24	6	18
Removals and transfers out	14	7	7
Employees on payroll at end of month	1 796	787	1 009

Overtime Payments During Month	November	October
Exempt	$ 7 646	$ 6 239
Nonexempt	31 581	26 627
Total	$ 39 227	$ 32 866

Gross Payroll Paid During Month		
Exempt	$ 766 698	$ 771 974
Nonexempt	575 013	558 299
Total	$1 341 711	$1 330 273

Participation in Employee Benefit Plans at Month End	November		October	
	Number	Percent	Number	Percent
Pension	1 570	99.4	1 565	99.4
Insurance Plan - Personal	410		418	
- Dependent	1 371	99.9	1 357	99.9
U.S. Savings Bonds				
Stock Bonus Plan	157	39.5	155	39.3
Savings Plan	67	3.7	67	3.7
Savings and Security Plan	1 236	88.2	1 237	88.7
Good Neighbor Fund	1 284	71.4	1 283	71.8

Insurance Claims	November		October	
	Number	Amount	Number	Amount
Employee Benefits				
Life Insurance	1	$13 530	-0-	$ -0-
Weekly Sickness and Accident	7	429	9	924
Comprehensive Medical	51	3 102	57	3 521
Dependent Benefits				
Comprehensive Medical	114	10 392	101	10 057
Total	173	$27 453	167	$14 502

TECHNICAL ADMINISTRATION

Employee Relations

Twenty nonexempt employees were hired; 20 open requisitions remain.

Suggestion plan activity included 46 submissions, 11 adoptions and 20 rejections.

Information and Presentations

Visitors Center activity:

November attendance	1 071
Average attendance each day open	43
Cumulative attendance since 6-13-62	59 184
Conducted groups	7 (totaling 101 people)

Plant tour activity:

	Number	Total People
General Public Relations Tours	7	111
Special tours	3	5
Cumulative attendance since 1-1-63 (all tours)	-	2 689

Documented information flow during the month was comprised of 1,173 titles (11,500 copies) received at Hanford and 56 titles (6,100 copies) sent off-site.

The lease with the City of Richland for Visitors Center space was secured during November and will run until October 31, 1964.

Professional Placement

Advanced Degree - Two Ph.D. applicants visited HAPO for employment interviews. Three offers were extended; no rejections were received. Two offers are currently open.

BS/MS (Direct Placement) - Two offers were extended. One acceptance and no rejections were received. There are three open offers.

BS/MS (Program) - No offers were extended. One acceptance and one rejection was received. Two offers are currently open.

Technical Graduate Program - Six Technical Graduates were placed on permanent assignment. No new members were added to the roll. One termination occurred. Current program members total 69.

FACILITIES ENGINEERING

At month end, Facilities Engineering Operation was responsible for nine active projects having total authorized funds in the amount of $6,505,500. The total estimated cost of these projects is $10,799,000. Expenditures on them through October 31, 1963 were $1,578,000.

The following summarizes project activity in November:

Number of authorized projects at month end ---------- 9

New projects authorized ---------- 2
- CAH-116, PRTR Decontamination and D_2O Cleanup
- CAH-119, PRTR Storage Basin and Experimental Facilities Modifications

Projects completed ---------- 0

New projects submitted to the AEC ---------- 1
- CAH-114, Critical Mass Laboratory Addition

Project proposals being prepared ---------- 6
- CAH-123, Laboratory Fire Protection System
- Waste Transport System
- Heat Transfer Apparatus for Model Studies
- 141-M Building Addition
- Geological and Hydrological Wells - FY 1964
- Modifications to 5201 Building

The status of active projects is reported hereafter:

CAH-916 - Fuels Recycle Pilot Plant - Construction is 18 percent complete compared to a scheduled 16 percent. Erection of concrete structures is still the major activity at the construction site. Design changes to eliminate piping interferences in the storage vault and to provide omitted pipe sleeves in the metallurgy cell floor were approved by the Commission. The cost of these changes is being negotiated with the Contractor.

CAH-922 - Burst Test Facility for Irradiated Zirconium Tubes - Construction is 80 percent complete compared to a scheduled 93 percent. Late delivery of the five containment vessels and the instrument panel is delaying project progress. The remaining project work is on schedule.

CAH-962 - Low Level Radiochemistry Building - The Architect-Engineer was scheduled to submit the completed detailed design package on November 8, 1963, after having received comments from the Company and the Commission. It now appears design will be complete early in December. The Company and the Commission are jointly preparing a project proposal revision requesting total project funds.

CAH-977 - Facilities for Radioactive Inhalation Studies - The Company approved the Architect-Engineer's design package and returned it to the Commission on November 22, 1963.

CAH-982 - Addition to Radionuclide Facilities, 141-C Building - Detailed design is 80 percent complete compared to a scheduled 96 percent. Company and Commission representatives reviewed comment drawings on most phases of the design. Final comment drawings are expected from the Architect-Engineer about December 2, 1963.

CAH-999 - Plutonium Recycle Critical Facility Conversion to Light Water - Construction is 73 percent complete and on schedule. Shop mock-up of major working components was completed on November 8. Field installation was started on November 11. Fabrication of the fuel portion of the control rods by Hanford Laboratories is not complete.

CAH-100 - High Temperature Lattice Test Reactor - Title I design by Vitro Engineering Company was completed November 21, 1963, and is being reviewed by the Company. A schedule for detailed design was submitted to the Commission for approval. The total design progress is only 18 percent. At the time the project proposal was approved, it was anticipated that 42 percent of all design would be completed by November 15, 1963.

There is a wide difference between Vitro Engineering Company's calculated shielding requirements for the reactor room walls and those used by the Company in preparing the original cost estimate. The Vitro Engineering Company has been asked to re-evaluate its calculations, and the Company is proceeding with a check calculation by computer in an attempt to reconcile the difference.

Development of prototype equipment and testing of potential reactor metals under anticipated HTLTR operating conditions is being continued by Hanford Laboratories.

CAH-114 - Addition to the Critical Mass Laboratory - The project proposal requesting authorization of design funds was submitted to the Commission on November 4, 1963, and subsequently approved by RLOO-AEC. A letter is being prepared for transmittal to Washington-AEC recommending authorization of the project.

CAH-116 - PRTR Decontamination and D_2O Cleanup Facilities - A directive was issued on October 28, authorizing the requested $43,500 for total design. A work authority dated November 1, 1963, authorized Vitro Engineering Company $37,000. The Commission stated it does not plan to authorize the Company the $6,500 requested in the proposal, but will retain these funds itself. Design has not started.

CAH-119 - PRTR Storage Basin and Experimental Facilities Modifications - The project proposal requesting design funds in the amount of $37,500 ($34,000 for Vitro Engineering Company and $3,500 for the Company) was transmitted to the Commission on October 24, 1963. A directive, authorizing the Commission $37,500 was issued November 15, 1963.

CAH-123 - Laboratory Fire Protection System - Following its review by the General Manager, the project proposal was withdrawn temporarily for modification to answer questions that might arise regarding the safety of installing sprinklers in some special purpose laboratories.

Engineering Services

Engineering work was performed in support of design and construction on active projects, project proposals, preliminary planning, and design criteria for new projects. Principal work items included: (1) field liaison, review of shop drawings, and approval of submitted materials on CAH-916, FRPP, (2) review of A-E design on CAH-962, Low Level Radiochemistry Building, (3) review of A-E preliminary design and consultation with A-E on CAH-982, Radionuclide Facilities, and (4) studies of 325 and 327 building additions and scope and design criteria for 300 Area fire sprinkler systems, waste transport system, modifications to 5201 building and critical mass laboratory addition.

Engineering and consulting work provided to research and development personnel included: (1) engineering assistance on experimental neutron spectrometer, 105-KE building, (2) engineering of cell door operator at 747 building, (3) study of feasibility of extending future waste disposal railroad spur to 324 building, (4) engineering assistance on high temperature water test loop, 314 building, (5) engineering assistance on installation of 329 building X-ray equipment, (6) operation and engineering of NPR charging machine modifications to utilize magazine loader, (7) study of relocation of Biology Laboratory to 300 Area, (8) operation and modification of 108-F source handling equipment, (9) engineering and test work for installation of 20,000 psi pressure bonding autoclave, (10) ventilation studies of 251 and 151-H buildings, (11)

assistance on set-up for air flow test of 314 gas loop fuel section, (12) preliminary engineering and scoping on Containment Systems Program, (13) engineering assistance on fast reactor concept study, (14) engineering on Waste Calcination Program, (15) engineering on HTLTR mock-up instrumentation, (16) engineering assistance on critical mass experiments, and (17) preliminary analysis of building requirements for KAPL Split Table Assembly.

Engineering assistance was provided to CPD for an audit of the 233-S ventilation system and consulting assistance on instrument engineering.

Plant Engineering

Engineering service was provided on numerous maintenance and laboratory modification and improvement jobs. Major items were: (1) engineering for a new 306 building process sewer and engineering analysis of ventilation system, (2) 308 building studies of effectiveness of acoustical treatment and investigation of means to reduce noise transmission from dynapak machines throughout the building, engineering of hood air system to reduce moisture content and trouble-shooting on the control system for the vacuum welding hood, (3) study of filter changing, 327 building, to reduce radiation exposure and analysis of switchgear circuit breaker settings, (4) engineering circuit modification, 3717-B building, engineering of temperature control system, room 40-C, 329 building, (5) engineering and layout for exhaust ductwork for thoria laboratory, and (6) establishment of procedure for review and analysis of new emergency power requirements prior to connection.

Pressure Systems

The C-1 loop was completed and emergency operations were tested. Operation with dummies will be initiated immediately, and a fueled test is scheduled to start January 15.

Inspection of the patch on the PRTR flash tank was made by the third-party inspector and the vessel approved.

Facilities Operation

Costs for the month of October were $160,657, which is 101 percent of the forecast for the month. Fiscal year costs for the first four months were $601,775, which is 106 percent of the predicted. During the month, the total maintenance expenditure continued higher than expected. Engineering costs remained low and tended to offset maintenance costs.

Criticality alarms were tested on November 26.

The following table summarizes waste disposal operations:

Item	September	October
Concrete waste barrels disposed to:		
300-N burial ground	8	0
300-Wye burial ground	10	4
200-W plutonium burial ground	10	0
Loadluggers of dry waste disposed from 325 building:		
to 300-N burial ground	1	0
to 300-Wye burial ground	2	2
to 200-W plutonium burial ground	1	1
Loadluggers of dry waste disposed from other 300 area sites to:		
300-N burial ground	7	0
300-Wye burial ground	13	25

The average cost per million gallons of water produced by the 315 filter plant for the period of March 1, 1963 through October 30, 1963 was $42.89.

Drafting

The equivalent of 153 drawings were produced during the month for an average of 28 man-hours per drawing.

Major jobs in progress are: (1) high temperature gas loop, (2) HTLTR mock-up core, (3) plutonium powder processing line, (4) inhalation hood as-builts, (5) fast super pressure power reactor concept, (6) HTLTR control rods, (7) thoria processing line, (8) dynapak furnace hood, (9) shielded inspection hood, 308, (10) automatic densitometer, (11) rotating crystal spectrometer, (12) capacitor discharge test apparatus, (13) 309 building piping service drawings, (14) modifications to 309 chilled water system, and (15) refraction compounds glove box. Work was also produced in support of engineering reported under previous sections of this report.

Activity during the month on construction work (J. A. Jones Company) being performed for Hanford Laboratories' components is given below:

	Unexpended Balance
Orders outstanding beginning of month	$285 733
Issued during the month (including suppl. and adj.)	153 470
J. A. Jones expenditures during month (includes C.O. cost)	169 755
Balance at month end	269 448
Orders closed during month	230 598

In addition, work on four maintenance work orders issued to plant forces and having a face value of $2,782 was supervised.

Major nonproject jobs in progress during the month were: (1) feeder stalls and feeder troughs, 141-C, (2) building modifications and laboratory furniture, 141-H, (3) concrete pad, 144-F, (4) hay storage area, 100-F, (5) construct maintenance shop addition, 231-Z, (6) install hoods and modify ventilation system, 242-B, (7) construct building addition, 292-T, (8) install electrical bus and water filter, 306, (9) relocate small dynapak and install new dynapak, and renovate room 125, 308, (10) construct block addition for gas loop, install PRTR tube replacement mock-up and install HTLTR mock-up, 314, (11) construct roof enclosure, construct thoria laboratory in room 417, install emergency generator, modify room 520 and construct retaining wall, 325, (12) construct exhaust system, install floor drains and install unit heater, 327, (13) install lighting, 3717-B, (14) enclose mock-up area, 309, and (15) fabricate waste calcination equipment, 324.

Four requisitions were issued during the month totaling $1,500. Total value of equipment being processed is $30,000.

W Sale

Manager
Finance and Administration

W Sale:JVM:whm

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output for November was 357 MWD, for an experimental time efficiency of 43.5% and a plant efficiency of 17%. There were nine operating periods during the month, one of which was terminated for scheduled refueling and planned maintenance, one was terminated manually due to indications of a leak inside of HX-5, three were terminated by scrams while switching to automatic control, two were terminated by spurious flow monitor trips, one was terminated by a manual shutdown when the charge on the 125 V DC batteries was found to be below specifications, and one was terminated by a high ΔP trip on the Rupture Loop. A summary of the fuel irradiation program as of November 30, 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	11	982.3	1	211.2	73	7571.1			85	8764.6
Maximum		99.6		211.2		178.5				
Average		89.3		211.2		103.7				
In Basin	32	2600.1	31	3603.8	14	237.5			77	6441.4
Buried							1	7.3	1	7.3
Chem. Proc.	32	2309.3	35	1965.8	—	—	—	—	67	4275.1
Prog. Totals	75	5891.7	67	5780.8	87	7808.6	1	7.3	230	19488.4

Note: (MWD/Element) X 20 = MWD/TU for UO_2 and PuO_2-UO_2.

D_2O and indicated helium losses for November were 1034 pounds and 136,385 scf., respectively.

A total of 194 reactor outage hours was charged to experimental time. Main items were:

Location and removal of three leaker fuel elements	54 hours
Rupture Loop tests	50 hours
Charge-discharge	42 hours
Core Level Instrumentation test	17 hours
Process tube inspection	15 hours

Equipment Experience

A total of 159 reactor outage hours was charged to repair work. Main items were:

HX-5 leak location and repair	114 hours
Valves	11 hours
Rupture Loop equipment	8 hours

Preventive maintenance required 350 hours or 6.5% of the total maintenance effort.

Control of the PRTR by the automatic controller was demonstrated during periods of constant power level. Operation during automatic control resulted in exceptional power level stability, with fluctuations of one-fourth to one-half megawatt, a factor of eight better than manual control. Problems during transfer from manual to automatic were corrected at month-end.

Work has been completed on placing either low speed primary pump on 384 emergency power. This provides an independent power source for primary coolant and greatly increases reliability of that system.

Core level instrumentation to measure the coolant level in the reactor was completed during the month. System testing remains.

A small leak developed in one of the HX-5 tubes. The defective tube was blocked off and the heat exchanger was returned to service.

Improvement Work Status (Significant items)

Work Completed

Backup Emergency Power to Primary Pumps
Helium Compressor Unloading Orifice Modification
Reactor Automatic Controller
PRTR Data Handling System
Remote Sampling and "A" Cell Sump Pump-Out Facilities

Work Partially Completed

In-Line Gas Sampling
Process Tubes Level Indicator
Inlet Gas Seal Replacement
125-Volt Battery Disconnect Contactor
Shim Rod Shroud to Top Cap Modification
Improved RTD Connector Sealant and Bracing
Instrument Power Transfer System
Installation of New Alarm Annunciator

Design Work Partially Completed

Additional Fuel Storage and Examination Facility
Vibration Snubbers for Earthquake Protection
Decontamination Building and D_2O Cleanup Facility
Flux Wire Scanning System
Supplemental Emergency Water Addition
Permanent Installation of Closed Circuit TV
Rupture Monitoring System Modifications
PRTR Increased Power Level
Battery Power for Galvanometer Light
Containment Valve By-pass for Sump Pumpout Lines

Process Engineering and Reactor Physics

Test 79 (Reactivity Worth of Voiding the PRTR FERTF Channel) was performed. The measured result was a gain in reactivity of + 0.8 mk upon removing the H_2O coolant from the channel, when the channel did not contain a fuel element. With a 1% PuO_2-UO_2 element in place, no reactivity change was observed upon voiding coolant.

PRTR Test No. 78 (Testing of Core Level Instrumentation) was performed during the month. The test demonstrated the ability to control the position of the primary coolant within the primary system but was unsuccessful in proving the desired operability of the core level instrumentation.

Procedures

Revised Operating Procedures issued	2
Revised Operating Standards issued	11
Temporary Deviations to Operating Standards issued	10
Process Specifications accepted for use	4
Equipment Standards issued	2

Drawing As-Built Status:

	November	Total
Approved for As-Built	28	1 060
In Drafting	(2)	18
In Approval	(11)	3
Deleted or voided		81
		1 162
Scheduled for review		329
		1 491

Personnel Training:	Man Hours
Qualification subjects	153
Specifications, Standards, Procedures	64
Emergency Procedures	21
Automatic Controller	13
	251

Status of Qualified Personnel at Month-End:

Qualified Reactor Engineers	9
Qualified Lead Technicians	6
Qualified Technicians	17

Experimental Reactor Services

The status of the various test elements at the end of November 1963, is shown below. Those elements which had reached their assigned goal exposure or had been permanently discharged for other reasons prior to November 1, 1963, have been deleted from the 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	--	110.0 Repad
14	1352	5098	Moxtyl-Vipac	5/8/62	--	178.5 Repad
14	1758	5099	Moxtyl-Vipac	5/8/62	--	131.3 Repad
48	1253	5150	Moxtyl ($\frac{1}{2}$" x $\frac{1}{2}$" Pads)	8/1/62	--	124.3
47	1649	5121	Unautoclaved LX PuAl	6/13/63	--	77.8
47	1552	5194	Unautoclaved LX PuAl	7/6/63	--	71.4
47	1449	5193	Unautoclaved LX PuAl	7/6/63	--	69.7
54	1542	5116	Moxtyl (clip-on pads)	5/8/62	--	126.2
54	1554	5118	Moxtyl (clip-on pads)	5/8/62	--	176.9
61	1249	5185	Moxtyl-Physics	5/28/63	--	89.1
61	1354	5186	Moxtyl-Physics	5/28/63	--	79.1
61	Basin	5187	Moxtyl-Physics	5/28/63	11/22/63	100.3
61	1556	5192	Moxtyl-Physics	6/13/63	--	88.9
67	1152	5119	Moxtyl(Repaired Wire)	10/20/63	--	26.7
67	1457	5117	Moxtyl(Repaired Wire)	10/20/63	--	69.6

Three fuel elements failed and were removed from the reactor. Six fuel elements were inspected in the basin.

Plutonium Recycle Critical Facility

The PRCF core was dismantled. The D_2O moderator and all fuel elements were removed to storage. Safety rods, control rods and miscellaneous hardware items for use with a D_2O moderator were moth-balled. Deactivation work was completed on November 8. Project work (CGH-999, conversion of facility to H_2O moderator) began on November 11 and lasted through month-end.

Inspection of the core vessel interior revealed no active corrosion over the past 10 months of moderator system operation.

Fuel Element Rupture Test Facility

Operation

Operation of the Rupture Loop was initiated during the month. An advanced rupture detection system, installed by Physics and Instruments, was also activated and base measurements established during initial operation prior to installing the first fuel element near the end of the month.

Procedures

Operating Standards issued	5
Temporary Deviations to Operating Standards issued	2
Process Specifications accepted for use	2
Training on Operating Procedures	48 Man Hours

TECHNICAL SHOPS OPERATION

Total productive time for the period was 21,564 hours. This includes 16,106 hours performed in Technical Shops, 3,616 hours assigned to J. A. Jones Company, 1,818 hours assigned to offsite vendors, and 24 hours to other project shops. Total shop backlog is 18,146 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 649 hours or 3.6% of the total available hours.

Distribution of time was as follows:

	Man Hours	% of Total
N Reactor Department	3 903	18.09
Irradiation Processing Department	4 002	18.56
Chemical Processing Department	555	2.58
Hanford Laboratories	13 104	60.77
Hanford Utilities and Purchasing Operation	0	-

LABORATORY MAINTENANCE OPERATION

Total productive time was 19,300 hours of 23,000 hours potentially available. Of the total productive time, 92.5% was expended in support of Hanford Laboratories components, with the remaining 7.5% directed toward providing service for other HAPO organizations.

Manpower utilization for November was as follows:

A.	Shop Work		2 700 hours
B.	Maintenance		7 200 hours
	1. Preventive Maintenance	1 900 hours	
	2. Emergency or Unscheduled Maintenance	1 700 hours	
	3. Normal Scheduled Maintenance	3 600 hours	
	4. Overtime (Included in above figures)	1 300 hours	
C.	R&D Assistance		9 400 hours

WD Richmond

WD Richmond:JGZ: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
J. M. Davidson	Thermal Neutron Flux Measurement by Boron Heating
C. L. Brown and W. P. Ingalls	A Design for Neutron Shielded (Class I) Shipping Containers
J. C. Spanner	A Method and the Apparatus for Measuring the Vibration of Fuel Element Assemblies in the PRTR (HWIR-1668)
W. W. Schulz	A Solvent Extraction Process for the Separation and Recovery of Aluminum from Redox Process-Type Aqueous Acid Waste Solutions
K. L. Chubb and D. C. Thompson	Magnetic Tape Search Circuitry for a Benson-Lehner Electro Plotter

Manager, Hanford Laboratories