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HANFORD LABORATORIES MONTHLY ACTIVITIES REPORT

JANUARY, 1963

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HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT
JANUARY, 1963

Compiled by
Section Managers

February 15, 1963

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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TABLE I - HANFORD LABORATORIES FORCE REPORT

DATE: January 31, 1963

	At Beginning of Month		At Close of Month		
	Exempt	Salaried	Exempt	Salaried	Total
Chemical Laboratory	133	133	131	132	263
Reactor & Fuels Laboratory	185	170	186	173	359
Physics & Instrument Laboratory	94	65	95	66	161
Biology Laboratory	41	58	39	60	99
Applied Mathematics Operation	18	4	18	4	22
Radiation Protection Operation	43	92	43	92	135
Finance & Administration Oper.	117	110	113	112	225
Programming Operation	12	3	11	3	14
General	2	3	3	3	6
Test Reactor & Auxiliaries Oper.	57	297	56	297	353
TOTAL	702	935	695	942	1637

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BUDGETS AND COSTS

January operating costs totaled $2, 355, 000, an increase of $131, 000 from the previous month; fiscal year-to-date costs are $15, 996, 000 or 53% of the $30, 139, 000 control budget. Hanford Laboratories' research and development costs for January, compared with last month and the control budget are shown below:

(Dollars in thousands)	COST Current Month	Previous Month	FY To Date	Budget	% Spent
HL Programs					
02 Program	$ 70	$ 68	$ 505	$ 1 069	47
03 Program	4	2	41	175	23
04 Program	942	759	6 573	12 170	54
05 Program	113	101	681	1 373	50
06 Program	282	241	1 797	3 154	57
08 Program	2	6	9	100	9
	1 413	1 177	9 606	18 041	53
N-RD Sponsored	152	149	301	1 270	24
IPD Sponsored	71	71	626	932	67
CPD Sponsored	106	103	844	1 421	59
FPD Sponsored	--	--	493	493	100
Total	$1 742	$1 500	$11 870	$22 157	54%

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

Production of supports for N fuel elements by an off-site vendor has been discontinued because of cracking of the fabricated strip during forming. Reactor and Fuels Laboratory, which had been providing satisfactory supports on an interim basis, will now continue to do so until a satisfactory commercial supplier can be established.

A process for the forming of ductile Zircaloy-2 sheet from Zircaloy-2 billets was demonstrated. An extrusion of a 6-inch billet to a 0.5-inch-thick by 3.5-inch-wide sheet bar was formed and reduced to 0.035-inch sheet by

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warm rolling. Laboratory tests indicated this product had good bend ductility and could be readily formed into N-inner fuel supports. Sheet formed by this process is being used in the production of a test lot of supports for N-RD.

Some specimens of the self-brazed closure fuel element have been exposed in a high temperature, out-of-reactor loop at KE-Reactor for the past 2 weeks. One of these had the weld bead removed to lay bare the braze layer, but examination revealed no evidence of accelerated attack on this braze layer. The slugs were returned for further exposure.

Detailed examination of boiling burnout data obtained with electrically heated models of that portion of the N fuel element forming the middle flow annulus and adjoining heated surface showed that when the annulus thickness was decreased by 80% along the bottom (as would occur if portions of the self supports broke off), the burnout heat fluxes were 50 to 55% than those obtained with a concentric test section. Even so, boiling burnout would not be expected for a single element with this degree of eccentricity in the N-Reactor under normal operating condition.

Room temperature measurements of the susceptibility to brittle fracture of 30% cold worked N process tubing were made. The tubing does not fail in a brittle manner at room temperature and a pressure below 75% of ultimate, but cracks do propagate at 85% of ultimate strength. This performance is nearly identical to that of 50% cold worked KER tubing previously tested.

A remote-fired-gun system to shoot shaped projectiles into unirradiated pressurized tube specimens for brittle fracture studies has been developed, tested, and is now being used in a test program.

Examination of a Zircaloy-2 clad heater rod exposed for 50 days to lithiated water at pH 10 and 628 F and operating with a heat flux of 280,000 Btu/hr-ft^2, showed rapid corrosion (up to at least 12 mils) and formation of thick films in crevice regions adjacent to fuel support feet where nucleate boiling concentrated the LiOH.

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Fuel elements from the single-tube test in KE-Reactor (in which quachrom glucosate is being evaluated as a replacement for dichromate to reduce effluent activity resulting from chromium) are scheduled for discharge and inspection in early February.

Coupon tests to determine corrosion rates of carbon steel in cold process water at pH of 6.6 and 7.0 indicate the lower pH increases the corrosion rate by 50% to about 6.0 mils per year.

Recent startup of the PRTR Fuel Element Examination Facility provided the first opportunity to carefully inspect the worn contact surfaces of the end brackets of a PRTR fuel element (No. 5110). These surfaces had caused excessive wear of a process tube. Severely worn bottom bracket contacts were scarred by charge-discharge operations. Very little wear was visible on the top bracket. Bundle wrap bands were displaced, presumably because of a charging mishap.

Examination of three PRTR elements equipped with extended surface clip-on wear pads revealed they had showed no measurable pad wear after 27 days of operation. Pads were applied remotely to six additional irradiated fuel elements.

Destructive examination of Zircaloy cladding from a swaged UO_2-PuO_2 PRTR fuel element revealed an inner surface reaction layer, 0.0005-inch thick, of unknown composition.

Evidence that the equilibrium location of fission products may shift as rod power is increased was revealed by studying high resolution autoradiographs of irradiated $(U, Pu)O_2$ pellets. Similar studies showed essentially no relocation of fission products in the immiscible MgO-PuO_2 fuel system, in contrast to extensive relocation in $(Zr, Pu)O_2$ solid solutions.

A fuel element capable of interchange between the PRTR and the Fermi reactor blanket was designed. Components for thermal hydraulics studies are being assembled.

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Initial reactivity burnup studies (in cooperation with P&IL personnel) in the series of Phoenix fuel irradiation experiments were completed. New irradiation tests to demonstrate Phoenix action are being studied.

The 4-rod cluster rejuvenation fuel element successfully completed one irradiation cycle in the MTR at the Idaho Testing Station.

A 1-foot-long prototype of an "arch-supported," one-piece, PRTR fuel element cladding was fabricated off-site by high-voltage electron beam welding.

Reactor quality UO_2 fuel material was produced by high-energy-rate impacting, using a tool steel punch, at 200,000 psi without subsequent heat treatment to achieve stoichiometry, to remove trapped gases, or to increase density by sintering.

The PRTR mixed oxide fuel contains one part of well homogenized, fine particle size 5% PuO_2-UO_2 and nine parts of larger particle size arc fused UO_2. There has been evidence of partial segregation of these two fractions into bands. Heat transfer and physics calculations indicate that peak-to-average heat flux ratios of 2.37 would occur for fuel rods loaded in 80 increments (1-inch total band length) if segregation were complete; this would result in a maximum heat flux of 660,000 Btu/hr-ft^2 for a fuel element operating at a tube power of 1200 kw.

Based on the excessive absorption of hydrogen from dry CO_2 and H_2 mixtures, CO_2, does not appear to be a satisfactory hydriding inhibitor for Zircaloy-2.

The presence of $KMnO_4$ or its reduction products partially inhibits stress corrosion cracking of AISI 304 and 316 stainless steel by 18% sodium hydroxide.

Corrosion rates for Zircaloy in high temperature D_2O are similar to rates in H_2O. The fraction of corrosion product D_2 absorbed in Zircaloy from D_2O is lower than fraction of H_2 absorbed from light water environment.

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Notification was received of AEC approval to operate the PRCF with D_2O-moderated zoned cores containing UO_2 and Pu-Al fuel elements. Safeguards studies supporting PRCF operation with other D_2O-moderated cores and "short cooled" irradiated fuel elements have been compiled in a comment draft.

After a study (reported in HW-76313) of fuel element cooling in the PRTR following a total electrical power failure it was concluded that, if liquid-phase convection circulation is inadequate, cooling of the fuel elements will be maintained by boiling convection circulation provided the fuel elements remain flooded.

Nine different PRTR pressure tubes were inspected during the month. A gas gap spacing between the pressure and shroud tube of about 45 mils was found in one channel, but this value is essentially unchanged from the last measurement made 2 operating months ago.

Variations in zirconium concentration in PRTR primary system water have been noted during reactor operation, but correlation of the data with frequency or severity of fretting corrosion in the PRTR have not been possible to date.

Using either 3 or 10% ammonium citrate, difficulty was experienced in reducing activity levels in the ETR G7-M3 Loop with the APACE decontaminating procedure. It is quite possible that these films are similar to the adherent films found in PRTR during its decontamination.

A suggestion of Draley and Ruther (ANL) that dynamic aluminum corrosion in high temperature water should be suppressed by the presence of corroding steel is not supported by loop experiments conducted in this Laboratory.

The oxidation characteristics of Haynes 25, a cobalt base alloy, have been studied at 1000 C. The effect of surface treatment and oxygen partial pressure has been determined qualitatively.

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Procurement of materials for the coordinated Irradiation Effects on Reactor Structural Materials Program is continuing. Ten tons each of A212B and A302B have been rolled and heat treated. The tensile strength of the A302B alloy , however, exceeded that specified and a redraw operation at 1250 F was necessary to improve its ductility. Bids have been received for quantities of Inconel 600, Inconel X-750, Am 355, AISI 304 SS, AISI 348 SS, and Zircaloy-2.

Tensile data on AISI 304 and 348 stainless steels which have received exposures up to 1.1×10^{20} nvt indicate a large increase in yield strength and a lesser increase in tensile strength resulting from neutron irradiation. These same steels show smaller but nonetheless large increases in yield strength after prolonged residence in 280 C water in the out-of-reactor loop with no neutron exposure whatsoever. Zircaloy-2 also displays an increase in yield strength and decrease in uniform elongation after out-of-reactor exposure to 280 C water, but the magnitude of the effect is less than for the steels.

A basic study of the effects of irradiation hardening on creep has been initiated. The material being used is high purity polycrystalline copper. Techniques have been developed to measure creep of irradiated copper specimens at temperatures as low as 77 K. Data obtained to date indicate unmistakable differences in the obstacle geometries which control creep in annealed and in irradiated (5×10^{17} nvt) polycrystalline copper.

Metallographic examination has continued on the ruptured, irradiated Inconel tube from the DR-1 gas loop. Samples taken from regions where temperatures were 700–800 F have 1 mil or less of oxide at the inner and outer surfaces. Samples taken from the vicinity of the transverse failure, where the temperature was 850–950 F, had oxide coatings up to 5 mils in depth.

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High purity molybdenum foils of 0.003-inch thickness were heated to successively higher temperatures—2100, 2200, 2300, and 2300+ C—and quenched rapidly in a flow of helium. The specimens were then thinned and examined by electron microscopy. No defects were observed in the quenched foils. The quenched foils will now be aged in attempts to permit vacancies to cluster.

Length changes in single crystal molybdenum specimens 1.5 inches in length containing carbon in one of three concentration levels have been determined as a function of irradiation to either 10^{18} or 10^{19} nvt. Of the six crystals measured to date, two show no change in length; whereas, the remainder show increases in the range 40 to 120 microinches. In progress is a detailed evaluation of these changes with respect to the known crystallographic orientation of the crystals and with respect to lattice parameter changes yet to be determined.

One-half-inch-diameter by 0.005-inch-thick unirradiated foils of molybdenum which had been annealed previously at 1900 C have been successfully strained 1 to 2% by a specialized technique.

Metallographic examination of uranium irradiated to 0.27 at.% burnup at 625 C indicates that the fission gas pores that have been observed to segregate at grain boundaries in lower burnup specimens have now coalesced into grain-boundary cracks and large irregular pores at triple points of grains.

Electron microscope examination of two specimens of Th–U^{235} irradiated to about 0.1 at.% burnup, and of one nonirradiated thorium control specimen has revealed no effect of irradiation with regard to etching characteristics of the matrix, nature of the second phase or existence of pores.

Successful coextrusion of Th–2-1/2% U (2.3% enriched)–1% Zr alloy was accomplished on January 18. Billets were prepared by double melting

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in the consumable vacuum arc furnace in the 306 Building and coextruded on the 2700-ton press in the 333 Building. Two extrusions of 17 feet and 21 feet were produced, each of 1.75-inch OD x 1.05-inch ID x 0.025-inch Zircaloy-2 cladding. Billets were canned in copper and extruded at 760 C from a 6-inch container at a reduction ratio of 17 to 1 and an extrusion constant of 18.2 tsi. The extrusions are currently being sampled for evaluation and will subsequently be fabricated into 8-inch fuel elements for irradiation tests in a high temperature, high pressure water loop in the ETR.

Preliminary results indicated that the region between O/Pu = 1.0 and 1.5 in the Pu-O phase system may be a two-phase field of PuO and ρ-Pu_2O_3.

The approximate melting point of PuO was determined to be 1800 C.

PuO_2 and MgO appear to be entirely insoluble in each other in the liquid state.

Analysis of X-ray powder patterns of a 48.2 at.% C—Pu-C alloy indicated the possible growing-in of the zeta phase during a period of 400 days at room temperature.

In cooperation with Analytical Laboratories, a combustion gravimetric method for determination of carbon in plutonium carbides was reduced to routine practice in glove boxes.

A compact of 10% UO_2 in molybdenum having a uniform dispersion of 0.5 micron UO_2 particles and 85% TD was successfully fabricated. Starting powder was made by coprecipitating uranium and molybdenum oxides, roasting in air, and reducing in hydrogen **at** at 1200 C. Cold compaction and sintering in hydrogen at 1600 C followed.

Encouraging preliminary data were obtained on the thermoelectric properties of UO_2 in a test simulating in-reactor heating.

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Examination of a W-UO_2 cermet irradiated at a cladding surface temperature greater than 2100 C revealed no reaction between the UO_2 and tungsten matrix and no dimensional change.

Significantly greater thermal conductivity of UO_2 single crystals than of polycrystalline UO_2 was demonstrated by irradiation test.

Positive identification of the UO_2 microstructure associated with the radial limit of initial melting was obtained by the discovery and location of previously molten tungsten particles in a UO_2 specimen from the marker wire irradiation experiment.

A 50 wt% uranium monosulfide-tungsten cermet was compacted by high energy impaction at 1200 C to 98% TD.

Autoradiographs of a fuel core exposed to approximately 20,000 Mwd/ton showed a concentration of fission products in the same area which contained a concentration of metallic inclusions.

Fabrication was completed on a capsule designed to investigate the initial melting and possible resolidification of a UO_2 fuel core under operating conditions.

Time-lapse moving picture techniques were used to show that the high temperature reactions occurring in a UO_2-W cermet during reflection electron microscopy begin abruptly and proceed rapdily to completion within a few minutes.

Polystyrene replication has been completed successfully on selected areas of a nonirradiated UO_2 fuel element.

Initial ultramicrotomy experiments have shown that UO_2 crystals can be thin-sectioned even with simple glass knives.

A new technique was used to determine nearly 0.1% free uranium metal in UO_2 previously analyzed by coulometric titration techniques as hyperstoichiometric material.

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Special fabrication activities included completion of the fission product transient samples for Phillips Petroleum Company, work on four types of elements for physics tests, and work on coextruded Pu-Al elements for corrosion tests.

Material procurement and design of facilities for making the EBWR plutonium loading continued on schedule. A project proposal was prepared for installation of vibrational compaction facilities; loading studies were initiated. Review and revision of the fuel design are in process.

In basic studies relating radiation damage of graphite with fundamental properties of the material, sonic-modulus determinations were used to determine the phonon mean free path which is analogous to a crystallite size. When the method was applied to irradiated lampblack graphite, a decrease in crystallite size was indicated. X-ray measurement, on the other hand, implied a doubling in size. A plausible explanation for the discrepancy exists.

It is believed that sonic-modulus data may be of considerable value in understanding the fundamentals of radiation damage in graphite.

Calculations for the fast supercritical pressure power reactor concept to determine the sensitivity of reactivity values to the group cross sections for U^{238} which span the resonance region showed that sufficient numbers of neutrons are captured in this portion of the spectrum to substantially affect the coolant coefficient. The use of unshielded cross sections in previous design calculations overestimates the void coefficient, making it more positive while simultaneously making the flooding coefficient more negative. The net effect of properly self-shielding these cross sections will be a decrease in enrichment and increased reactor stability.

The application of the "moderator-segmented fast core" concept to the Plutonium Fuel Space Reactor, with the addition of boron as a burnable poison adjacent to the moderated area, may decrease mechanical control requirements from about 35% Δk to 15% Δk. The corresponding increase in reactor diameter is only about 15% above that for the straight plutonium-fueled core.

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Nineteen boiling burnout points were obtained with an electrically heated model of a 19-rod fuel bundle with "warts" to maintain 0.050-inch spacing between rods; wire wraps are usually used to maintain the spacing. The tests were run with a 19-1/2-inch-long test section in a horizontal position and the data covered heat fluxes from 300,000 to 1,600,000 Btu/hr-ft^2. Preliminary comparison of the data with previous data obtained with wire wrapped test sections indicated almost no difference in the results over the range of variables investigated.

The dome seal and nozzle assembly being tested in support of the AEC-AECL Cooperative Program were completed and successfully passed a hydrostatic test and 116-hour leakage test with no evidence of leakage.

2. Physics and Instruments

Analog simulation of the N pressure injection system is satisfactorily underway. Other N system studies and simulations are progressing in conformance with N-RD schedules through operation of the analog computer on a two-shift basis.

The studies on reactor automatic control systems continued with the development of analog simulations of equivalent reactor models in one-, two-, or three-dimension representations. Analyses further demonstrated that the equivalent models provide a simplified method for measuring and specifying reactor control parameters.

An exponential mockup of N-Reactor was used to determine that the reactivity worth of a control spline inserted in a cross coolant channel is about 1/5 that of a horizontal control rod.

Criticality measurements continued on plutonium nitrate solutions using an 11.5-inch-diameter, water-reflected spherical vessel and solutions in the 300–450 g Pu/ℓ concentration range. It was found that the stainless steel vessel wall becomes more effective in reducing the worth of the

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reflector as the plutonium concentration is increased. A method for calculating the buckling of partially emptied spheres also was developed to aid these studies.

Preparations for criticality experiments with plastic plutonium-oxide mixtures continued. Safe storage cabinets for the fuel blocks were completed, and development and testing of the split table criticality machine continued.

Theoretical critical mass studies included transport theory calculations on plutonium spheres dissolving in H_2O, completion of a report (HW-75887) on subcritical interaction studies, and inclusion of polystyrene in the work on developing scattering kernels for various moderators.

Neutron safety consulting services were provided regarding a plutonium metal mold, a molten salt bath, a plutonium fuel element storage array, and a storage problem for low enrichment uranium.

In the cross-section program, the triple axis spectrometer was prepared for measurements of inelastic neutron scattering from H_2O at elevated temperatures, and fast neutron cross-section data obtained previously were processed using the IBM-7090 computer.

The last of the first three L_x Pu-Al PRTR physics elements has been disassembled and samples have been cut from selected rods for burnup analysis. The remaining three elements will be tested in the PRCF prior to destructive analysis. In other PRTR studies, several physics computer codes are being set up for detailed analysis of experiments performed in the PRTR.

Several physics methods and cross-section sets that are employed in fast reactor studies are being checked against data from two fast critical experiments. These evaluations will give some insight on the reliability of these methods.

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The calculated flat reactivity-lifetime response of Phoenix-fueled Zr-H_2O reactors over a wide range of fuel loadings has now been confirmed for initial plutonium compositions of 5, 10, 15, and 20% Pu^{240}.

In fuel cycle code developments, changes have been made in the Monte Carlo portion of reactor burnup that yield considerable reduction in computer running time. Three cases have been successfully run using the CALX burnup code, a constant flux burnup, a constant power burnup, and a constant flux recycle case. These were run using an input data tape.

PRTR measurements were made on high exposure (20.6% Pu^{240}) Pu-Al fuel in a 10.5-inch graphite lattice. Preliminary calculations also indicated that the PCTR can be used for measurements on large Pu-H_2O systems by using polyethylene to mockup the H_2O moderator.

Neutron flux traverses through the PRCF showed significant photo-neutron generation in the D_2O moderator due to gammas from irradiated fuel stored adjacent to the PRCF vessel.

The cost of the High Temperature Lattice Test Reactor as currently scoped was estimated to be $2,350,000; AEC assigned a budget figure cost of $2,500,000.

An invention disclosure was submitted on a new concept for an in-core neutron flux monitor. Signal current is generated by the beta decay of B^{12} following the $B^{11}(n, \gamma)\ B^{12}$ reaction. Calculations predict excellent sensitivity, an extremely long operating life, and capability for operation at temperatures up to 1000 C. Preparations for experimental tests are under way.

In nondestructive testing research, studies of ultrasonic wave propagation in attenuating media have revealed that the ordinary law of refraction (Snell's Law) must be modified to suitably account for attenuation effects which may be important in practical testing applications.

A Ceramic Fuels laser utilizing a ruby crystal and a Xenon flash tube was successfully placed in operation. Early tests revealed that the brilliant red output beam was not as coherent as desired.

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Instruments are being developed and designed for telemetering data to central receiving stations from sensors implanted in animals used in biology research.

Orderly progress was made on a number of radiological physics programs, including: the study of differences in potassium measurements by different whole body counters, the development of P^{32} counting capability for a mobile counter, the development of X-ray scintillation counting methods and pulse shape discrimination neutron dosimetry methods, and preparation for gamma-ray calorimetry measurements on Pu^{147}.

Analyses of the large quantity of atmospheric dispersion data collected at Hanford, Cape Canaveral, and Vandenberg Air Base reached the point at which comparisons between sites could be made. For similar conditions of terrain and micrometeorology, the dispersion measurements were strikingly similar at all three locations. Significant differences between sites due to terrain and meteorological regimes not common to all three sites provide valuable data for evaluating the local effects.

The developmental real-time air sampler for use in dispersion experiments was successfully field tested to a distance of 1 mile from a ground level source.

3. Chemistry

A recently developed computer program is being used to better estimate the flow of contaminated water from the N-Reactor crib and subsequent fission product release to the Columbia River. The only fission products expected to reach the river in significant concentrations are I^{131} and Sr^{90} with travel times of 11 days and 3.5 years, respectively.

In reactor effluent radioisotope reduction studies, anodized and sealed, autoclaved and silicate-treated aluminum coupons were tested for arsenate absorption. Autoclaved and anodized specimens showed the least and greatest arsenate absorption, respectively.

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The marked increase of gross beta-emitter concentration in ground water beneath the Purex A-10 process condensate crib is correlated with a Purex process change whereby single-stage, vice double-stage, acid distillation is employed.

Scintillation probing of vertical wells in the Purex 241-A tank farm resulted in the detection of possible process line or tank leaks on the north side of the 101 tank and on the south side of the 106 tank.

In solid state electromigration studies, the passage of a heavy, low voltage, electrical current through a 99.9+% aluminum specimen containing radioactive Fe^{59} failed to affect migration of the tracer over an extended time period at an operating temperature of 530 C.

Phosphotungstic acid is found to be adversely affected by alkaline solutions to the extent it cannot be used a second time for the extraction of Cs^{137} from Purex FTW.

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

Laboratory and engineering studies were concluded testing a flowsheet for the destruction of nitric acid in Purex 1WW by sugar treatment. Efficiency of nitrate destruction (moles of nitrate destroyed per mole of sugar consumed) was about 20 and residual carbon contents were low, i.e., from 0.4 to ~3%.

An effective solvent has been found to be 0.3M tri-lauryl amine (TLA) in Soltrol, for the extraction of neptunium and plutonium from Purex 1WW. Over 93 and 98% of the neptunium and plutonium, respectively, are readily extracted. Stripping the organic phase with a comparable volume of oxalic acid removes 85 to 95% of the above products, respectively.

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A solution of ammonium carbonate and ammonium hydroxide was found to be quite effective in removing sodium from Duolite C-3 resin following the cesium (and some sodium) extraction from Purex high-level, alkaline supernate waste. Ninety-nine percent of the sodium on the ion exchange bed was readily removed with very small cesium losses.

About 10,000 curies of Pm^{147} were recovered and highly purified by a cation exchange process. The aqueous product was converted to the oxide which will be used as the starting material for measurement of pertinent chemical and physical qualities which have not been determined previously or with high precision.

Use of a vibrating air hammer has been found to be an effective means of removing unirradiated or irradiated UO_2 from 1/2-inch-diameter, Zircaloy-clad PRTR fuel rods. Uranium losses to the cladding range between 0.01 to 0.06%, depending upon the mode of initial fuel rod manufacture.

Graphite electrodes coated with 1-mil-thick pyrolytic graphite are found to be far superior to nuclear grade graphite as a cathode material for the electrolytic preparation of UO_2. The UO_2 deposit is very easily removed from the coated cathodes, and the carbon contamination of the UO_2 is substantially reduced from that observed with uncoated cathodes.

Continued studies of the electrodeposition of PuO_2-UO_2 in the attainment of plutonium enrichment factors as high as 13.5. The enrichment is achieved by electrolyzing an equimolar LiCl-KCl melt containing uranium and plutonium while sparging with 80% O_2—20% Cl_2. Recovery of PuO_2 from the PuO_2-UO_2 co-deposits is possible by selective dissolution of the UO_2 in fresh melts which are sparged with O_2-Cl_2 mixtures. Overall plutonium decontamination factors for uranium and promethium are about 500 and 300, respectively.

Engineering studies of the operation of the 18-inch radiant heat spray calciner show that the use of a large-channel recycle tube is effective in greatly reducing calcine buildup on the reactor walls and increasing throughput capacities.

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Study flowsheets were prepared (based on existing technology, where possible) for the manufacturing-scale purification and packaging of selected fission products, namely, Sr^{90}, Cs^{137}, Ce^{144}, and Pm^{147}. Feed for the packaging plant is assumed to be the output of the Hanford Waste Management operations.

4. Biology

Remodeling of 144-F Building is completed. Four offices and the lunch room were converted into one large physiology laboratory and one animal colony record office. Crowding in the dog colony continues to be a problem. Most all of the runs now house three adult dogs—a hazardous practice due to tendency toward fighting.

Water from melting snows combined with ice pressures to wash out both dams at Rattlesnake Springs. Some recording equipment was lost but most survived the flood.

SULFAM-3 and BISULF-16, reactor tube cleaners, are toxic (LD-10) to young salmon at 146 and 175 ppm, respectively.

Fish from 100-K aquatic biology troughs continue to yield viable columnaris from 36% of the gills sampled.

Yearling trout fed Zn^{65} retained 26% of the dose when killed after 31 days. Highest concentrations of Zn^{65} were present in the gill filaments.

Sr^{90} sources consisting of Sr^{90} plaster of Paris mixtures on glass plates were prepared for use as autoradiography standards and dose rates measured with an extrapolation chamber.

All three female miniature goats have become pregnant, and one delivered triplets. Daily low-level I^{131} feeding was started during pregnancy and thyroid uptake determinations made.

Thyroid I^{131} concentrations in cows range from 1.5 to 3 times the daily intake. The I^{131} concentration in milk has gradually increased from a level of 6 to 18% of the daily dose.

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Male rats were found to display more resistance to acute Np^{237} toxicity than females. This observation correlates with differences in liver lipid content and subcellular distribution of the neptunium deposited in the liver.

Dogs exposed 4 months ago to $Ce^{144}O_2$ aerosols are displaying toxicity symptoms which are significantly different from those observed in plutonium-exposed dogs.

Increasing carrier iodine by a factor of 10^8 in rat inhalation experiments reduced the proportion of total body I^{131} deposited in the thyroid by a factor of 10.

Three years after inhalation of less than 10 mc $Pu^{239}O_2$, 5 of 17 dogs have respiratory rates approximately four times normal. Twelve of the dogs show significant lymphopenia. Dogs from this group which died or were sacrificed during the past year showed translocation of as much as 50% of plutonium to bronchial lymph nodes.

Rats exposed for 60 days to Y^{90} in their drinking water showed a 25% incidence of corneal opacity and ulceration with no evidence of cataract formation. This observation is possibly explained by the lack of penetration of the Y^{90} beta to the lens and should be of some clinical interest in view of the use of Y^{90} in therapy for human eye lesions.

One species of Ostracod identified from Rattlesnake Springs had previously been reported only from Yucatan and Trinidad.

Radioiodine concentrations in North American deer and elk thyroids decreased during the month. Thyroids from Montana elk contained concentrations of I^{131} comparable to those from Maryland deer (1 to 2 nc/g). California deer thyroids were at the lower limit for I^{131} detection (0.05 nc/g).

Wintering waterfowl populations within the Hanford Reservation increased steadily during the month to a maximum of 330,000 ducks and

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geese near the month's end. This was the greatest number of waterfowl observed this season and was 60% above peak populations last year.

5. Programming

In certain circumstances it may be attractive to enrich depleted uranium with U^{233}. To calculate the burnup cost of such a fuel loading, the cost of separating the U^{233} from U^{238} has been examined. The separation of U^{233} from U^{238} requires roughly one-third the energy required to separate U^{235} from U^{238}; this leads to the calculation of unusually low values (less than 0.1%) for the optimum tails composition from the diffusion cascade. Degrading U^{233} by mixing with U^{238} to an enrichment of 2% results in decreasing the value of the U^{233} by \$2.00 to \$2.50 per gram.

TECHNICAL AND OTHER SERVICES

A CPD process operator assigned to the 234-5 Building sustained a contaminated puncture wound in the hand in June 1962. Surgical excision of the contaminated area at the time of the injury was successful in removing about 1 μc plutonium. Subsequently, 0.01 μc plutonium was excreted as a result of treatment with DTPA. The medical treatment with DTPA delayed the evaluation of the employee's body burden. The current evaluation now shows that the body burden is 0.018 μc plutonium or about 45% of the maximum permissible body burden. The original wound site before excision was estimated to have 25–30 times the maximum permissible body burden.

Four new plutonium deposition cases were confirmed by bioassay analyses during the month of January. The new deposition cases resulted from three previously reported contamination incidents involving CPD employees at the 234-5 Building in November and December, and one contamination incident that occurred this month at the same facility. In three of the cases, including the one that occurred in January, inhalation was attributed as the mode of intake and the plutonium body burden was estimated to be 1% or less of the permissible body burden (MPBB = 0.04 μc). The fourth case occurred as the result of a plutonium contaminated injury received by a CPD

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process operator in November. Medical excision of tissue at the time of the accident reduced the plutonium contamination in the puncture wound from 10^{-2} μc to 10^{-3} μc as measured by the wound counter. The current evaluation of this case places the body burdens at less than 10% of the permissible body burden.

The total number of plutonium deposition cases that have occurred at Hanford is 316 of which 230 of the cases involve employees now at Hanford. The number of new deposition cases that occurred in CY-1962 was 32, all resulting from known contamination incidents.

A document, written jointly with IPD personnel, was issued. It updated postirradiation results from the Quality Certification Program. In addition to "fitting" the data in assessing the significance of apparent trends and cycles over time, comparisons were made between reactors.

The document presenting the probabilistic functions which describe unscheduled reactor outages was issued. These functions were derived for use in generating outages in the Reactor Simulation Study.

During the past month, work started on the N-Reactor reliability study; four-state reliability algebra will be applied. Areas of progress to date include: scoping the analysis, investigation of the physical and logical relationships between the reactor subsystems, and logical representation of the criteria used to determine the reactor's mode of operation.

A document was issued giving comprehensive rail accident statistics. Various cause, damage, and speed relationships were discussed in this document. These will be of use in reaching decisions regarding procedures to follow in shipping radioactive materials.

The study of security infractions at HAPO was completed and an oral presentation of results was made. The study was expanded to include a review of HAPO incidents involving lost or forgotten badges during 1962. Study results were put into an informal written report which included suggestions for modification of current data collection efforts.

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A mathematical model of the N-Reactor stack gas chemical reactions has been completed as part of the general study of the zirconium-graphite compatibility problem. The model consists of a simultaneous set of first order nonlinear differential equations whose solutions yield the stack gas composition as a function of position in the reactor. An attempt is being made to study these solutions as a function of the controllable parameters on analogue computing equipment. A digital program is also being written so that the findings of the analogue studies can be investigated in greater detail.

A new model was fitted to two sets of gamma absorptiometer calibration data. Each set of data consisted of calibrations on four sample cells each run four times at nine uranium concentrations. The new model is a modification of Beer's Law to include a second exponential term representing a scattering of gamma photons at higher uranium concentrations.

The results of a series of experimental runs of shear-spinning certain metallic shapes has led the customer to choose a preferable method of designing the original blanks. The mathematics for this mode of design has been programmed for the digital computer. This program computes the details of the blank contours and the requisite tool center coordinates for either machining blanks directly or forming appropriate molds in which blanks may be cast.

The EDPM program which produces tables of the Lamb wave frequency equation for arbitrary elastic material constants is virtually complete. These tables permit a more accurate interpretation of experimental data obtained by ultrasonic nondestructive testing devices which employ Lamb waves.

The program to index hexagonal crystals was made operational during January. A section was added to the program which computes the standard deviations of the lattice constant estimates. Approximately ten sample cases have been successfully indexed.

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Authorized funds for seven active projects total $1,229,500. The total estimated cost of these projects is $7,524,000, of which $613,000 had been spent through December 31, 1962.

SUPPORTING FUNCTIONS

The output of the Plutonium Recycle Test Reactor for January was 617 Mwd for an experimental time efficiency of 63% and a plant efficiency of 44%.

There were six PRTR operating periods during the month; one of which lasted through the month-end. Each of the other five were terminated manually as a result of these operational difficulties: highly dissolved O_2 content in the primary system, once; excessive D_2O recovery system collection (cap gasket failures), twice; suspected fuel element defects, twice.

The core loading at startup on January 3 consisted of 26 UO_2 elements, 29 Pu-Al elements, and 30 UO_2-PuO_2 elements. At month-end the loadings were: 24 UO_2, 24 Pu-Al, and 37 UO_2-PuO_2 elements. Fuel exposure history at month-end:

Maximum UO_2 exposure/element	2560 Mwd/ton_U
Average UO_2 exposure/element	1651 Mwd/ton_U
Maximum Pu-Al exposure/element	73.0 Mwd
Average Pu-Al exposure/element	52.3 Mwd
Maximum Moxtyl exposure/element	27.9 Mwd ($\sim$558 Mwd/ton_U)
Average Moxtyl exposure/element	12.8 Mwd ($\sim$ 257 Mwd/ton_U)

Unusual activity indications on PRTR's rupture detection system resulted in two unscheduled reactor outages. In both cases, a total-stagnant-water test was conducted, and in a third instance, five tubes were checked in this manner. As a result of these tests five elements were removed; it was

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suspected that they were leaking. One element was later cleared as a "suspect" and was recharged. Confirmation tests on all "suspects" were initiated.

A total of 81 outage hours for PRTR were charged to repair work. Major contributors were tube-to-nozzle gasket repair and helium compressor problems. In addition, a total of 45 hours of outage time was charged to high O_2 in the helium and D_2O systems. O_2 problems were traced to shaft seal in-leakage on the core blanket blower.

The mechanical seal on the Plutonium Recycle Test Reactor's PP #1 was replaced after approximately 550 hours of running time.

During the month of January, there were 90 fuel element movements, in and out of the PRTR process tubes. Since the reactor was first loaded, there have been over 2000 such movements—some to the storage basin and back, and some from process tube to process tube. The process tubes now in service have averaged 23 movements with one tube having 43 movements to date. One fuel element, a UO_2, has been moved 21 times.

Cell leak rate testing of the PRCF required considerable time. Minimum consistent value at 6 inches water gage was 1200 ft^3/day with inability to reduce below 1000 ft^3/day. A discrepancy (22–23 inches) between weir level and moderator level was detected with the full flow moderator addition pump in service. Piping changes relieved hydraulic resistances thus correcting this problem.

Base count measurements using the source were taken in preparation to loading the PRCF.

The Fuel Element Rupture Test Facility's design features which will permit complete loop testing out-of-reactor were completed. Work progressed on the structural steel for shielding, revised high- and low-pressure relief systems, the emergency depressuring valve, and the inlet piping shut-off valve. No Design Test items were completed.

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The project completion date for the Gas Cooled Loop was extended to June 30, 1963, by Directive No. AEC-145, Modification 8. The project is 94% complete. Reinsulation of main loop piping following heater installation is essentially complete. The second and third blower units were tested by Bristol-Siddeley at the minimum conditions. Tests at more severe conditions were planned. Work has been completed on several of the punch-list items.

Total productive time for the period for Technical Shop's Operation was 23,884 hours. Distribution of time was as follows:

	Man-Hours	% of Total
N-Reactor Department	4 948	20.72
Irradiation Processing Department	4 321	18.10
Chemical Processing Department	662	2.77
Hanford Laboratories	13 953	58.41
Construction Engineering and Utilities	0	0

Total productive time realized for Laboratory Maintenance Operation was 16,900 hours of a possible 17,600 hours potentially available. Of the total productive time realized, 91% was expended in support of Hanford Laboratories components with the remaining 9% directed toward providing service for other HAPO organizations. Overtime worked during the month was 3.3% of total available hours.

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

A. Shop Work		2 500 hours
B. Maintenance		8 200 hours
1. Preventive Maintenance	1 600 hours	
2. Emergency or Unscheduled Maintenance	1 600 hours	
3. Normal Scheduled Maintenance	5 000 hours	
4. Overtime	585 hours	
C. R&D Assistance		6 200 hours

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Heavy water physical inventory at the end of January indicated a loss of 3241 pounds valued at $44,860. Scrap generated during the month amounted to 3408 pounds resulting in a charge to operations of $2897.

Fifty-seven Ph. D. applicants were invited to visit HAPO for employment interviews. Five visits were made; two offers extended; two acceptances and one rejection were received. Three offers are currently open.

Nineteen Program and 11 direct placement offers were extended. Program activity included six acceptances and 10 rejections with 71 offers active at month's end. Direct placement results included five acceptances and one direct placement. Eight direct placement offers remain open.

Three graduates were added to the roll while five present members were placed on permanent assignment. Current program strength is 50.

Information requested by the AEC for its annual research report was assembled. Research performed for the Divisions of Reactor Development, Research, and Biology and Medicine was included.

Manager, Hanford Laboratories

HM Parker:JEB:dph

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REACTOR AND FUELS LABORATORY MONTHLY REPORT

JANUARY 1963

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 02 PROGRAM

1. Metallic Fuel Development

Fuel Irradiations. The irradiation of fluted single-tube N-Reactor fuel elements was initiated in the ETR. Two 10.5-inch long test elements were charged in the M-3 high pressure loop facility and are operating at full reactor power.

Three 18-inch long fluted N-inner test elements have received an exposure of 786 MWD/T in cold water irradiation at Hanford. In the high temperature irradiation of another such element in the ETR-7 facility four cycles of reactor operation have been completed satisfactorily to give an equivalent exposure.

Additional single-tube KSE-5 test elements have been completed and measured. These elements will be irradiated in KER loops 1 and 2 to study high temperature swelling performance. N-inner fuel components, with and without 2 w/o zirconium addition, are being fabricated for in-reactor testing to establish comparative irradiation performance of the two fuel compositions.

Examination of Irradiated N-Fuel Elements. The post-irradiation evaluation of 14 NAE's (28 components) which were successfully irradiated at near prototypic N-Reactor conditions to approximately 1700 MWD/T is continuing. Study of the pre- and post-irradiation dimensional data has revealed the following dimensional behavior.

	Inner Component (NIE)	Outer Component (NOE)
OD Increase		
Range	.003 to .007"	.000 to .004"
Ave.	.005"	.002"
Max. Calc. Clad Strain	0.56%	0.16%
Residual Warp-Single Throw		
Range	.005 to .025"	.007 to .029"
Ave.	.0135"	.0115"

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	Inner Component (NIE)	Outer Component (NOE)
Warp Change-Single Throw		
Range	-.007 to +.015"	-.001 to +.016"
Ave.	+.007"	+.006"

The planes of maximum warp in both sets of components were observed to shift in a random fashion. Lengths of the components are being remeasured at the KE basin facility because of an erratic pattern in the data.

One outer component (NOE) has been sectioned and examined in Radiometallurgy. No unusual features or shortcomings have been observed. The eutectic Be-Zr brazed closures have shown no indications of unsatisfactory behavior.

Cladding Deformation Studies. Thirty-six NaK capsules, including four which have thermocoupled uranium rods, were charged in four process tubes in F-Reactor. All aspects of the charging were satisfactory. The four thermocouples are recording the center uranium temperatures satisfactorily, and excellent agreement exists between the measured and calculated temperatures. Preliminary calculations indicate that the Zr-2 cladding temperatures are very close to those desired. The fuel samples in these capsules will provide data on the effects of cladding thickness uniformity, temperature, and exposure on the strain capabilities of the Zr-2 cladding.

Hot Headed Closure Studies. A die design and an element end configuration were developed for N-inner hot heading which produce thicker Zircaloy-2 ID shoulders with less tendency to fold. This modification permits machining of the shoulders square without seriously thinning the Zircaloy-2. Initial attempts to thicken the ID shoulders resulted in folds through the shoulders. A crevice resulted between the closure cap and the inner shoulders on previously headed elements due to the thin rounded shoulders.

Several short (4" long) N-inner elements are being headed and machined for subsequent welding evaluation.

Self-Brazed Closures. Three elements having closures installed by the "self-brazing" process were thermally cycled ten times between room temperature and 300 C. Ultrasonic tests indicated no discernible change in bond quality as a result of the cycling.

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Similar results were obtained in tests applied to specimens thermally cycled 4 to 10 times according to standard beta-heat treating procedures. These specimens are being sectioned for metallographic verification of the non-destructive test results. Another three specimens, one of which had been faced off on one end to lay bare the annulus of braze material between cap and clad, were placed in a long-term high temperature loop in K-East Area. After two weeks of exposure, no evidence of corrosion was evident indicating that the Cu-Ni braze material is reasonably resistant to corrosive attack.

Twelve 6-inch long test elements with self-brazed closures have been made, six from unheat-treated stock and six from already heat-treated stock. These specimens are intended for comparison of behavior throughout processing, testing, and irradiation.

N-Reactor Fuel Support Development. In the investigation of sheet fabrication procedures, extrusions in the form of $3\frac{1}{2}$ inch x $\frac{1}{2}$ inch bar were made and evaluated as starting material. The 0.035-inch sheet produced from this material was found to have good bend ductility in the longitudinal direction and exceptional ductility in the transverse direction. A fabrication schedule based on the use of extruded sheet bar was selected for the production of material for supports.

In the program to develop a production forming method for N-inner fuel supports, the primary problem of forming the parts without cracks in the loops appears to have been completely solved. In making about 2000 supports, no cracking has been observed, although several minor dimensional discrepancies have restricted the yield of acceptable parts. The necessary process and equipment changes are being made to improve the uniformity of the product.

Dies and fixtures for producing N-outer fuel supports of the original "suitcase handle" configuration are 90% complete. These supports of the shape originally proposed for the outer fuel element are to be used until the arch-type design proposed by NRD is proofed and in production.

N-Reactor Process Tube Scratching Studies. Laboratory tests to study possible damage to the N-Reactor process tubes by charging of N-fuel elements are proceeding. To determine the effect on scratch depth of repeated corrosion of the bare Zr-2 exposed after fuel element charging, steel support shoes were passed over a given section of process tube a total of 40 times. The autoclave film was broken through within three passes, after which scratching continued through to the end of the test. Depth of the scratch varied from

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0.0001 to 0.0008 inch at various locations along the tube with no particular pattern to the variations. The process tube section was then autoclaved in 400 C, 1500 psi steam for 14 days to simulate the amount of corrosion the scratched region of the process tube would experience in one charge-discharge cycle. An additional 40 passes were then made on the same damaged area over which the first test was run. On measurement the scratch depth again varied from 0.0001 to 0.0008 inch as in the first test. Additional tests are being run to determine the relationship of additional corrosion and charging cycles to the scratch depth.

Fuel Element Straightening. The semi-automatic handling equipment fabricated for the three roll straightener has been received, and installation is approximately 50% complete. Hot straightening of NIE fuel elements in the three roll straightener will be resumed immediately upon completion of this installation.

Brazing Alloy Development. A program to evaluate experimental brazing alloys is under way. Equipment necessary for this evaluation program has been designed and is being built. Because of the brittle nature of most of these alloys, physical testing will be done on investment cast specimens. To do this, a tensile and impact bar wax pattern mold has been made and the first investment mold prepared. A vacuum-induction melting point determination unit has been built and the vacuum system is being debugged. This unit will allow determination of the melting point of high melting temperature braze alloys at temperatures up to 1500 C. Temperatures will be determined by the thermal arrest method using both optical two-color pyrometers and thermocouples. The small specimen autoclaves are installed and currently under test. Hopefully, the first specimens will be placed in the autoclave during February. A high temperature heating furnace, making use of the magnetic induction concentration principle, has been built and will be started as soon as the vacuum system is completed. This furnace will be capable of reaching temperatures of 3600 C under proper conditions. Highly localized heating which results from the concentrated magnetic field may allow an improvement in the brazing process, perhaps even using some of the very high melting brazing alloys without affecting the uranium core beneath the cap.

Brazing studies have shown that copper can form an excellent diffusion bond between two Zircaloy-2 surfaces and between Zircaloy and uranium. In order to obtain more information about the mechanical properties and the corrosion properties of the Zircaloy plus copper alloys, a series of these alloys ranging from ½ w/o copper to 80 w/o copper, remainder Zircaloy-2, are being formed

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by arc melting. Also being investigated are possible brazing alloys in the Zr-Ti-Cu system. Melting points, corrosion rates, impact strengths, and tensile strengths will be used for comparisons of all alloys.

Cladding Alloy Development. In an attempt to improve the inner cladding quality of the coextruded fuel, a stiffer alloy, with the composition Zr + 3Nb + 1Sn, has been chosen for evaluation purposes. This alloy has approximately the same corrosion and nuclear properties as Zircaloy-2, but is nearly twice as strong in the annealed condition. In the heat treated condition, that is, when heated above the eutectoid decomposition temperature, the alloy becomes considerably harder due to the high proportion of β_{Zr} which holds both the niobium and tin in solution. Observed hardnesses are:

Condition	Brinnel Hardness
As extruded	22.1 Kg/mm^2
580 C 2 hrs (Below eutectoid)	20.6 "
750 C 2 hrs	22.7 "
850 C 2 hrs	27.9 "

Thus, to obtain the hardest structure possible prior to extrusion, the specimens should be heat treated at 850 C. In addition to the coextruded fuel tubes several rod coextrusions are planned to evaluate the compatibility of this cladding alloy with the uranium core in heat treatment and closure studies.

Supporting Fabrication. The Zircaloy clad thermocouple to Zircaloy flange joint has been successfully extended to include projection welded flanges. By using projection instead of spot welds, thick flanges the shape of end caps can be attached. Two pieces with simulated end caps are undergoing autoclave testing.

Another approach used was the replacement of projections by a titanium ring. Six titanium welds (or braze joints) were tried. A section of one is being autoclaved to determine corrosion behavior in the joint. Sections of the others have shown varying degress of bonding.

KVNS Self-Support. Proof parts of the K-5 rail have been provided for IPD. However, IPD has suggested that the base of the Tee section in the support crown be extended to touch the fuel cladding. This represents a modification of the original specification which required 0.030 inch collapsibility at this point. The die modification is in progress to provide the changes proposed by them.

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Dispersion Strengthened Fuel Materials. The co-precipitation method of producing sub-micron UO_2 particles dispersed uniformly in a metal matrix was extended to include trials of molybdenum and tungsten as the matrix. It was found that neither of these metals precipitated at the same pH as the uranium with the method tried. This resulted in a powder in which the UO_2 was not uniformly distributed. One hot compact (1200 C) was made from the 10% UO_2-Mo powder resulting in a 95% theoretical density. The distribution of the UO_2 was non-uniform to the same extent noted in the powder. Cold compacts (40 tsi pressure) of the 10% UO_2-Mo powder were hydrogen sintered at 1600 C to 75% theoretical density. A reasonably uniform dispersion of 0.5 micron particles was obtained. The 10% UO_2-W powder was cold compacted and sintered at 1600 C. This temperature is too low for tungsten and resulted in little or no sintering.

Investigation of NPR Fuel Element Rupture. An NPR outer fuel element ruptured in the autoclave after approximately 19 hours in 400 C - 1500 psi steam. Initial findings indicated that the rupture in the cladding was associated with one of the spot welds attaching the licking clip to the inside cladding. The failed section is in a form of a blister with the spot weld at the apex. The spot weld is also cracked through the cladding. This type of blister is similar to those obtained during ruptures of purposely defected fuel elements. These ruptures would form an initial blister on the Zircaloy cladding with the defected hole at the apex.

The assumption, by analogy with rupture experiments, that the blister at the clip weld indicates the point of initial failure, has been further substantiated by metallography. Cross sections taken through the spot weld nugget and Zircaloy cladding have revealed that the blistered section of the Zircaloy cladding, including the weld nugget and locking clip, contained much less hydride than portions of adjacent cladding.

This is the distribution of hydride noted in fuel element rupture experiments. The low hydrogen at the initial defect can be rationalized by the following line of argument. In the immediate vicinity of the defect, water vapor inhibits hydriding of the cladding by hydrogen generated from the uranium water reaction. The water vapor is rapidly consumed by the reactive uranium so that at distances of the order of half a centimeter, only hydrogen is present at the uranium-zirconium interface. Catastrophic zirconium hydriding and cladding disintegration results.

The actual cause of the failed spot weld is not yet known. Successive metallographic sections through the weld area are being taken for study. To date the following abnormalities have been detected: (1) a weld penetration approaching 80 to 90% of the Zircaloy cladding; (2) a 30% offset between the weld nugget and the spot weld dimple on the locking clip; and (3) heavy oxide formation in the cracks and in the crevices between the locking clip and the cladding which could have been caused by fluoride contamination.

2. Corrosion and Water Quality Studies

Corrosion Rates in Low pH Water. Measurements of carbon steel corrosion in cold water systems are being obtained from corrosometer probes and coupon samples. Single coupon measurements, after only 3-1/3 months of exposure, indicated 4.2 mils/year in pH 7.0 water in D plant and 6.0 mils/year in pH 6.6 water at KE. The measurements from the corrosometer probes do not agree very closely and may be related to the manner of probe placement. Probes are scheduled to be inserted directly into the bulk effluent lines of reactors and in the outlet nozzles of the once-through (SP) tubes in KE to give more reliable measurements and more directly comparable data.

A half-plant test (PT 442A) is scheduled to start next month at D-Reactor to compare 6.6 pH and 7.0 pH (with dichromate concentration of 1.8 ppm). Four coupon holders, with 48 aluminum and 8 carbon steel samples, are being prepared to monitor this test.

Nucleate Boiling on Zircaloy Samples. Operation of the TF-3 loop to evaluate the possible Zr-2 corrosion problems which could occur in the crevice formed between an NPR support and the heater rod cladding was continued. Metallurgical examination was completed on a Zr-2 clad heater rod with NPR supports exposed to lithiated water at pH 10.0, with a bulk temperature of 628 F, a pressure of 2000 psi, and a heat flux of 280,000 Btu/ ft^2 hr. In crevice regions where nucleate boiling concentrated the LiOH, corrosion greater than 12 mils had occurred, resulting in complete perforation of the Zr-2 cladding. In most of the areas of high corrosion, large quantities of corrosion product (ZrO_2) were present, firmly attached to the metal and compact in appearance although some cracks in the oxide were apparent. No LiOH concentration was found on the heater rod. Hydride concentrations up to 75-100 ppm were found in the worst areas; however, no evidence of any preferred orientation was observed. Normal corrosion was found on the heater rod in areas free of crevices.

In a second test under similar conditions except for a bulk temperature of 598 F, a thermocouple was placed underneath the support rail to monitor the cladding temperature. A continuously increasing temperature was noted. The initial temperature was 709 F and had increased to 733 F in 16 days. Following a loop shutdown, the temperature dropped to 720 F and then increased to 744 F after 35 days. The rate of rise appeared to be undiminished and either severe crud deposition or corrosion was occurring, which in turn indicated that nucleate boiling was probably occurring.

Reactor Decontamination After Rupture. Laboratory tests have verified that the peroxide-oxalic-peracetic and peroxide-oxalic-gluconic solutions are effective oxidants and dissolution agents for uranium oxides over the temperature range of 25 C to 80 C. Loop tests are scheduled to evaluate the efficiency of these solutions at room temperature. Low temperature procedures would simplify reactor decontamination and reduce corrosion problems.

Decontamination of ETR G7-M3 Loop. In November 1962, the G7-M3 Loop in the ETR was contaminated by a metallic fuel element rupture to the point that decontamination was required to reduce activity levels. Prior to this failure, the main portion of the G7-M3 Loop system had been operating since August of 1961. Some portions had been in operation since 1959 as the single G7 system. The G7 system experienced two uranium oxide element ruptures during its operation before modification and integration with the M3 system. After startup, the G7-M3 system operated for nine months before the system became active from a fuel element with suspected uranium-contaminated welds.

Loop decontamination was attempted in January 1963. The system was cleaned with a 10% HNO_3 solution to dissolve rupture debris, followed by the APACE procedure using a 3% citrate concentration. A citrate sample taken before draining read 30 mr/hr. The system activity levels were still high so a second APACE was used with a 10% citrate concentration. A citrate sample taken before draining read 17 mr/hr. The system activity level was still approximately 400 mr/hr after draining.

A section of the loop piping has been cut out for detailed study at HAPO. The activity reading on the piece is 600 mr/hr with smearable activity of 1500 mrad/hr, including 5 mr/hr. The reason for the inadequate decontamination from standard decontamination techniques is unknown.

Descaling and Derusting. Procedures are being evaluated for derusting NPR should rusting occur in the piping systems during the lay-up period following chemical removal of mill scale. Initial tests were made in the TF-14 loop which was derusted using 1 lb/gal Turco 4507 at 200 F for 1/2 hour. This procedure adequately derusted the loop and prevented flash rust from re-forming. However, after one week, copious quantities of rust were found in the loop. A similar test was planned for the K-1 loop, but because of the TF-14 experience and the anticipated difficulty in rinsing the solution from K-1, an ammoniated citric acid procedure has been recommended for the second step in the K-1 derusting test.

Removal of Vapor Phase Inhibitor. When the welds on the NPR front face risers were annealed, a considerable amount of the powdered vapor phase rust inhibitor present melted and solidified. Laboratory tests indicated that several organic solvents or 350 F water dissolves the solids. Strong acids or bases had no effect. Since it is both expensive and potentially dangerous to fill these large volume pipes with organic solutions, the high-temperature water method of removal is being further evaluated. A test section will be constructed to run prototypical tests in TF-7.

Evaluation of NH_4OH in K-1. Tests are planned for the K-1 loop in which loop pH control will be changed from LiOH to NH_4OH to simulate a similar changeover proposed for N-Reactor. In preparation for the test K-1 loop was descaled to remove the rust and mill scale present in the piping from recent modifications. Following the descaling, deionized water with the pH adjusted to 10 using lithium hydroxide was recirculated in the ex-reactor piping for approximately four days to form a protective magnetite film. The water temperature was relatively low ($<$ 200 C) as the only heat source was from the pumps. Lithium hydroxide injection was stopped and ammonium hydroxide injection began. Ammonium hydroxide injection was continued for approximately 14 days. During the entire test, including the lithium hydroxide addition period, high levels of crud were found in the loop.

Corrosion samples of A245 carbon steel were placed in the loop to monitor periods of lithium hydroxide pH control, the transition period, and ammonium hydroxide pH control, and the entire period of the test. All samples had a very adherent, black film covered with a very thin, loose black film. These film characteristics are typical of numerous previous tests. Corrosion penetrations were low on the samples exposed in the lithiated water and on the samples exposed in the ammoniated water.

The ammonia concentration was monitored continuously during this test, and it was observed that equilibrium conditions were easier to maintain using the indicated ammonia concentration rather than pH as the control variable. Furthermore, it was possible to maintain the pH in the 9.9-10.1 range over a relatively wide range of ammonia concentrations - 6 to 11 parts per million ammonia.

The cause of the magnetite (crud) in the water has not been definitely determined. However, since it was experienced during the preliminary operation with lithium hydroxide, it does not appear attributable to the use of ammonium hydroxide.

The crud problem abated somewhat during the latter portion of this test, indicating that continued operation or the use of ammonium hydroxide, or both, had a beneficial effect. Since the results obtained during this test did not indicate any difficulties in the use of ammonium hydroxide for pH control in this system, a similar test is scheduled for the entire loop, with fuel elements present, as soon as K-1 resumes operation.

Thermocouple Elements for K-1 Operation. During the changeover in K-1 loop from pH control by LiOH to NH_4OH, it is planned to monitor fuel element crudding with thermocouple slugs. Experience to date with Zircaloy-clad thermocouples in Zircaloy-clad fuel elements show them to be so prone to rupture that use in-reactor is precluded. Alternate designs are being developed to avoid brazing thermocouples into the fuel element end caps. At present five Zr-2 capsules which use mechanical seals to accomplish the thermocouple penetration are being thermocycled from 225-500 F in TF-7. These include: (1) Zr-2 Swagelok, (2) Navy Seal S/S, (3) Navy Seal Zr-2, (4) English Rivet (large), (5) English Rivet (small). The Zr-2 capsules contain nickel chloride ($NiCl_2$) to determine if a leak small enough not to be detectable by weight gain has occurred. The $NiCl_2$ will change from yellow to green when it becomes hydrated.

Instrumentation for NPR. Evaluation of an electrode system for sodium ion concentration measurements in the 0-1 part per million range was resumed. A continuous single-pass system was constructed to perform calibration tests with solutions having known sodium concentrations. Hydrogen ion interference was eliminated by adding ammonia to elevate the pH of the sodium chloride samples (in deionized water) to 10.5-11.0. Solutions were prepared and stored under an argon gas "blanket" to prevent contamination of the samples by contact with the atmosphere. Many process variables were investigated during these tests, and it was concluded that either:

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(1) it is impossible to prepare and preserve accurate standards in the concentration range below 200 parts per billion sodium, or (2) the electrodes do not perform satisfactorily in this low concentration range.

Installation of a laboratory system to evaluate the Hays oxygen analyzer in the presence of hydrazine was completed. This work is necessary to determine the analyzer characteristics anticipated during use in the NPR secondary coolant system. The Hays analyzer was operated to determine the performance of the deionization and deoxygenation system installed for treatment of the feedwater. The pretreatment system performs exceptionally well - effluent specific resistance is approximately 12-14 x 10^6 ohm-centimeters and dissolved oxygen concentration is approximately 10 parts per billion.

A continuous chemical analyzer is installed in series with the oxygen analyzer to measure changes in the hydrazine concentration. At present work is in progress to modify the hydrazine analytical procedure to permit analyses in the parts per million range. (The present procedure is only useful in the parts per billion range.) Operation of this system for simultaneous oxygen and hydrazine analyses will be initiated as soon as the hydrazine procedure is satisfactory.

3. Gas Atmosphere Studies

Water Formation Over Graphite. The rate of the reverse water gas reaction, $H_2 + CO \longrightarrow C + H_2O$, is being measured to determine to what extent this source will provide water to oxidize Zircaloy process tubes penetrating a hot graphite moderator stack. Dry mixtures of CO and H_2 in helium are passed over two graphite bars in series and the increase in water concentration measured. In the initial runs at 800 C, the H_2 and CO combined rapidly to form an equilibrium concentration of water. However, in two runs at 700 C the reaction was far from equilibrium. These results indicate there is a strong temperature dependence for the reaction in the temperature range of interest.

Effect of CO_2 as an Inhibitor of Zircaloy Hydriding. The effectiveness of CO_2 as a hydriding inhibitor is being investigated by passing dry mixtures of 1% CO_2 and 2% H_2, in helium, over Zircaloy-2 coupons at 375 C and 425 C (no graphite present to catalyze H_2O formation). Results from a 7-day run indicate that Zircaloy which was sensitized by vapor blasting absorbed excessive amounts of hydrogen; however, preautoclaved and freshly etched samples did not

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hydride. In the past, hydriding of the vapor blasted samples was a reliable guide to the ultimate long-term performance of preautoclaved samples which usually show temporary protection, but fail after a variable induction time. This test is continuing.

Graphite Burnout Monitoring. Burnout-monitoring samples in channel 3478 at D-Reactor from October 22, 1962 to January 3, 1963 were measured. A profile of the rates showed a sharp peak of 15% per 1000 operating days (KOD) at approximately 80 inches into the graphite stack. A lower peak of 4%/KOD appeared at approximately stack center line, and very slight weight gains occurred on the monitors near 310 inches. The location of these two peaks was the same as for the previous test from July 10 to October 22, 1962; however, the height of the front peak was reduced by a factor of three during the latter period.

Graphite-Zirconium Compatibility in the NPR. In order to prevent hydriding of the zirconium process tubes in the NPR, it is necessary to maintain a protective, oxidizing, atmosphere around the tube. The design of the reactor is such that this protection must be achieved by diffusion of water vapor from the gas channels into the tube-block annulus.

Mathematical equations are being developed to simulate the NPR atmosphere problem on an analog computer. The program will update the approach used by G. E. Zima (HW-69870, Rev). The program developed will include a more realistic effect for hydrogen inhibition on the water vapor-graphite reaction and will also include the effect of carbon monoxide on the reaction scheme. The reactions included in the analysis are given below:

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

$$CO + H_2O \rightleftarrows CO_2 + H_2$$

$$C + CO_2 \longrightarrow 2CO$$

$$Zr + 2H_2O \longrightarrow ZrO_2 + 2H_2$$

4. Process Tube Development

Stress Rupture of Reactor Pressure Tubes. The extensometer under development to remotely measure circumferential strain of tube specimens in elevated temperature stress rupture test is still

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operating satisfactorily. The past 21 days of testing have shown the electro micrometer reading to be 0.0005 inch different than the mechanical reading.

A remote fired gun system to shoot shaped projectiles into unirradiated pressurized tube specimens for brittle fracture studies was developed, tested, and is now being used in a test program.

Brittle Fracture Resistance of Reactor Pressure Tubes. The test for susceptibility of tubing to brittle fracture was developed using process tubing of KER dimensions. The best of this tubing tested does not fail in a brittle manner when a crack is introduced at room temperature and at pressures below 75% of ultimate. Cracks do propagate when the test is conducted at 80% of ultimate pressure. Samples of the same tubing when tested at 150 C and 300 C resisted the propagation of cracks at 90% of ultimate pressure, the highest pressure that can feasibly be employed in this test.

Tests have just begun on samples of 30% cold worked NPR process tubing. Room temperature tests have been completed and the performance of this material is nearly identical to that of the 50% cold worked KER tubing described above. The crack introduced by the test procedure does not propagate at 75% of ultimate pressure but does at 85%. If the similarity between the two materials holds at higher temperatures, the 30% cold worked NPR tubing will prove resistant to crack propagation at 150 C and 300 C under the highest pressures attainable in the test. These pressures are approximately seven times the maximum anticipated under reactor operation conditions.

5. Thermal Hydraulic Studies

NPR Boiling Burnout Studies. A detailed examination was completed of the boiling burnout data obtained with an annular test section where the two surfaces forming the annulus were both heated. In addition, this particular test section was constructed to represent the case where the supports which position the inner tube of the fuel element within the outer tube were sheared off and leaving only the weld tabs to separate the two fuel pieces. This resulted in an 80% decrease of the annulus thickness along the bottom of the two fuel pieces. During the experiments, the two fuel pieces were represented by electrically heated tubes which were instrumented with thermocouples to detect the temperature increases associated with boiling burnout.

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Results of these experiments were compared with the results of burnout experiments using a similar, concentric, test section. For mass velocities in the range of 1×10^6 to 4×10^6 Btu/hr-ft^2, the comparison showed the burnout heat flux for the eccentric test section to be 50 to 55% of those for the concentric test section, for the same bulk coolant enthalpy. (This is slightly higher than the 40-50% indicated by the preliminary results reported earlier.) Bulk coolant conditions for these experiments ranged from about 125 F subcooled to steam qualities of 5 to 10%.

Results of these experiments were also examined for application to NPR operation. The calculations indicate that, with the inner tube of a fuel element 80% eccentric in a maximum power central zone tube, flow through the inner flow annulus could be reduced by more than one-third before burnout would occur. This conclusion is valid only for the case of eccentricity in a single fuel element. If the same degree of eccentricity existed in two or more adjacent fuel elements, giving an eccentric section length greater than that of the test section, the flow and enthalpy distributions around the annulus might be different than those in the test section. It is impossible to predict the burnout heat fluxes for such a case from the experimental results obtained.

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

Inspection of PRTR Fuel Elements. The UO_2-PuO_2 fuel element which has attained the highest exposure in the PRTR (approximately 700 MWD/T_U) was examined in the Fuel Element Examination Facility. The contact points on the bottom end bracket of this element previously caused excessive process tube wear. Extended surface, clip-on wear pads will be attached to the end brackets before it is reinserted into the reactor. The examination revealed: (1) severely worn contact points on the bottom end bracket, (2) very little wear of the contact points on the top end bracket, (3) a corroded wire wrap weld area on the bottom of one rod (weld zone completely covered with white corrosion product), (4) a limited amount of white corrosion product on wire wrap welds on the top ends of some rods, (5) three damaged circumferential strip bands, apparently caused by charging of the element (the resulting upward displacement of the bands uncovered discolored spots on the surface of the fuel rods at the band contact points), (6) satisfactory fuel rod surfaces and no discoloration, and (7) no loose wires or worn spots on the visible wire wrap surfaces.

The in- and out-of-reactor operating performance of the extended surface, clip-on wear pads is encouraging. Three 19-rod cluster, UO_2-PuO_2 fuel elements operated for approximately 27 days in the PRTR without any measurable wear of the extended surface ($\frac{1}{4}$-inch wide by $\frac{1}{2}$-inch long) wear pads. These pads were remotely attached to the elements in the PRTR basin.

A 19-rod cluster element with clip-on pads installed was subjected to PRTR coolant temperature and flow conditions for 74 days in the TF-7 (out-of-reactor) test loop. No measurable wear occurred on the pads.

Pads were applied remotely to six additional irradiated UO_2-PuO_2 PRTR fuel elements.

Fuel Cladding Interaction in Swaged UO_2 Fuel Rods. A limited reaction layer of uniform thickness (approximately 0.0005-inch thick) was observed on the inner surface of the Zircaloy cladding of an irradiated, incrementally loaded and swage compacted UO_2-PuO_2 PRTR fuel rod. During post-irradiation study of the effect of PuO_2 segregation the reaction layer was observed in a 160-

increment rod, but not in an 80-increment rod, even though the constituents of the two rods were thought to be identical. The inner surface of the cladding with the reaction layer appears black, whereas the inner surface of the cladding without the reaction layer retains its original metallic luster. Efforts will be made to identify the reaction layer.

PRTR Fuel Element Fabrication. Nine, swage compacted, mixed oxide, PRTR fuel assemblies were completed.

Impaction of UO_2-PuO_2. Impaction of UO_2 powder containing 2.5 w/o PuO_2 was successfully demonstrated. Density achieved on the first PuO_2-UO_2 compaction was comparable to that obtained with UO_2 under similar conditions. Density of the PuO_2-UO_2 was 10.45 gm/cc, or 95% of the theoretical. The compaction pressure was 145,000 psi. Tooling has been changed to permit higher impact pressures, and probably higher densities.

Conceptual design was started of facilities for rapid processing of impacted material. These facilities will include blending and canning of powders before impaction and decanning, crushing, and sizing of the material after impaction. As part of this study, a standard pipe cutting tool was used to open the impacted can of material in a fraction of the time required by previous methods.

Fuel Element Rejuvenation. The four-rod cluster rejuvenation fuel element successfully completed one irradiation cycle in the MTR at the Idaho Testing Station.

Plutonium Fuels Testing and Evaluation Laboratories. Detail design is 70% complete, with a scheduled completion date of February 15, 1963. Design is slightly ahead of schedule, and the completion date should be met without difficulty. Specifications for the metallograph were completed.

Cladding Studies. Three self-standing cladding assemblies, each 9 feet long, were fabricated off-site by high voltage electron beam welding. The fuel cladding is supported by ribs attached to an outer larger diameter fuel cladding. Comparison of the collapsing behavior of a cladding tube 2.3 inches in diameter with this self-standing cladding was made by autoclaving one specimen of each type. Preliminary indications are that the self-standing cladding is far superior to plain tubular cladding for withstanding external pressure.

A one-foot-long sample of arch-supported fuel element cladding was also fabricated by the high voltage electron beam welding process. This new cladding concept uses a five-sided, concave star configuration to enclose the fuel. The maximum diameter is three inches. This star-shaped cladding is inserted into a tube and a weld made through the outer cladding and along the seam at each point of the star. This configuration has the advantage of greater resistance to both internal and external pressures.

Irradiation of UO_2-PuO_2 Mixtures. Sharply defined bands, about 0.040-inch wide, that are more highly radioactive than the adjacent fuel material, were found in irradiated high density $(U,Pu)O_2$ pellets from three capsules (GEH-14-91,-85, and -86). These sintered fuel pellets contain 0.0259, 2.57, and 4.13 mole percent PuO_2, respectively. The bands, only faintly visible on earlier glass autoradiographs, are clearly revealed by beta-sensitive, high resolution film plates. The fuel pellet from GEH-14-86, which operated during the last of several irradiation periods at twice the rod power of the other two specimens, exhibits a band ID of 0.25 inch, approximately double that of the other two fuel pellets. This may be related to increased relative effectiveness of thermal diffusion transport of fission products after the gross fuel structural changes are completed.

The improved autoradiography technique is being applied to many of the remaining 21 test capsules of the series.

Irradiation Performance of MgO-PuO_2 and ZrO_2-PuO_2 Fuels. Microstructural features different from those ordinarily observed in many other irradiated ceramic fuels were observed in irradiated MgO-PuO_2 and ZrO_2-PuO_2 sintered pellets. High resolution autoradiographs of transverse cross-sections of the irradiated specimens reveal the location of fission products. Limited fission product migration occurs in the immiscible MgO-PuO_2 type fuel material in spite of extensive recrystallization and columnar grain growth, whereas the fission products migrate to preferred locations in recrystallized ZrO_2-PuO_2 solid-solution fuel material. Micro-analyses of the samples will be conducted to determine the fission product species which have migrated and the extent of their migration. Since no apparent fission product migration occurred in the immiscible MgO-PuO_2 fuel, it is possible that what is observed in ZrO_2-PuO_2 fuel reflects migration of plutonium atoms in the solid-solution.

Fast Reactor-Thermal Reactor Exchange Element. Improved fuel utilization may be achievable by alternately irradiating high durability uranium-plutonium fuel elements in a fast reactor blanket, and a thermal reactor core. Such a fuel element, to contain depleted UO_2 initially, was designed for a fuel element exchanging study utilizing the Fermi Reactor and the PRTR. Components are being assembled for a dummy element to be used in the thermal hydraulics studies.

Phoenix Fuel Experiment. The reactivities of three irradiated Al-Pu specimens containing plutonium which initially had 6.25, 16.33, or 27.17% Pu-240 were measured in the ARMF.

Two additional specimens, one containing no plutonium, were fabricated for tests designed to show the effect of an aluminum-plutonium Phoenix fuel on the ARMF neutron energy spectrum. Details of this calibration test have been established and the experiment will now be conducted.

Corrosion and Water Quality Studies

Oxidation and Deuterium Absorption in D_2O. Oxidation rates and deuteron absorption rates for Zircaloy-2, Zircaloy-4, and crystal bar zirconium were measured in 400 C, 1500 psi static deoxygenated D_2O (99% purity) for a period of 140 days. The results showed no significant differences in oxidation rate between D_2O and H_2O corrosion rates; however, D_2 absorption rates were somewhat lower averaging 17% of the corrosion product hydrogen produced for Zr-4, 31% for Zr-2, and 11% for Zr compared with H_2 absorption rates of 27% for Zr-4 and 42% for Zr-2 in deoxygenated H_2O. This experiment has been terminated with new tests in 360 C and 280 C D_2O about to start.

Corrosion Cracking of Stainless Steel in PRTR. The effect of various ions on the speed of sodium hydroxide cracking of stainless steels is being examined. The effect of $KMnO_4$ and its reduction products was studied because an 18% NaOH - 3% $KMnO_4$ solution was used in the decontamination of PRTR. Although this solution causes no adverse corrosion of stainless steel at the temperature at which it is used, the effect at normal operating temperature of small residuals trapped in equipment was of interest. The samples were 3/8-inch flared tubing connectors made of type 316 stainless steel with a short flared piece of type 304 stainless steel tubing in place. The connectors were assembled in the contaminating solution, removed, wiped on the outside, tightened, dried at 110 C for two hours, exposed to

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275 C atmospheric pressure steam, disassembled and examined for cracks. Control samples were included with each batch exposed, one clean and one contaminated with pure sodium hydroxide solution. The sodium hydroxide samples cracked through the stressed portion of the nut in 3 to 20 hours at 275 C. Of six fittings contaminated with 18% NaOH - 3% $KMnO_4$ only four showed minor cracks in the flared tubing after 150 hours. One fitting contaminated with 18% NaOH - 0.1% $KMnO_4$ cracked in less than 27 hours as did another fitting contaminated with 18% NaOH - 3% $KMnO_4$ which had been reduced with H_2O_2 to MnO_2. Tests with other concentrations of permanganate and metallurgical sections of the cracks are being made.

Fretting Corrosion. The fretting corrosion test of the PRTR fuel element (without hanger) continued in TF-7. The fuel element being tested is a 19-rod wire wrapped bundle with clip-on supports having a wider contact area. A vibrator is attached to the test section and operated at different frequencies and impact force to evaluate the effects of these variables on the rate of attack. Both the frequency and the impact force of the vibrator are regulated by the air pressure. In the initial test previously reported, the test section was vibrated at about 80 cps (10 psi air pressure) for a period of $11\frac{1}{2}$ days. During this period significant fretting corrosion was detected, with a rod wire wrap penetrating the tube 3 to 4 mils and the wire wrap being eroded to about 50% at one location. A subsequent 24-day test was run in which the vibrator pressure was increased to 20 psi. Little additional attack was noted. A 15-day test at the original pressure setting disclosed minor fretting on the rod wire wraps and a slight color change occurred on the supports. During the two-month exposure period the large size clip-on support pads have produced negligible fretting penetration of the pressure tube.

A new PRTR Zircaloy-2 pressure tube section was inserted in the test section at the last discharge. The old tube had numerous fretted areas making it difficult to tell any new marks from the old ones.

Measurement of Zirconium Concentration in PRTR Primary Coolant Water. A program was initiated to monitor the zirconium concentration in the PRTR primary coolant using more sensitive emission spectroscopy procedures than were available previously. The purpose of this program is to investigate the feasibility of using this technique to detect the onset of fretting corrosion in the PRTR. Samples are analyzed daily or once each shift during periods when the coolant is recirculating at full flow to detect

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the abrupt concentration changes that would probably be obtained if severe fretting corrosion occurs. Data obtained to date indicate that the zirconium concentration normally encountered is in the 0-6 parts per billion range. Results have occasionally indicated significant concentration changes - sometimes as much as a ten-fold increase - for samples collected at 8-hour intervals. There are still insufficient data available at present to permit a realistic appraisal of whether or not changes of this order of magnitude conclusively demonstrate that accelerated corrosion is occurring. This program is being continued and the results will be evaluated in light of the information obtained from routine examinations of the PRTR fuel elements and pressure tubes to determine whether this analytical procedure is a satisfactory method of detecting fretting corrosion. Future work will involve comparison of concentrations in the effluent from selected tubes and the bulk coolant to determine whether there is a noticeable concentration difference in known fretting areas.

Concurrently, work is in progress to evaluate the feasibility of developing a continuous colorimetric procedure for zirconium concentration measurements in this concentration range.

Fuel Element Core Dissolution Tests in IRP. An additional series of decontamination tests was run in the IRP using sections of MgO-PuO_2 fuel element. The tests were designed to duplicate the high decontamination factors obtained previously using the oxalate-peroxide-peracetic solution, but with ferron added as a complexing agent.

The fuel piece was run at 300 C for four hours to dissolve the sample. The filter activity rose to 4.05 r/hr at the end of the run and decreased to 3.0 r/hr after draining the loop water. After adding the oxalic-peroxide-peracetic-ferron solution and running the solution at 20 C for 20 minutes, the activity decreased to 1.75 r/hr. The solution was then heated to 80 C and the filter activity rose to 2.15 r/hr. The rise in filter activity upon heating was unexpected and was decidedly different from the previous run, in which the filter activity during the dissolution step dropped from 2.8 r/hr to 0.7 r/hr. When the solution was then drained, the filter activity read 1.65 r/hr. A new solution of the same chemicals added and circulated to determine if depletion of one of the original chemicals had stopped the initial reaction. There was no decrease in filter activity during this second dissolution step.

After the dissolution step, the permanganate-hydroxide solution was used and was followed by an ammonium citrate solution. During the run with the permanganate-peroxide solution, the activity dropped a small amount; during the ammonium citrate run the activity dropped from 1.05 r/hr to 0.8 r/hr in 10 minutes, but then increased to 1.35 r/hr after an additional 40 minutes. Back-flushing the filter lowered the activity to 0.35 r/hr. The permanganate-peroxide solution followed by Turco 4518 had a small effect, lowering the activity to 0.25 r/hr.

Thermal Hydraulic Studies

Segregated PRTR UO_2-PuO_2 Fuel. Calculations were continued to determine the local surface heat fluxes and fuel temperatures in a PRTR mixed oxide fuel element in which the PuO_2 enrichment is segregated into narrow bands along the axis of the fuel rods as a result of the incremental loading process used to fabricate the fuel. Using the most recent heat generation information, the calculations indicated that peak to average heat flux ratios of 2.37 could occur for fuel rods loaded in 80 increments (1-inch total band length) if no mixing of the PuO_2 bearing powder occurs with the UO_2 powder. This would result in a maximum heat flux of 660,000 B/hr-sq ft for a fuel element operating at a tube power of 1200 KW. For a rod loaded in 160 increments ($\frac{1}{2}$-inch total band length) the peak heat flux ratio was calculated to be 1.35 (475,000 B/hr sq ft for a 1200 KW fuel element). A report summarizing the results of this calculation was written and issued as HW-76115.

Since various destructive and non-destructive tests of completed PRTR fuel rods have indicated the occurrence of radial segregation of the PuO_2-bearing powders in addition to the axial segregation considered in these calculations, preparations were made to consider this radial segregation when information regarding its extent and heat generation distributions becomes available. An existing computer code was modified to handle the three dimensional conduction configuration presented by a PRTR fuel rod with combined axial and radial segregation of the PuO_2.

Reactor Components Development

Second Generation Mechanical Shim Rod for PRTR. Detailed design on the second generation shim rods for PRTR was resumed with primary effort on the transmission parts. All gears, shafts, and bearings for the transmission have been received. The selsyn position readout system is scheduled for delivery in February.

EDEL-1 Renovation. X-raying of the field welds and hydrostatic testing of those parts of the EDEL-I piping system which have been changed will complete the modifications to the pressure piping system.

The rebuilt motor and variable speed drive for the EDEL-I have been checked for excessive vibration. The maximum vibration was 0.004" measured at the outboard motor bearing and exceeds the maximum allowable vibration for continuous operation by about 0.002". Apparent shipping damage could be the cause for the excessive unbalance now in the unit since maximum vibration at the factory was reported as 0.0017". The unit is being rebalanced to reduce this vibration, and necessary measures are under way to make the unit operable as soon as possible.

Fretting Corrosion Investigation. Work orders have been issued to Physical Measurements Operation to design, procure, and fabricate the instrumentation required to measure the relative movement due to vibration between the fuel element and PRTR pressure tube employing the eddy current technique.

Design of a prototypical PRTR jumper installation in the EDEL-I test pit has been initiated.

A rough draft document, HW-75951 RD, "Proposed Ex-Reactor Fretting Corrosion Study Program in EDEL-I," was issued for comment.

PRTR Rupture Loop Components. Leak tests of the Grayloc coupling for the PRTR Rupture Loop outlet connection has started. This testing is being done in EDEL-II at 2100 psig pressure and 600 F. The coupling is to undergo 50 thermal cycles during which time leakage rate and bolt stress data will be taken.

To facilitate installation of the coupling and mounting of the strain gauges, the temperature compensating strain gauge was mounted on a free bolt. During testing, however, it was found to be impossible to keep the compensating and stress bolt at the same temperature. Because of this problem, the temperature compensating strain gauge has been mounted on the stress bolt.

To date, two attempts have been made to conduct this test. Both times the test had to be stopped after one or two thermal cycles due to excessive leakage. After the second failure, dimensional checks of the hubs disclosed that the circular sealing face of each hub was out-of-round by 0.005". It was the vendor's opinion that 0.005" out-of-round is sufficient to prevent proper seating of the seal ring. This distortion may have resulted from welding the hub. A new coupling assembly has been fabricated for testing in which the circular sealing faces are round within less than 0.001".

Miscellaneous "cleanup" items associated with discharge tools and equipment are about 95% complete. The final report, HW-75643, "Development, Testing, and Preliminary Operating Instructions Fuel Element Rupture Testing Facility - Discharge Equipment," was issued.

Installation of the B-F ferrule seal in the inlet bellows assembly was initiated upon removal of the in-reactor section from the EDEL-I test facility. The modifications required to the bellows for the ferrule seal installation were minor in nature.

Properties of Irradiated PRTR Process Tubes. The investigation is continuing on tube 5679, removed from channel 1643 on May 2, 1962, because of a flaw observed on the inside surface of the tube. The burst properties of sections of this irradiated tube were reported previously and the flaw had no effect on the burst strength.

Metallographic sections through the flaw reveal it to be a lenticular nonmetallic inclusion, 0.020-inch thick by 0.1-inch wide. A thin layer of Zircaloy-2 separating the inclusion from the inside of the tube contains 300 ppm of hydrogen whereas the Zircaloy-2 on the other side of the inclusion contains the normal amount of hydride. An unusual concentration of intermetallics, apparently only in longitudinal metallographic sections through the tube, is being investigated further.

Investigation was started on tube 0720 to determine if the $MgO-PuO_2$ fuel element, which ruptured in this tube in August 1962, caused hydriding of the tube in the region of the rupture. A 6-inch long portion of this tube, adjacent to the rupture, was obtained and is now in Radiometallurgy. The piece will be cut into $\frac{1}{2}$-inch rings and metallography will then be used to determine if hydriding occurred.

Pressure Tube Monitoring. Twelve process tubes were inspected during the past month. Three of the 12 tubes (channels 1356, 1443, and 1948) operated for two nominal 10-day periods with the new large ($\frac{1}{4}$-inch by $\frac{1}{2}$-inch) fuel centering pads. These were inspected at the end of each of the two operating periods. The purpose of these inspections was to see if increasing the contact area, at the points where the fuel element end supports contact the process tube, would reduce fretting corrosion as it had in the

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ex-reactor loop TF-7. Slight fretting corrosion was revealed by these inspections, but the amount was probably less than small spacers would have caused. In all three tubes the wall penetrations were considerably less than one mil.

Two of the 12 tubes inspected had been re-orificed to increase coolant flow by about 6% to determine whether or not there was a measurable effect of the magnitude of fretting corrosion. These inspections showed that, to date, there has been no pronounced effect.

Inside diameter and gas gap measurements were obtained for all tubes. Data analyses are not yet complete; however, a preliminary check for channel 1348 shows that the minimum spacing between the pressure and shroud tubes is about 0.050 inch and has not changed since the last measurements obtained in March 1962.

Hazards Analysis

Plutonium Recycle Test Reactor. A study of cooling the PRTR following a total loss of electrical power has been completed. The report of this study will be published as document HW-76313. Design analyses of the PRTR indicated that the liquid-phase convection circulation of coolant should be adequate for cooling the fuel elements. The liquid-phase convection circulation rate was estimated to be 0.75 ft^3/sec compared to a required flow rate of 0.63 ft^3/sec. However, a low flow rate pressure drop test, performed on sections of the primary coolant loop and then summed, gave an estimate of 0.5 ft^3 sec flow rate under liquid-phase convection circulation conditions. Data obtained from the control panel flowmeter and an auxiliary differential pressure gauge confirmed the earlier tests. A mathematical analysis of the transition from liquid-phase convection circulation to boiling convection circulation shows that this would take place automatically if liquid-phase cooling were inadequate. The transition to cooling by boiling would incur the loss of about 35 cu ft of heavy water in accommodating the steam void formed in the primary coolant system, but the fuel elements would be adequately cooled provided they were kept covered by water. The liquid level in the pressure tubes could be maintained by intermittent injection of light water into the primary coolant system.

Supplement 6 to the PRTR Final Safeguards Analysis has been written. This supplement presents justification for increasing the maximum point heat transfer flux from 400,000 Btu/(hr)(ft^2) to 600,000 Btu/(hr)(ft^2) and the maximum tube power from 1200 to 1800 Kw. Proposed limits for fuel element types that may be charged into PRTR are also presented. Kinetics analyses of typical reactor accidents for uniformly-enriched mixed oxide fuel loadings are given and compared with similar analyses in the original safeguards analyses for spike-enriched loadings. No significant increase in the magnitude of the effects of a reactor accident with the new fuel loadings were found.

Two process specifications were revised.

<u>Plutonium Recycle Critical Facility</u>. A comment draft of a supplement to the Plutonium Recycle Critical Facility Final Safeguards Analysis, HW-69168, was issued. This supplement was prepared to present additional information requested by RLOO, AEC. Kinetics studies of typical reactor accidents were made to characterize the behavior of the different core loadings proposed for D_2O moderated experiments in the PRCF. These studies indicated that all of the core loadings can be operated as safely as the zoned loadings described in the final safeguards analysis.

Eighteen of the 19 process specifications needed for PRCF startup tests have been approved.

[redacted]

2. Plutonium Ceramic Fuels Research [redacted]

<u>Determination of Carbon in Plutonium Carbide</u>. In cooperation with Analytical Laboratories, a combustion gravimetric method for determination of carbon in plutonium carbides was reduced to routine practice in glove boxes. The method was found to have better than 1% precision for 1 to 5 w/o carbon in specimens of 200 to 1000 mgs.

<u>Plutonium Carbide Studies</u>. Analysis of X-ray powder patterns of a 48.2 a/o C alloy indicate the possible growing-in of the zeta phase of the Pu-C system while standing at room temperature for a period of 400 days. The X-ray powder patterns were taken at approximately 4-month intervals and were performed on a double-sealed glass capillary sample of the as-cast alloy.

<u>PuO_2-Carbon Reactions</u>. Analysis of X-ray powder patterns of carbon-reduced PuO_2 samples indicate that the route leading to the formation of PuO may proceed in the following manner.

$$PuO_2 \xrightarrow{+C} \alpha\text{-}Pu_2O_3 \xrightarrow{+C} \beta\text{-}Pu_2O_3 \xrightarrow{+C} Pu_2C_3 \xrightarrow{+\ \beta\text{-}Pu_2O_3} PuC \xrightarrow{+\beta\text{-}Pu_2O_3} PuO$$

PuO_2-MgO Phase Studies. Plutonium dioxide and magnesium oxide appear to be entirely insoluble in each other in the liquid state. The melting points of PuO_2 and MgO were unaffected by additions of the other component and were 2300 and 2800 C, respectively, under one atmosphere of helium.

Plutonium Oxides Research. Preliminary results indicate that the region between O/Pu = 1.5 and O/Pu = 1.0 in the plutonium-oxygen phases system may be a two-phase field of PuO and β-Pu_2O_3. The approximate melting point of PuO has been determined as 1800 C.

3. Ceramic (Uranium) Fuels Research

Irradiation of UO_2 Single Crystal. Additional evidence that the thermal conductivity of large grain UO_2 is much greater than that of small grain UO_2 was provided by irradiating $\frac{1}{2}$-inch diameter single crystal and tri-crystal UO_2 pellets. Gross structure alteration (other than fracture) did not occur during irradiation conditions known to cause central melting and extensive columnar grain growth in fine grained, polycrystalline samples. These results support the hypothesis that columnar grain growth in fine grained or powdered UO_2 causes rapid improvement in thermal conductivity, often leading to solidification of an initially molten core. The improvement in thermal conductivity markedly changes the radial fuel temperature profile and reduces central fuel temperatures.

Fabrication of Cermets. A 50 w/o uranium monosulfide-tungsten cermet was compacted by high-energy-rate impaction at 1200 C. An impact pressure of approximately 500,000 psi was exerted by the modified Bridgman anvil technique used. A uniform tungsten matrix was obtained. The density of the cermet was 13.68 g/cc (98.0% TD).

Impacted UO_2. Reactor quality UO_2 fuel material (98.9% TD, 2.0097 O/U ratio), was produced directly from sintered scrap by impaction with a tool steel punch at 200,000 psi. One hundred pounds of sintered UO_2 scrap was pulverized to approximately -60 mesh and roasted at 140 C for 16 hours. This oxidized only the smallest particles and did not affect the larger particles. Sixteen-pound lots of the material were then placed in stainless steel containers 3.75 inches in diameter and 5.75 inches long, heated in vacuum to 1200 C and impacted at 200,000 psi. Satisfactory density and O/U ratio were achieved without subsequent heat treatment. The combination of a vacuum atmosphere and partial oxidation of fine particles in the UO_2 is apparently responsible for the improved results.

Free Metal in UO_2. A new technique was used to determine approximately 0.1% free uranium metal in UO_2 which previously had been analyzed by coulometric titration techniques as hyperstoichiometric ($O/U = 2.003\underline{9}$) material.

The technique involves manometric measurement of the volume of hydrogen gas released during HCl dissolution and oxidation of free metal to an equilibrium condition between the fourth and fifth valence states. The gravimetric oxidation method yielded a value of $0.08\underline{4}$% free metal, while the replicate value by the new technique gave $0.09\underline{2}$% metal. The manometric method presently appears to be only semi-quantitative but possibly can be improved.

Electron Microscopy of UO_2. Time-lapse motion picture techniques were used to show that the high temperature reactions occurring in a UO_2-W cermet during reflection electron microscopy begin abruptly and proceed rapidly to completion within a few minutes. The reaction temperature for the UO_2 surface is about 800 C for nonsintered material and about 1200 C for sintered material. The reaction temperature for W is about 1200 C for either case.

Polystyrene replication has been completed successfully on selected areas of a non-irradiated UO_2 fuel element containing tungsten wire markers; the technique appears to be feasible for irradiated materials.

A chemical thinning apparatus similar to that designed by Dr. Amelinckx (Belgium) has been fabricated and investigations begun on the preparation of thin specimens of UO_2 suitable for transmission electron microscopy. Initial ultramicrotomy experiments have indicated it may be possible to thin-section UO_2 even with simple glass knives.

High Temperature Microscopy. Reflecting optics for the high temperature microscope were received and final assembly is in progress. The instrument is expected to provide 1μ resolution at specimen temperatures to 3000 C through the use of an ellipsoidal reflecting objective.

Fission Product Migration. Autoradiographs of a fuel core exposed to approximately 20,000 MWD/T revealed a concentration of metallic inclusions. The band of increased concentrations occurred approximately midway between the outer termini of the UO_2 columnar grains and the central void.

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Identification of the Molten Zone in Irradiated UO_2. Microstructural features characteristic of once-molten UO_2 were positively identified during examination of an irradiated pellet that originally contained small tungsten marker wires at known locations. During the initial reactor startup, UO_2 in the central region of the capsule melted and the marker wires in that region sank to the bottom of the melted zone. The positions of the displaced tungsten particles (some of which appear to have been melted) clearly equates the radial limit of initial melting to the radial limit of sub-grain structure.

Capsule for "Snout" Irradiation. Fabrication was completed on a capsule designed to investigate the initial melting and possible resolidification of a UO_2 fuel core under operating conditions. The capsule, which contains 1 v/o randomly distributed tungsten shot in UO_2, will be irradiated in a horizontal position in the "Snout" facility. Rotation of the capsule 180 degrees while in-reactor will provide a final distribution of tungsten shot from which the degree of melting and presence or lack of resolidification can be determined.

Uranium Monosulfide. An uranium monosulfide-tungsten cermet (50 w/o) was impacted at 1200 C to 98% of the theoretical density.

Irradiation of Tungsten-UO_2 Cermets. Tungsten-UO_2 cermets made by impaction were shown to be chemically and dimensionally stable during short term, high temperature irradiation. Post-irradiation examination of a UO_2-tungsten cermet capsule irradiated for two hours at a surface temperature greater than 2100 C revealed no evidence of reaction between the UO_2 and tungsten. The post-irradiation diameter of the tungsten clad capsule was 0.250 $\pm$ 0.001 inch - the same as the pre-irradiation value.

4. Basic Swelling Program

Irradiation Program. A capsule was discharged after reaching its goal exposure and another capsule was charged. Two capsules, each containing three hollow, split cylinders of uranium, are being irradiated at this time. The control temperature on one capsule is 625 C and on the other 525 C regardless of reactor operating conditions. A third capsule is at the reactor awaiting its turn for charging and a fourth capsule is complete except for installation of the nozzle cap, connectors, and protective conduit for the leads. The disassembly of two capsules irradiated to 0.15 a/o B.U. at 575 C is complete and five of the six uranium specimens have been recovered. One specimen is locked onto the specimen holder and

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will require special techniques for removal. Future irradiations are being shortened in duration to lower burnup of U-235 to minimize specimen distortion which creates post-irradiation examination difficulties. The specimen holder on the next capsule scheduled for assembly is being redesigned to accommodate four hollow, split cylinders instead of the former three hollow, split cylinders.

This capsule will contain two cylinders of high purity uranium - one cylinder of uranium with known amounts of iron and aluminum, and one cylinder with known amounts of iron and silicon.

The thermocouple wire for the capsule instrumentation at the reactor has arrived and will be installed shortly. Special transducers for use with the temperature controllers and the new digital readout system are on order. It is planned to begin testing of the reworked controllers and digital system in about a month.

Post-Irradiation Examination. The samples from two capsules (irradiated to 0.27 a/o B.U. at control temperatures of 575 C and 625 C, respectively) that were removed from their holders have been completely processed. The observations made on these specimens are consistent with data obtained previously. Growth and tearing persist at temperatures of irradiation as high as 600 C. At an irradiation temperature of 625 C, pores have segregated at the boundaries of the small grains originally present in the specimen prior to irradiation; pore-free bands on either side of the boundaries have formed. Heating into the beta phase results in transformation and in large grains whose boundaries do not correspond with prior boundaries and do not contain gas pores. At a burnup of 0.27 a/o (specimen 14-B), the segregated pores have coalesced into cracks at the boundaries of the original grains and into large irregular pores at many of the points where three grains intersect. Gas mobility must therefore be much more rapid along grain boundaries.

Two additional capsules (irradiated to 0.15 a/o B.U. at control temperatures of 575 C) were opened and the specimens recovered. Each capsule contained four as-extruded and two beta treated tubular specimens with a 0.030-inch wall. One of the beta treated samples was broken during removal from the holder and another specimen is still stuck on the holder. All specimens exhibited irradiation growth; the beta treated samples were worse in all instances than the as-extruded specimens. The samples that fissioned at the higher specific fissioning rate exhibited somewhat less damage than did other samples having the same burnup. Minor differences between the specimens with regard to metallurgical history and irradiation conditions may have contributed to the apparent fission rate-damage observation. Additional study is required to correlate reduced irradiation growth with increased fission rate per se. The samples are being processed for metallography and density.

Thorium. The examination of replicas stripped from the surface of each of three polished and cathodically vacuum etched thorium specimens (grip ends of broken tensile specimens) indicates no effect of irradiation or post-irradiation annealing on either the porosity or etching characteristics of the matrix thorium as compared with the unirradiated control specimen. Post-irradiation annealing at a temperature of 750 C resulted in a small but definite increase in the amount of second phase constituent (tentatively identified as Th_7Fe_3) that outlines prior boundaries. No difference was observed between the as-irradiated sample and the non-irradiated control sample. All three samples were quite similar with regard to the angular oxide inclusions present and the light gray precipitate thought to be thorium carbide.

The hardness of these samples has been measured and is summarized as follows:

Sample No.	Condition	Hardness, Rockwell "f" (Avg. of 10 Readings/Sample)
CT-9	Unirradiated Control	62 ± 8
FX	Irradiated to 0.085 a/o B.U. at ~ 200 C	92 ± 6
JX	Irradiated to 0.095 a/o B.U. at ~ 200 C and then vacuum annealed at 750 C for 100 hours	75 ± 2

It is obvious that irradiation has caused hardening and that annealing has resulted in appreciable recovery. This correlates with the tensile data that had been reported in HW-66693. Density values on these specimens will be available shortly.

Two additional specimens of thorium irradiated to 0.18 and 0.92 a/o B.U., respectively, at less than 300 C, are being processed for metallography, hardness, and density.

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5. Irradiation Damage to Reactor Metals

Alloy Selection. Procurement of materials to be used as test specimens for the Irradiation Effects on Reactor Structural Materials program is continuing. Bids have been received and are being reviewed for quantities of Inconel 600, Inconel X-750, AM-355, 304 SS, and 348 SS. In addition, negotiations for plate and rod of Zircaloy-2 have been initiated. Ten tons each of A212B and A302B pressure vessel steel have been rolled and heat treated at the Homestead Mill of United States Steel Corporation. The A212B alloy has successfully met all requirements of the specifications to date. The A302B alloy was found to exhibit strength properties higher than specified which would indicate inferior ductility properties. This material will be redrawn at 1250 F to improve upon these properties.

Three slabs of Hastelloy X-280, five inches in thickness, were rolled at the Haynes Stellite Company rolling mill to plates $2\frac{1}{2}$ inches, 1 inch, and $\frac{1}{2}$ inch thick. The fabrication history of these plates was recorded at the mill and will be part of the documentation history of this alloy. These plates were hot rolled at 1800-2150 F to final size.

Charpy impact specimens of Hastelloy X-280 and type AISI 406 alloy were completed this month together with tensile specimens of R-27, R-41, R-235, Inconel 718, Inconel 625, and TD nickel. These specimens will be irradiated at 1700 F to an exposure of 1.2×10^{20} nvt.

High temperature refractory metals may be required as hardware pieces for the helium gas loop facility to be located in the ATR. To protect the metal from oxidizing environments a protective coating will be required. One possible alloy acceptable for this application from a strength and fabrication standpoint is a columbium, 2.5 zirconium, 10 tungsten alloy, Cb752. A sheet specimen of this alloy, coated with a silicide coating, was exposed to CO_2 at a temperature of 2300 F. Data showed this coating failed after 24 hours of exposure. Further examination of the specimen is continuing to determine the method of failure.

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In-Reactor Measurement of Mechanical Properties. A creep capsule has been charged into the reactor to begin a series of tests for measuring the stress dependency of Zircaloy-2 creep under irradiation. The test conditions are: 350 C test temperature, 20,000 psi stress, on a 20% cold worked Zircaloy-2 specimen. This capsule has an improved set of heating elements. It replaces a previous capsule in which the heating elements failed. All systems in the newly charged capsule are operating within established limits. Stress was applied to the specimen as the reactor power leveled off. There has not been sufficient time at the test conditions to cite the rates; however, the curve appears to be following the same general pattern established with the series of in-reactor creep tests at 30,000 psi stress.

The capsule in which the heating elements failed has been equipped with new pressure instrumentation so that an in-reactor tensile test can be run on the specimen. All systems in the capsule are operating satisfactorily with the exception of the heaters. The loss of the heaters presents no particular problem for the tensile test as the temperature of the specimen can be controlled entirely with gamma heating which will remain constant during the time required for the tensile test. The in-reactor tensile test is scheduled for completion next month.

A series of tests on heating element wires are in progress to establish a criteria of selection for long term in-reactor use. The in-reactor heater lifetime test capsule is awaiting reactor charging. Irradiation of the heater wire tensile test specimens has been completed and the capsule shipped to Radiometallurgy Laboratory for opening and removal of the test specimens. The modifications to the Instron jaws permit the heating of the irradiated specimens by electrical current through the section of the heater wire under test.

A new sealing gland was developed this month to allow higher pressures in the capsule with continuous operation of the micropositioner drive shaft. This sealing gland was required in the instrumentation set up to run the tensile test in the in-reactor capsule. Previously, a standard conical rubber seal was used to seal the rotating drive shaft from the pressure applied to the capsule. The seal was intended for high pressures, 1000 psi; however, the continual operation of the micropositioner drive for only a few minutes created leaks around the rubber gland at pressures exceeding 800 psi. The new seal uses a highly polished and exactly dimensioned O-ring assembly which has withstood continuous operation for a period of time exceeding 100 hours at a pressure of 1000 psi.

At the end of this time no leak was detected with a helium leak detector. The sealing gland was disassembled, inspected, and a new O-ring installed for the use of the gland on the tensile test capsule.

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

During the month, a total of 41 control tensile specimens of AISI 304, 348, 410, and AM-350 were tested at room temperature. These control specimens were exposed for varying lengths of time in the ex-reactor hot water loop. Tensile tests were also performed on five irradiated Zircaloy-2 specimens at room temperature. Data from these tests are being processed on the IBM 7090 computer.

Results of previous tensile tests performed on AISI Types 304, 348, 410, and AM-350 stainless steel specimens in the irradiated, unirradiated but exposed in ex-reactor loop, and as-fabricated conditions have been obtained. These results indicate that the usual changes in tensile properties resulting from neutron irradiation were noted in AISI 304 and 348; namely, a large increase in yield strength, a lesser increase in ultimate tensile strength, and a decrease in total elongation. Preliminary data indicate that specimens of Type 348 stainless steel in a 25% cold-worked condition tend to approach a saturation in strength at relatively low neutron exposures. The remainder of the data on the annealed type 348 and both the annealed and cold-worked type 304 show a continuing increase in strength up to 1.1×10^{20} nvt, the highest exposure obtained to date. It was also noted that the decrease in uniform elongation due to neutron irradiation was much less than that due to prior cold work. This plus the observation that directional effects seem to be preserved in the irradiated state for type 348 stainless indicate that prior cold-working and thermal environment may have a greater effect on properties of these materials than damage resulting from irradiation.

Examination of control data (specimens which were exposed in ex-reactor hot water loop) show that both types 304 and 348 stainless steels exhibit a similar increase in strength with increasing amounts of cold work. However, type 348 stainless has a considerably lower value of uniform elongation than type 304 stainless in both the annealed and cold-worked conditions. Cold-worked specimens of both materials exposed in the ex-reactor loop exhibit an increase in yield strength and to a lesser extent an increase in ultimate tensile strength. The most significant increase, from 104,000 to 116,000 psi, was noted in the yield strength of type 304 stainless steel. Long term exposure at 280 C in the ex-reactor loop also caused a slight reduction in uniform elongation. This is more pronounced in type 304 stainless steel, which exhibits reductions from 78.7 to 69.8% in the annealed state and from 16.3 to 3.5% in the cold worked state. Such changes in properties noted above may result from the inherent instability of the austenite in these steels. One possible explanation for this is the reversion of austenite to martensite which may be brought about by long term annealing, resulting in an increase in strength and decrease in ductility. Such a reversion would be more pronounced in the cold-worked state due to the stored energy and shear strains induced.

A bending test jig for large capacity loading has been designed and is being fabricated. This jig is to be used in studying the fracturing characteristics of notched beams tested in bending.

Over 100 flat tensile specimens have been fabricated from AISI 406 stainless steel made by Carpenter Steel Company. Specimens of this material will be inserted in both the ETR and ex-reactor hot water loops.

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

Three round subsize tensile specimens were machined from the one-eighth-inch swaged stock to evaluate the specimen design and the material. The use of a subsize specimen is dictated by the limited amount of material available, the cost of the material, and the necessity to cold work the material to reduce the grain size. It has been shown by other investigation that a grain size such that there are at least 20 grains across the reduced section of the tensile specimens is necessary to essentially eliminate the grain size as a variable. The gage length for these specimens was 1.000 inch with a diameter of 0.100 ± 0.003 inch. The tests were pulled with an Instron testing machine at a strain rate of 0.005 inch/min, crosshead motion, using a Class B1 extensometer for the initial portion of the curves. Sample 1 had a machine finish while

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sample 2 was etched with 50-50% HNO_3-H_2O to remove two mils of surface metal and sample 3 was a reject which had a reduced section not axially symetric and was etched to remove six mils of surface metal.

All three samples were recrystallized at 650 C for one hour in vacuum following the previously specified surface treatment. Tensile tests showed that sample 1 had a pronounced yield point with an upper yield strength of 29.1×10^3 psi and a lower yield strength of 21.8×10^3 psi, while samples 2 and 3 did not show a yield point and had proportional limits of 14.6 and 16.1 x 10^3 psi, respectively. These values are in good agreement with the limited data published on material of comparable purity of 16.0 and 15.0×10^3 psi. The uniform elongation, a sensitive indicator of purity, was 26% for sample 1, and 31 and 32% for samples 2 and 3.

The most striking feature of these results was the difference in yielding behavior between the samples which had been etched prior to recrystallization and the one which had not. Apparently the heating of the sample during machining is great enough to cause a surface reaction with carbon, nitrogen or oxygen, and if not removed by etching, these interstitials diffuse into the sample during the subsequent recrystallization treatment and form a Cottrell atmosphere which gives rise to the heterogeneous yielding observed. Analytical data are not yet available to confirm this hypothesis.

Evaluation of density measurements has continued using a three-gram sample of single crystal molybdenum. Fine determinations have yielded an average value of 10.220 ± 0.010 g/cc which compares very well with a density determination by lattice constant measurements of 10.220 ± 0.005 g/cc. It is hoped that further refinements in experimental technique will improve the reproducibility of the pycnometric method to $\pm$ 0.005 g/cc.

A model for radiation damage has been proposed on the basis of the investigation to determine the mechanical properties of structural material during irradiation. In order to substantiate this model, a study of irradiated copper at low temperatures has been started. From this, the basic nature of irradiation hardening, which affects creep as well as other mechanical properties, will be determined. The specific parameters being investigated are: the force distance relationships between glide dislocations and obstacles which act as barriers to glide, the spacing of these obstacles, and the back stress acting on glide dislocations due

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to cold work. Techniques have been developed to measure creep of irradiated copper specimens in the range of 77 K to 293 K. Initial data, comparing unirradiated polycrystalline copper specimens with those irradiated to 5×10^{17} nvt, indicates a considerable difference in the force distance relationships between the glide dislocations and obstacles. Preliminary analysis indicates the irradiation damage model first proposed by Seeger and later refined by Basinski ("Thermally Activated Glide in FCC Metals," Phil. Mag., 1959, v 4, p 393) is closest to our observations.

Oxidation of Heat Resistant Alloys. The oxidation characteristics of Haynes 25, a cobalt base alloy, have been studied at 1000 C. The effect of surface treatment and oxygen partial pressure has been determined qualitatively. Specimens of the alloy were in the form of coupons $\frac{1}{2}$-inch x 2 inches x 0.030-inch. High purity research-grade oxygen and laboratory-atmosphere air were used as oxidants. A continuously recording Ainsworth semi-micro balance was used to follow the oxidation gravimetrically.

Coupons oxidized in atmospheric pressure air showed weight gains of about 1.5 times the weight gains observed in oxygen at 25 mm partial pressure. However, a specimen oxidized at 3 mm of O_2 gained weight at about the same rate as one oxidized at 25 mm of O_2.

For a given atmosphere, surface preparation (abraded, electrolytically polished, as received) has a minor but unpredictable effect on oxidation rate.

TD nickel (thoria dispersion strengthened nickel alloy) oxidizes at a rate several times that of Haynes 25. After 1000 minutes at 1000 C, 25 mm O_2 pressure, a specimen of Haynes 25 has gained one-fifth the weight of a TD nickel specimen oxidized under similar conditions of atmosphere and surface preparation.

Irradiation Damage to Inconel. Metallographic examination of the Inconel tube from the DR-1 gas loop is about 90% complete. Samples taken from a location, toward the shield, three feet from the transverse break, showed sporadic oxidation with a maximum penetration of two mils at both the inner and outer surfaces. The tube temperature at this point ranged from 750 F to 200 F depending on the specific test being run. Samples taken seven feet from the break had a light oxide coating, with a maximum depth of one mil. At 11 feet from the break, there was a slight oxidation at the outer surface. and no indication of any attack at the inner surface. Tube temperature at 7 and 11 feet was about 700 F. The oxidation of the tube away from the transverse break is generally greater on the outer surface, with more indication of sulfur contamination on this surface. The oxidation at the break was 5 mils deep on both surfaces, but intergranular attack is several times the oxide penetration.

Gas Loop Development. Methods of joining pipe of superalloys, stainless steels and possibly TD nickel in unusual combinations are being devised for construction of the model ATR high temperature gas loop. Piping has been obtained for welding studies involving the following candidate metals: Hastelloy X, Hastelloy C, Haynes Alloy 25 and stainless steels 347 and 316. The stainless pipe was purchased as an off-the-shelf item. However, because of long commercial delivery times and because of the tight loop construction schedule, test samples of 1½-inch superalloy pipe were fabricated at Hanford. Available Hastelloy X and Hastelloy C plate were preheated to 2050 F, hot rolled to about 0.150-inch thickness, subsequently annealed for one hour at 2150 and 2225 F, respectively, and air quenched. After surface cleaning, the Hastelloy X and C along with some Haynes Alloy 25 plate were sheared, machined, press formed, cleaned and inert-gas, tungsten-arc welded into seamed pipe.

Using this pipe, butt welds have been made of the superalloys to themselves, to each other and to the stainless steels. All possible metal combinations were inert-gas, tungsten-arc welded in both the horizontal and vertical positions. The pipe ends were beveled 37½ degrees to a feather edge for welding and a root opening of 1/16 - 3/32 inch was used. The filler wire (1/16 or 3/32 inch in diameter) used was: base metal composition where pipes of the same metal were joined; Hastelloy W where superalloys and superalloys or stainless steels were joined; and stainless steel 316 where stainless steel 316 and 347 were joined. If the weld quality is satisfactory, the welding procedures will be qualified for pressure piping in accordance with the requirements of Section IX of the ASME Code.

The welds appear to be physically sound. No cracking or incomplete fusion has been detected by visual examination, radiographing, or fluorescent penetrant testing. What are apparently a few small isolated inclusions have been found in most of the 36 welds. Positive identification is being established by destructive testing. The inclusions are not considered to significantly compromise weld quality and it is expected that they can be largely avoided in the future by better weld joint preparation and by more closely controlled welding conditions.

Test Section Liner. Silicon carbide is being considered as the inner liner material of the test section in the model ATR loop. It has a high structural capacity at elevated temperatures and it has a high resistance to damage from thermal shock. Some commercially available materials have been obtained and subjected to initial thermal cycling screening tests. Three-inch long sections of a hollow cylinder of SiC bonded with Si_3N_4 (four-inch OD, three-inch ID) and one of self-bonded SiC (three-inch OD, two and one-half inch ID) have been thermal cycled between room temperature and 2000 F, and room temperature and 2350 F. Cooling was in still air for one set of samples and by an air quench for another set for each of the two cycles involved. After 20 cycles, the silicon carbide showed no deterioration. Also, a small cube of silicon carbide foam was cycled 20 times between room temperature and 2350 F. The thermal cycling did not do apparent damage to the SiC foam. However, its strength was so low that it was damaged by normal handling.

TD Nickel. In evaluating TD Nickel (Ni-2 w/o ThO_2) for high temperature gas applications, there have been some unexpected results. Vendor characterization of the metal includes the following: the thoria dispersion particles are about 0.1 micron in diameter, the recrystallization temperature is well above 2000 F, and except possibly for high temperatures (> 2550 F), annealing should produce a fine grain size (< 3 micron diameter). The TD Nickel under study at Hanford does not have this character. It had thoria particles up to two microns in diameter and exhibited recrystallization and growth of grains to 100 microns in dimension when annealed between 1400 and 2000 F. The fine dispersion apparently is needed to achieve the high recrystallization temperature range and a small recrystallized grain size. Further, the larger thoria particle size with its greater interparticle spacing has probably compromised the high temperature properties of this particular metal. The vendor has indicated he will replace the initial order with TD Nickel representative of normal production.

Neutron Dosimetry and Radiation Effects Studies. Cross-sections, $\overline{\sigma_i}$, which are useful for computing the flux, φ, of neutrons lying between energy E, and 10^7 ev can be obtained from the following equation:

$$\overline{\sigma_i} = \frac{\int_0^{10^7} \sigma_i(E)\varphi(E)\, dE}{\int_{D_1}^{10^7} \varphi(E)\, dE}$$

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Values of $\overline{\sigma_i}$ for sulfur, nickel and iron (listed in decreasing order of reliability) were calculated using best available data for the energy dependent absorption cross sections, $\sigma_i(E)$, and values of $\varphi(E)$ computed for a simulated C-Reactor lattice from one dimensional multi-group, multi-region transport theory (Table I). The values, to good approximation, permit calculation of integrated C-Reactor fluxes of neutrons having energies greater than 0.18 Mev or 1.0 Mev (the thresholds commonly used in radiation damage studies) for different positions in the lattice cell. The large variation in neutron flux over the cell dimension, as shown in Table I, implies even larger changes with reactor type and design, and illustrates the necessity for determining cross-section values for different irradiation facilities if radiation damage data are to be compared meaningfully.

TABLE I

$\overline{\sigma_i}$ VALUES CALCULATED FOR C-REACTOR LATTICE BY TRANSPORT-THEORY ANALYSIS (GE-HAPO-S-XI)

Lattice Position		$\overline{\sigma_{Ni}}$, mb	$\overline{\sigma_{Fe}}$, mb	$\overline{\sigma_S}$, mb	Ratio, $\frac{\varphi(E > 0.18 \text{ Mev})}{\varphi(E > 1.0 \text{ Mev})}$
Cylindrical cell boundary	$E_1 = 1$	91	24	97	2.65
r = 12.0 cm	$E_1=0.18$	34	9	37	
Center of process-tube wall	$E_1 = 1$	106	28	72	2.19
r = 2.14 cm	$E_1=0.18$	48	13	33	
Center of fuel	$E_1 = 1$	103	28	47	1.97
r = 0.0 cm	$E_1=0.18$	52	14.2	24	

A two-dimensional transport-theory analysis is in progress for the N-Reactor lattice. The results should provide an insight into potential flux gradients along the tube channels that would give rise to differences in graphite contraction rates.

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6. Gas-Cooled Reactor Studies

B_4C Graphite Irradiations. Capsule #6 containing five samples of B_4C graphite is currently under irradiation in the Snout II facility of KW. Operation is satisfactory but the temperature is higher than anticipated for exposure conditions.

Hot-Capsule Temperature Calculations. Many irradiations of graphite have been performed in the MTR and ETR in "hot capsules". The hot capsule is a small device which utilizes gamma heating to achieve desired irradiation temperatures, but the temperature is not monitored. The internal construction of the capsule consists of four quarter-round samples, 4 inches long, which form a 0.750-inch diameter cylinder. At each end a 0.375-inch long graphite collar holds the samples in place and increases the diameter to 0.875-inch. The irregular graphite cylinder thus formed is centered within an aluminum capsule by two vitreous-alumina insulating rods which engage in shallow holes machined axially in the ends of the cylinder. Helium fills the annular space.

Expected temperatures of "hot capsules" as a function of gamma heating were hand-calculated in the past using relatively simple approximative methods. Recently the 7090 computer has been used to obtain much more sophisticated estimates of temperatures. For specimens exposed in the ETR at gamma heating rates of about 8 watts/g the average sample temperature has been reported in the range 500 to 700 C on the basis of previous calculations. The newly calculated temperatures fit into this range, although they are on the high side. Typical temperatures calculated for heating rates of 4, 8, 16 and 20 watts/g were respectively 407, 668, 1040, and 1180 C. The radial temperature drop for an average sample temperature of 650 C is 5 C and the longitudinal drop is 60 C.

In-Reactor Creep of Graphite. The second tensile-creep capsule GEH-13-91 has been removed from the ETR after three cycles. The third capsule, GEH-13-92, has been fabricated and is at the ETR for installation this month. It is scheduled for three to six cycles of operation depending on the exposure per cycle.

SiC Coatings for Graphite. The third boat of coated pieces has been loaded and is scheduled for the hot test hole at K-Reactor. This boat, the last of the series, contains tubes, rods, spheres, and rectangular parallelepipeds of 901-S and AX-25 graphite coated with SiC. The samples have been cycled between 250 and 1200 C in flowing air about 900 times to test coating integrity

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prior to irradiation. After irradiation they will be thermally cycled again to determine whether reactor exposure had any effect on coating integrity.

7. Graphite Radiation Damage Studies

Annealing of Irradiated Graphite. An argon-atmosphere apparatus for determining the coefficient of thermal expansion (CTE) at temperatures up to 1000 C is under construction. Using four 0.0001-inch dial gauges, it will handle four specimens simultaneously in routine thermal expansion measurements. Also under construction is another argon-atmosphere, quartz CTE apparatus, capable of measuring length changes during annealing to 2×10^{-6} in. The feasibility of extending dilatometric measurements on irradiated graphites to temperatures of 1000 to 2500 C is also being investigated.

Radiation Damage in Graphite. Sonic-modulus determinations yield direct information on the elastic properties of a material under microscopic deformations. Additional information can be derived using formulas relating modulus to thermal conductivity and thermal expansion.

The apparent crystallite size in graphite is commonly calculated from the broadening of x-ray diffraction lines. However, the line broadening is caused by only that portion of the material perfect enough to produce diffraction. It is also possible to calculate a phonon mean free path, which is analogous to a crystallite size, using the following relationship:

$$K = 1/3\ C_p\ \lambda\ v$$

where K is the thermal conductivity, C_p is a volumetric specific heat, λ is the phonon mean free path, and v is the phonon velocity. The phonon velocity is assumed to be equal to the velocity of sound, v_s, which is proportional to the square root of the elastic modulus, E.

Using these relations, the following ratio is obtained for irradiated graphite relative to unirradiated graphite.

$$\lambda/\lambda_o = K/K_o \sqrt{E_o/E}$$

Such ratios were calculated from K and E for graphite made from a lampblack base and heat treated to 3000 C. After irradiation to 2600 Mwd/At_K at 650 C (equivalent to 6×10^{20} nvt, E > 0.18 Mev),

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the ratio λ/λ_o for two samples was 0.29 ± 0.01. This decrease by a factor of three in the mean free path is equivalent to a ten-fold increase in scattering centers.

Although the mean free path for phonon scattering was reduced by this irradiation, the apparent crystallite size measured from x-ray line broadening increased from 170 to 350 A. A plausible explanation is that the apparent increase in L_a after irradiation is not due to crystallite growth but is due to relief of microscopic stresses believed to exist in lampblack-based materials.

An additional quantity obtained thermodynamically is Gruniesen's constant, γ, which is given by

$$\gamma = \frac{3V\alpha}{Kc_v}$$

where V is the molar volume, α is the coefficient of thermal expansion, K is the compressibility, and c_v is the molar specific heat. γ is a measure of the dependence of the lattice vibrational frequencies on the volume. As the crystals are compressed, they become harder, the restoring forces between atoms become larger, and the vibrational frequencies increase.

Since radiation-induced changes in volume and specific heat are small, to good approximation the relative change in γ is given by

$$\gamma/\gamma_o = \frac{\alpha/E}{\alpha_o/E_o}$$

For the case of irradiated lampblack samples $\gamma/\gamma_o = 2/3$, which implies that the lattice frequencies in this material are less sensitive to volume changes after irradiation.

8. Aluminum Corrosion and Alloy Development

A new in-reactor test of unfueled aluminum clad fuel elements was prepared and charged into H-1 Loop. It will run for three months under the same conditions as the discharged test (i.e., 550 F, pH 4.5 H_3PO_4, flow velocity 25 ft/sec).

Design is progressing on the modification of C-1 Loop for corrosion testing of aluminum-clad fuel elements. Orders have been placed for the valves, piping, and control equipment required. The loop is ready to remove from 105-C and start modification work in the H-Area shops.

Effect of Corroding Iron on the Oxidation Rate of Aluminum. Draley and Ruther (Conference on the Corrosion of Reactor Materials, Salzburg, June 1962) proposed a corrosion mechanism for aluminum based on deposition of colloids from the corrosion environment. It was assumed that the colloid inhibits corrosion by plugging pores and cracks in the aluminum oxide film. While the major source of the precipitating colloid was assumed to be aluminum oxide corrosion product, corrosion of loop materials was also considered as a source. Possible effects of iron oxide on the dynamic corrosion of aluminum were investigated by insertion of carbon steel coupons in the center of the normal loading of aluminum coupons.

Corrosion rates were compared for aluminum coupons upstream and downstream from the corroding carbon steel. The dynamic corrosion facility was operated with demineralized water at 330 C, 9.4 gph refreshment and 25 fps linear flow rate. After a 10-day exposure, penetrations on X-8001 alloy coupons were about equal for coupons upstream and downstream from the carbon steel. Average penetrations for two upstream and two downstream coupons were 0.90 and 0.96 mil, respectively. In a previous run at the same conditions in the absence of carbon steel, the penetration at 10 days was 0.85 mil. The exposure has been continued to 30 days; however, the data at 10 days indicate no significant effect of corroding iron on the aluminum corrosion rate under the conditions of this test.

9. Metallic Fuel Element Development

Metallic Thorium Fuel Element Fabrication. Two Th-2.5 w/o U-1 w/o Zr (fully enriched uranium) alloy ingots, 5.50-inch diameter, were vacuum welded in Si-Cu cans of 0.195-inch wall thickness and upset at 615 C in a 6.090-inch diameter container. The upset material, approximately 5.650-inch diameter, was machined and bored to form billets for coextrusion. The billets were assembled by electron beam vacuum welding both the Zircaloy and Si-Cu end plates. The final billets, 109 and 133 pounds of alloy, were coextruded on the 2700-ton, 333 Bldg., extrusion press. Extrusion conditions were: 4½-hour billet preheat at 760 C (1400 F), billet lubrication - Aqua-Dag, container temperature - 425 C (800 F), 1.790-inch die, 1.050-inch mandrel, ram speed - 14 inches per minute, extrusion ratio - 17:1. The material was extruded into steel catch tubes to prevent contamination in event of break-through and rolled in the catch tube on the run-out table.

The peak forces required for extrusion were 1670 and 1535 tons. The resulting extrusion constants were:

Billet No.	Extrusion No.	Extrusion Constant - "K" - TSI Start	Run
129	L-6	21.0	18.5
130	L-7	19.2	17.95

The over-all extrusion lengths were 18'-4½" and 22'. These tubes were sectioned to provide samples at front, middle and rear for clad thickness measurement, examination of clad-core bonding, and alloy structure. Clad thickness measurements are not completed, but visual inspection indicates extremely smooth clad-core interface for both extrusions with no problems expected from clad variation. The copper was chemically stripped from a section of each extrusion for examination of the Zircaloy surfaces. The surfaces cleaned uniformly, were relatively smooth and exhibit no problem with Cu-Zr compound formation. Dimensions over the Zircaloy clad and as-extruded hardness are given below.

Billet No.	Location	OD (inches)*	ID (inches)*	Hardness RF
129	Front	1.753	1.054	75
	Middle	1.759	1.058	70
	Rear	1.760	1.060	71
130	Front	1.752	1.055	69
	Middle	1.758	1.059	71
	Rear	1.759	1.060	70

*Average of four readings at 45°. Variation ~ 0.001.

The extrusion constants and as extruded hardness values are in good agreement with those determined for this alloy in the pilot heats using natural uranium.

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

19-Rod Boiling Burnout Report. The report of boiling burnout data for 19-rod bundles was completed in rough draft form. An abstract of the report is as follows:

"Boiling burnout heat flux data were obtained with electrically heated 19-rod bundle test sections in a 3.25-inch ID cylindrical tube with axial flow at 1200 psig. Five test sections were used which included heated lengths of 18.5, 19.5, and 76 inches and spacings between rods of 0.015, 0.050, 0.074 inch as maintained by spiral wire wraps on 12 of the 19 rods. Mass velocities from 500,000 to 5,000,000 lb/hr-sq ft were investigated with sub-cooled coolant inlet conditions and outlet conditions up to 36 w/o steam.

"The data show an effect of both rod spacing and mass velocity on burnout heat flux for a 19-rod bundle. A 19-rod bundle with a 0.074-inch rod spacing would have a burnout heat flux comparable to the more regular geometries (such as annuli, circular tubes, or rectangular channels) for outlet conditions near saturation. However, for rod bundles with 0.015-inch or even 0.050-inch rod spacing, the mass velocity must be at least 2,000,000 to 3,000,000 lb/hr-sq ft in order to avoid very severe reductions in burnout heat flux compared to the 0.074-inch spaced rod bundle."

Boiling Burnout with 19-Rod Test Sections Without Wire Wraps. Nineteen boiling burnout points were obtained with an electrically heated model of a 19-rod fuel bundle without the usual wire wraps to maintain a 0.050-inch spacing between rods. Burnout heat flux ranged from 300,000 to 1,600,000 Btu/hr-sq ft for mass velocities of 500,000 to 5,000,000 lb/hr-sq ft and exit conditions varied from 13.5% bulk quality to 160 F degrees bulk subcooling. All data were obtained at 1200 psig with the test section in the horizontal position.

The bundle consisted of 19 Inconel X tubes, 0.587-inch OD by 0.0125-inch wall and with a heated length 19.5 inches long. The 0.050-inch thick "warts" used for spacing were ceramic supports located at two cross-sections on the heated portion of the bundle, 6.25 inches from the ends of the heated section. The bundle was housed in a 3.25-inch ID pressure tube. Each of the tubes had a thermocouple installed at its downstream end to measure an average inside wall temperature. Five thermocouples

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were also installed between the rods at the downstream end to provide some measure of the differences in local coolant temperature.

Eight experiments were done initially, all at moderate or low coolant flow rates. The eighth experiment resulted in physical burnout with gross electrical arcing and burning of the rods. All of the center seven rods were replaced as well as two outer rods which had developed leaks. Ten additional experiments were made with the repaired test section. The tenth experiment again resulted in physical burnout. The test section has not been disassembled, but indications are that most or all of the inner rods are badly damaged.

Analysis of the data has not been started. However, a preliminary perusal of the data indicates that the burnout heat fluxes obtained are very nearly the same as those obtained with a test section identical to this one except that wire wraps were used to maintain the rod spacing.

Dome Seal Type Nozzle Closures. Fabrication of seal, test nozzle, and test apparatus is complete. The seal has been successfully tested at a hydrostatic pressure of 2650 psig, 150% of the adjusted working pressure. The working pressure of 1500 psig was adjusted upward by a ratio of allowable design stress at 70 F divided by the design stress at 570 F. There was no trace of leakage.

The seal was leak tested at steady state conditions of 570 F and 1500 psig for 116 hours. There was no indication of leakage. The space above the dome was sealed and had a water cooled condensate line leading to a collection flask. At completion of this test, the seal was disassembled. No damage or wear was observed on seal ring, dome, or nozzle.

To establish the initial seal in this type closure, the dome is preloaded to expand the seal ring into contact with the nozzle. This preload is obtained by a nut on a stud which is an integral part of the dome. In these tests the initial torque on this nut when the seal is assembled is 100 lb-ft. At the conclusion of the 116-hour test run with the seal cold and depressurized the torque required to loosen the nut was zero. The cause of loosening is not understood at this time. Examination of component parts did not reveal any unusual deformation or wear. The possibility exists that the tightening load was inadvertently relieved during hydrostatic testing as the seal was not

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disassembled or retorqued between the hydrotest and the steady state leak test. This phenomenon will be investigated more thoroughly in the remaining tests.

11. Advanced Reactor Concept Studies

Fast Supercritical Pressure Power Reactor. Calculations were carried out to investigate the effect of group capture cross sections in U-238 which span the resonance region on reactor physics characteristics of the Fast Supercritical Pressure Power Reactor. Because of the relatively large amount of hydrogen in this fast system, it was suspected that there may be substantial absorption in U-238 resonances due to spectral degradation. On the other hand, neutrons which escape capture in this region interact with the lower lying Pu-239 fission resonances resulting in increased multiplication. These effects are thus important in arriving at a more accurate value for the coolant reactivity coefficient.

The 18-group cross section set being used to design the Fast Supercritical Pressure Power Reactor core utilizes unshielded U-238 capture cross sections. To evaluate the importance of self-shielding, the resonance cross sections in 5 groups spanning the energy range 6-1000 ev were arbitrarily reduced by a factor of two. This had the effect of increasing K_{eff} by about 6% $\Delta k/k$ and reducing the coolant void coefficient to about 75% of its former value. It is therefore apparent that this effect is too significant to neglect. Self-shielding factors are now being calculated for the appropriate resonances. The net effects of using properly self-shielded cross sections will be a decrease in the predicted enrichment requirements and in the void coefficient.

Work is 80% complete on obtaining a limited 18-group cross section set for use with Sx computer code (transport theory) to provide an independent check of design calculations for the Fast Supercritical Pressure Power Reactor core. The cross section set is limited in that it only contains cross sections for the material in the Fast Supercritical Pressure Power Reactor core. Also, the U-238 cross sections are not corrected for self-shielding.

The effect of redesigning first pass coolant channels to reduce coolant inventory was calculated. The enrichment level was reduced approximately 10% as a result of decreasing coolant inventory. Further parametric studies will be performed to establish fuel design.

The study of Fast Supercritical Pressure Power Reactor fuel element temperatures and heat losses is continuing using the STHTP code and a new heat conduction model. This new model has the coolant tubes in a triangular lattice and permits study of cases with tubes next to the fuel element outer cladding. It is possible to study varying fuel element shapes, insulation thickness, and cladding thickness. The model has been split into 440 small nodes for greater flexibility and accuracy in calculations. A technique was developed for changing node size without excessive input change. The new model is presently being debugged in the STHTP code. More recent data on physical properties of the ZrO_2 insulation agree with previous values used in Supercritical Pressure Power Reactor design calculations.

Segmented Core for Metal Cooled Fast Reactors. Additional exploratory calculations were completed using the YOM 16 group cross section set to investigate the feasibility of the segmented core design concept in a liquid metal cooled fast breeder reactor. Substantial disagreement in calculated reactivity values compared with FCR results indicate that the YOM 16 group set may need revision in the low energy region for fast systems with degraded specta. The use of beryllium as the moderating segment although not yet completely evaluated appears promising.

Use of Plutonium in Space and Rocket Reactors. Preparation of a document describing the original Plutonium Fuel Spacecraft Reactor concept is continuing. Completion of the rough draft is scheduled for early February.

Preliminary results indicate that the application of the "moderator-segmented fast core" concept to the Plutonium Fuel Spacecraft Reactor, with the addition of boron as a burnable poison adjacent to the moderated area, may significantly decrease mechanical control requirements for the Plutonium Fuel Spacecraft Reactor. Results indicate an inherent control span of up to 20% Δk with this scheme; manipulation of the moderating elements could provide another 10% Δk. The corresponding increase in reactor diameter is only about 15% above that for the straight plutonium-fueled core. By further modifications to the moderating zone, even greater inherent control spans may be feasible.

Total control requirements for the Plutonium Fuel Spacecraft Reactor have previously been estimated at about 35% Δk to achieve the desired 20,000-hour core life. Adaptation of the

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moderator-segmented core concept to compact fast reactors such as the Plutonium Fuel Spacecraft Reactor appears to offer considerable promise of providing a major increase in endurance without undue sacrifice of controllability.

Studies are continuing of the effects of varying moderating-zone composition and geometry, and the use of burnable poisons, fertile materials, or combinations.

Moon-Base Reactor. Preliminary physics calculations on a 10 to 50 Mwe, "Phoenix fuel" reactor for powering a lunar base are being carried out by Physics and Instruments Laboratory personnel. The moon-base reactor is currently envisioned as an epithermal reactor utilizing a combined moderator-fuel loading; with a niobium alloy as in-core structural material. A lithium-cooled, indirect cycle reactor is being considered initially; it is expected that other coolants will be considered at a later date.

D. DIVISION OF RESEARCH - 05 PROGRAM

1. Radiation Effects on Metals

This program is aimed at establishing the effect of neutron irradiation on the properties and structure of specific metals containing known impurities, and to determine how the damage state is altered by thermally activated recovery processes. Studies on single crystal and polycrystalline specimens of molybdenum which have been irradiated to 10^{18} and 10^{19} nvt ($E > 1$ Mev) are progressing. Carbon in three concentration levels is the intentional impurity.

Single Crystal Molybdenum. Length changes in irradiated single-crystal molybdenum samples have been determined using gage-block comparator techniques sensitive to 10 microinches. Samples had sustained exposures to two levels, 10^{18} and 10^{19} nvt ($E > 1$ Mev), and contained carbon in three concentration ranges. After irradiation to 10^{18} nvt the sample lengths changed very slightly, from less than 10 to 20 microinches for a 1.5-inch sample, or about 0.001%. After 10^{19} nvt, the length change was 100 to 120 microinches, a change of the order of 0.01%. The level of carbon content had no effect on the length changes. These values will be used for comparison with lattice parameter changes, measurements of which are currently in progress.

Deformation modes in molybdenum single crystals irradiated to 10^{18} nvt ($E > 1$ Mev) are being examined. Large subgrains are

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formed in the deformed irradiated crystals, and the Laue spots become dispersed into patterns of smaller spots. This effect is greater the higher the carbon content. Deformation results in elongated Laue spots, or streaks, which are in some instances wavy and may intersect an adjacent streak. This behavior is being studied to determine if irradiation reduces the number of available slip planes, limits the amount of deformation on a given plane, or encourages the formation of twins.

Tensile deformation data for molybdenum single crystals are currently being analyzed in an effort to obtain resolved shear stress versus glide strain curves. Several problem areas exist: (1) the uncertainty in selection of the slip system upon which the stresses and strains are resolved, because of the difficulty in establishing the initially operative slip system; (2) accurate determination of true tensile strain at strains exceeding the range of normal extensometry techniques; (3) accurate determination of lattice rotations at strains prior to the onset of necking; and (4) the existence of a complex stress condition after necking occurs, as evidenced by the elliptical and quasi-elliptical specimen cross-sections observed in the necked regions. Regardless of the system of resolution and assuming that for any slip system the glide strain may be calculated from knowledge of the original orientation of the crystal axis and the true tensile strain, the flow curves for a given crystal will have the same general characteristics. An optical extensometry technique is now under investigation. This technique involves photoengraving a calibrated grid on the specimen and following the deformation by time-lapse photography. A grid has been prepared with lines 0.1-inch wide, spaced on one inch centers. The grid was photographed at a reduction of one-tenth and will be further reduced one-fifth by projection onto the gage length of the single crystal tensile specimen, the surface of which is coated with a photographic emulsion. Development of the emulsion by standard methods will yield a grid network on the surface of the specimen, with grid lines 0.002-inch wide and separated by 0.020-inch.

The problem of determining lattice rotations prior to necking still remains. With the x-ray technique presently in use, only the axial orientation at the maximum load is obtained. Calculations made to date have been based on the assumption that lattice rotations are a linear function of tensile strain.

Attempts are being made to determine the nature of the triaxial stress conditions prevailing in the necked regions of the tensile specimens so that a true resolved shear stress may be calculated.

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These combined stresses are being determined by study of the geometry of the necked cross-section at various values of the necking strain.

Resolved shear stress-glide strain curves have been determined for three high-carbon crystals of similar orientation and three levels of neutron exposure (0, 10^{18} nvt, and 10^{19} nvt). Stresses and strains were resolved onto three slip systems: (101) [111], (112) [111], and (213) [111]. The critical resolved shear stresses for slip on these three systems were determined to be as follows:

Specimen	Axial Orientation	Exposure, nvt	Critical Resolved Shear Stress, Kg/mm^2 (101)	(112)	(213)
72 cc1	[2 5 44]	Unirradiated	14.2	14.6	14.1
62 cc1	[1 0 3]	10^{18}	18.8	19.0	19.8
72 cc2	[7 11 45]	10^{19}	23.8	28.3	26.8

The unirradiated crystal exhibited a very limited easy glide region, a limited region of linear hardening with a very high rate of strain hardening (θ_{II} = 250 - 360 Kg/mm^2, depending on slip plane in question), and an extensive parabolic hardening region. At 10^{18} and 10^{19} nvt, extensive regions of easy glide were observed, being greater for the higher exposure. Both irradiated crystals also showed an extensive linear hardening stage with θ_{II} values of 45-59 Kg/mm^2 and 20-35 Kg/mm^2 at 10^{18} and 10^{19} nvt, respectively. These results indicate that the effect of neutron irradiation is to facilitate the operation of conjugate slip systems (i.e., on planes of one type) while restricting the ease with which dislocations may shift from one type of slip system to another.

The reduction of tensile data on single crystals to usable form is a tedious and time-consuming process. For this reason, a FORTRAN program is being prepared for the 7090 computer before analyses are begun.

Polycrystalline Molybdenum. Tensile testing of polycrystalline molybdenum specimens with exposures of 0, 10^{18}, and 10^{19} nvt is nearly complete. The specimens were machined from high-purity molybdenum containing less than 20 ppm carbon and less than 100 ppm total impurities. The specimens were separated into

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three groups and annealed for 16 hours at temperatures of 1050, 1300, and 1500 C, the objective being production of a completely recrystallized structure with three different average grain sizes. The average grain diameters for the material annealed at 1050, 1300, and 1500 C were 0.020, 0.025, and 0.029 mm, respectively. The three materials showed great differences in tensile properties. The material annealed at 1050 C exhibited a pronounced yield point in the unirradiated condition, while the other two did not. The specimens annealed at 1500 C fractured before the ultimate tensile stress was reached, but with an elongation of 20%. After an exposure of 10^{18} nvt, the 1050 C material again showed a yield point but also deformed inhomogeneously by Lüders band propagation after yielding. The same material with an exposure of 10^{19} nvt began necking immediately after yielding. The behavior of the irradiated 1300 C specimens was similar. The irradiated 1500 C specimens, however, failed in a brittle manner before reaching a yield point. It is believed that the gross differences in behavior of the three materials cannot be attributed to grain size. Although no evidence was observed metallographically of any change in microstructure, the specimens annealed at 1300 and 1500 C showed slight surface contamination. The deleterious effect of oxygen in molybdenum is well established and is felt to be the contributing factor in this case. The occurrence of a yield point in the unirradiated molybdenum annealed at 1050 C is explained by the fact that the material was only partially recrystallized.

Foils of molybdenum, 0.003-inch thick, show spot defects due to irradiation; after post-irradiation annealing the spot defects disappear and larger loops appear. The identity of the spot defects, clustered vacancies or interstitials, and the identity of the larger loop defects, possibly originating from growth of the spot defects or due to carbide-defect interactions, are not known. Various techniques have been considered for moving dislocations through the irradiated $\frac{1}{2}$-inch diameter by 0.003-inch thick molybdenum to establish how dislocations interact with the clustered defects. Preliminary experiments on unirradiated foils show promise. The foils were glued to recessed circular regions on the flat side of steel tensile specimens which were strained axially in the range one to two percent. From a comparison of the pre-strained and strained microstructure revealed by transmission electron microscopy it was concluded that dislocation movements, multiplication, and interaction with carbide precipitate particles had been achieved. The experiments are now being repeated with irradiated foils.

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As disclosed in previous reports, molybdenum foil stock, being utilized in the radiation damage and recovery studies, is also being studied by quenching experiments. In one phase of this work Johnson-Matthey high purity molybdenum foil strips, 0.003-inch thick, were heated to successively higher temperatures, 2100, 2200, and 2300, and $\sim$ 2350 C, and then quenched rapidly by a flow of helium. After thinning and electron microscopy, no defects were observed. If aging at several temperatures produces no clustering, then it will be concluded that defects introduced by quenching are not discernible by electron microscopy techniques. Similar experiments on unirradiated molybdenum containing carbon are in progress to aid in the formulation of damage mechanisms in molybdenum.

Polycrystalline $\frac{1}{2}$-inch diameter rod stock of molybdenum, from which stored energy release, length change, hardness, and tensile specimens were prepared and irradiated, has been upset in four repeat operations at a temperature of 500 C to a diameter of 0.790 inch. Attempts at upsetting at lower temperatures resulted in twisted non-uniform shapes, and failure of the die. If metallography indicates that the metal is free of fabrication defects, then additional specimens will be similarly upset to provide stock which can be swaged to a smaller diameter. These specimens will be used to measure cold work energy release as a function of temperature in the differential calorimeter.

E. CUSTOMER WORK

1. Radiometallurgy Examinations

Examination of the second hole failure from 2166D was completed. Cause of the failure was determined as groove pitting in the spire. Groove pitting was also found on the outer can wall. The pitting in this element was less severe than that in the first enriched element examined. (RM C-409)

The lathe in "F" Cell was used to remove excess boat resin from metallographic mounts of fuel elements submitted for etched number studies. This instance further illustrates the capability and versatility of this new equipment. (RM C-411)

The transverse cracks on two process tubes (1271D and 1386F) were examined and identified as fatigue failures. (RM C-412 and C-413)

Six pieces of KER aluminum process tube were sectioned into 17 tensile blanks. (RM A457)

Modulus of rupture tests were completed on nine ceramic material specimens from Boat #6. (RM A-459)

2. Equipment Projects

Critical Radiation Detection System. Installation of the critical radiation detection system is 95% complete.

Pulse Annealing Furnace. Installation of the pulse annealing furnace in "D" Cell was completed.

Chip Collection Equipment. Equipment for the collection of chips from holes milled in Pu-Al alloy fuel samples was installed in "F" Cell. These chips will be dissolved for burnup analysis.

Manipulators. Two light-duty extended-reach manipulators, for use in "D" Cell and "H" Cell, were received.

Cathodic Etchers. A new sealing gasket for the etcher bell jar was designed and fabricated from PVC plastisol.

"C" Cell Outage. "C" Cell was shut down on 12/31/62, the first time in over three years, for maintenance work on the cathodic etcher and the remote metallograph. Work on the cathodic etcher included the installation of two bell jars, two cathode plates, and new inlet and outlet water tubing with quick coupling shutoff valves. The metallograph was removed and disassembled for inspection, adjustment, and lubrication. In addition, a new water manifold was installed for the grinder-polishers. It is noteworthy that careful planning and coordination between the Maintenance, Examination, and Equipment groups resulted in only four days outage time.

Decontamination Cell Support Structure. Installation of the 5-ton support structure over the decontamination cell was completed. This structure will relieve the crane work load.

Project CGH-857, Physical and Mechanical Properties Testing Cell.

a. Construction - Final equipment installation is awaiting receipt of equipment.

b. General - The installation of one Model 8 Extended Reach CRL Manipulator was completed. Checkout of the manipulator with the B&L Optical Gage was satisfactory.

Modifications were completed as required to increase the speed of travel of the three hydraulic lifts.

Instron Tensile Grips. Fabrication and checkout of the new air-operated grip was completed. The grip is now in use for operational tensile testing. Fabrication of the mating lower grip was begun.

Neutron Radiography. A study was completed to determine the feasibility of using neutron radiography for inspection and examination purposes in the Radiometallurgy Laboratory. An outstanding advantage of this process, over x-ray radiography, is the elimination of interference on x-ray film caused by radioactive decay radiation. However, need for this equipment at present does not seem to justify an expenditure of $25,000.

3. N-Reactor Design Testing

NPR Charging Machine

During the report period a number of modifications were made to the N-Reactor charging machine. These included:

(1) Six drive rollers of a new design were fabricated and installed. No difficulty has been encountered with them to date.

(2) The electrical cables, which are connected to the rear photo cell-light combination, which indicates when a magazine is too far to the rear, were damaged during welding of the rear pressure roller bracket. These were repaired.

(3) A second vane operated limit switch was installed in series with the one that limits the upper travel of the vertical lift mechanism.

(4) All mechanical joints in the differential selsyn system were aligned and pinned.

(5) A relay which did not function adequately was replaced by a more adequate one. The rewiring necessary to make this replacement has been completed.

(6) Twenty-six fuel loading magazines were approximately 0.050 inch large on the end that fits into the nozzle adapter. A device was assembled that would grind the ends of these magazines to the correct size. Fifteen of the magazines have been ground to the proper dimensions.

A rough draft report on Design Test No. 18, Magazine Support Functional Testing, has been completed.

All structural modifications to the mockup, which were required by Design Test 22, Part A, have been completed. The water filter has been received. Fuel for use in this test probably will not be available until the first of February. The nozzle-process tube combination for use in this test has been received but requires some additional inspection prior to use.

NPR Fuel Handling Tongs

The prints and a work order covering fabrication and testing of the NPR fuel handling multiple tongs were received. Required parts are on order and shop estimates are presently being made for fabrication.

NPR Fuel Magazine Loader

Effort has been directed toward provision of laboratory space for installation and testing of the NPR magazine loader which is scheduled to arrive near the end of this month.

4. Special Plutonium Fabrication

Fission Product Transient Samples for Phillips Petroleum Company. Twenty-two, Al-clad, coextruded, tubular fission product transient samples were completed and shipped. Thirteen contained Al-U-233 alloy cores, five contained Al-U-235 cores, and four contained Al-Pu alloy cores. This brings the total number of elements shipped to 62 and completes the job.

High Exposure Plutonium-Aluminum Fuel for Physics Tests. Another 250 of the 3-foot long Pu-Al rods were completed and delivered for physics tests in the PCTR and PRCF. A total of 450 of the 1000 rods required now have been delivered.

High Exposure PuO_2-UO_2 Fuel for Flux Monitor Tests. The first batch of 600 in in process. These are pellets needed for PRTR flux monitor test fuel elements. Three rods in each of six 19-rod cluster assemblies will contain the pellets.

Low Exposure PuO_2-UO_2 Fuel for Physics Tests. Sintering of PuO_2-UO_2 pellets for PCTR test rods was postponed to permit completion of the pellets for the flux monitor elements described above.

High Exposure PuO_2-UO_2 Fuel for Physics Tests. Initial blending of PuO_2-UO_2 required for PCTR test rods was completed. Approximately two pounds of the material was processed by high energy impaction as an initial step in a program to determine the feasibility and economics of making the fuel material in this manner rather than by pelletizing.

High Exposure Plutonium-Aluminum Fuel for Corrosion Tests. The 40 Pu-Al, single rod elements required are cut from coextruded rods containing Al-8 w/o Pu-2 w/o Ni clad in aluminum. End caps are welded on after counterboring and decontaminating the cladding. (Initial closure attempts were unsuccessful because of Pu contamination in the weld.) Efforts are continuing to provide a weld area free of plutonium.

EBWR Plutonium Fuel Elements. Material procurement continued on schedule. The unclassified plutonium required for the job (60 Kg) was received, and burning to PuO_2 was initiated. An order was placed for Zircaloy rod to make end caps. Delivery is expected in April. All bids received for the Zircaloy tubing had exceptions to the ultrasonic test requirement. The exceptions were resolved and new bids are due January 29. Bids were requested for fusing the 5000 lbs of depleted uranium needed, and the order should be placed by February 15.

A project proposal for the facility to vibrationally compact the EBWR fuel rods was prepared and is circulating for approval. Meanwhile, design of the facilities continued with completion expected in February. Other activities initiated included a criticality review of the fabrication process and a determination of the optimum blend of powder sizes required for high density compaction.

Some revision of the fuel rod and assembly design was required. Modifications were made to permit remote removal and replacement of individual rods in the cluster. The need for greater fission gas release and retention capability is under investigation, and studies of the fuel assembly heat transfer characteristics were initiated.

FW Albaugh

Manager, Reactor and Fuels Laboratory

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PHYSICS AND INSTRUMENTS LABORATORY

MONTHLY REPORT

JANUARY 1963

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

N-Reactor Experiments

A preliminary determination has been made of the control strength of a spline in an NPR cross coolant channel (H_2O filled). The results show that the spline has a control strength about one-fifth as strong as the boron horizontal control rod.

Calculations of M^2 and k_∞ have been made as a function of pile flooding. The P-3 program was used to determine the lattice flux distributions. The calculations were normalized to the measured values for no pile flooding.

Data from the NPR condensed and mockup PCTR experiments are being reviewed before the final draft of a document giving the N-reactor lattice parameters is prepared.

Optimization of Retubed Lattices

All measurements planned for CIIN (regular "C") and wet CVIN (Overbore) fuel have been completed. Measurements for both fuels without coolant are being prepared.

Angular Distribution of Thermal Neutrons

An article entitled "Measurement of the Angular Distribution of Thermal Neutrons at the Surfaces of Cadmium Rods" has been completed for submission to Nuclear Science and Engineering.

Calculations of the space-energy-angle distribution of the neutron flux in the PCTR with a copper bar on the axis are in progress with HAPO Program S-XI. The thermal multigroup transfer matrix was calculated using the SUMMIT kernel. Early runs indicate that the spectrum is a little too fast. This result indicates the possibility of a basic weakness in the SUMMIT kernel in calculating transfer cross sections for graphite.

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Computational Programming Services

The necessary arrangements for placing ICEDT, the comprehensive exponential data processing code, in production status have been completed with Data Processing Control. Forms have been designed, printed, and distributed. Test data are now being run through the procedure.

Theory Experiment Correlation for Program GROUSS

Monoenergetic flux depression factors $(\bar{\phi}_{fuel}/\phi_{moderator})$ vs. Σa_{fuel} have been derived for a PCTR experiment being analyzed, namely, a natural uranium tube in tube-fueled cell and a natural uranium slug-fueled cell. Both cells are being run with air or water as the coolant. These flux depression factors were obtained for three different values of $\Sigma_{s_{fuel}}$ ($\Sigma_s = 0$, $\Sigma_s = 4\ \Sigma_{a_{fuel}}$, and $\Sigma_s = 8\ \Sigma_{a_{fuel}}$).

P-3 solutions were utilized up to the point where the analytic solutions fail ($\Sigma_{a_{fuel}}$ becomes too large, resulting in negative fluxes). For the air cooled P-3 cell cases, an effective diffusion coefficient[(1)] was assumed for the voids. S-4 analysis was utilized for the higher $\Sigma_{a_{fuel}}$ cases.

Instrumentation

Performance data for the NPR nuclear instrumentation were received from GE-APED and evaluated. Tentative agreements were reached on portions of the results. Plans are to ship the instruments in their present form.

Design and procurement work progressed on the NPR Fuel Rupture Experimental Testing Loop being installed at PRTR. The order was placed for the multichannel analyzer and associated equipment. Bids were reviewed for the flow measurement and control equipment. Detector shield assemblies were redesigned to simplify fabrication and maintenance, and specifications were written for the test oscilloscope, an oscilloscope camera, and for an X-Y recorder.

Corrections and revisions were made on the rough draft of Design Test 1133 for the prototype gamma energy spectrometer and differential alarm module for the gamma energy monitor portion of the NPR Fuel Rupture Monitoring

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(1) Garelis, E., "Treatment of Annular Voids in Diffusion Theory," Nuc. Sci. & Eng. 12, 547 (1962).

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System. Graphs are being drawn. Further tests will be conducted at GE-APED.

Design criteria were established for a new shield plug for experimental evaluation of reactor instruments. The work was in cooperation with Operational Physics and Equipment Development of IPD.

Relative radiation leakage rates were calculated as a function of clearance for the annular region between control rods and bushings. An alternate method was suggested for obtaining relief without compromising shield effectiveness. The work was done at the request of Applied Reactor Engineering, IPD.

Systems Studies

At the request of Reactor Engineering, IPD, an estimate was made of the heat generation rate in the shields of existing reactors during extended outages. The estimates were required to permit establishment of emergency cooling requirements.

Automatic reactor control studies continued with analog computer investigations of two dimensional models. Included in the study was a development of a two dimensional formulation of the one group diffusion equation. Equations for a two-by-two and three-by-three simulation, and an analog simulation of the two-by-two system were also developed.

A one-dimensional simulation of D-reactor has shown that the automatic controller can be driven directly with the voltage output from an analog simulation of a reactor. This will allow a test of the controller in a closed loop circuit prior to actual use on the reactor. Plans are made to collect D-reactor data in order to demonstrate equivalent structure concepts and to develop analog simulation of that plant. This will offer a convenient method of demonstrating automatic control concepts which are to be used in later reactor tests. An equivalent structure concept was develcped for a large reactor which is suitable for use as a one, two, or three-dimensional representation. Measurements on analog simulations indicate that the equivalent structure represents a simplified way to measure reactor parameters and to specify control relationships. The investigations revealed that control parameters can be readily measured on an operating reactor. The use of reactor equivalent structures is expected to eventually lead to improved and more safe reactor control characteristics.

N reactor system studies and simulations are continuing on schedule. The entire N-reactor plant will be simulated a portion at a time on the EASE

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analog computer. Each simulation then will be used in design and standardization studies, and for parametric optimization studies. Computer facilities are being operated on a two-shift basis in an attempt to meet the rapidly increasing work load.

The NPR pressurizer injection system simulation circuit was completed, checked out, and made operational on the analog computer. Four external automatic controllers and their associated circuitry are now being connected into the computer simulation. Preliminary tests indicate that the control system which controls the pressure differential between the primary pump section and the injection pump discharge header has marginal stability as a result of a long-time delay in an associated pneumatic tube. Methods of improving this stability are being studied.

The gas used to cool the graphite of the NPR is composed of He, CO_2, CO, H_2O, and H_2. Reactions of these gases with each other and with graphite at reactor operating temperatures cause graphite burnout. At the same time, the gases diffuse through the graphite and react with the zirconium process tubes. It is desirable to maintain the process tubes in an oxidizing atmosphere to preserve the zirconium oxide coating on the process tube and prevent hydriding. Reactor and Fuels Laboratory, HL, has submitted the problem of determining gas compositions required to maintain optimum graphite burnout rates and process tube atmosphere. To date the preliminary studies and the initial analog circuit diagram have been completed.

SEPARATIONS

Experiments with Plutonium Solutions

Criticality experiments continued with concentrated plutonium nitrate solutions in an 11.5-inch diameter stainless steel sphere. The measured volume of the sphere is 12.95 liters; the vessel wall thickness is 0.049-inch. Fifteen experiments were done bringing the total number of experiments to date to 179.

Criticality in the water reflected sphere was studied as a function of nitrate concentration. Plutonium concentrations were in the range of ~ 50 to ~ 450 g Pu/ℓ with nitric acid molarity varying from ~ 1 to about 2. The vessel was subcritical when filled with plutonium solution at a concentration of ~ 450 g Pu/ℓ with an acid molarity of ~ 2; for this solution the extrapolated critical volume from the inverse multiplication curves was ~ 13.8 liters. Criticality was achieved on subsequent dilution of the solution to reduce the nitric acid molarity. The Pu concentration at which criticality was achieved was about 300 g Pu/ℓ, but the chemical

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analyses of the solutions have not been received.

The concentrated solutions used in these experiments were obtained by boiling off excess nitric acid and water from dilute solutions in previous experiments. In the process of reconcentration, a portion of the plutonium was converted from the valence state of four, $Pu(NO_3)_4$, to the valence state of six as $PuO_2(NO_3)_2$ (perhaps 60% of the latter). The Pu reverts back to the valence state of four in time, but the chemical analysis of the mixture (with two valence states) is more difficult. The limiting factor in the determination of the critical mass values is currently the accuracy of the chemical analyses of the solutions.

A series of stainless steel nesting shells in thicknesses up to $\frac{1}{4}$-inch were used to study the effect of the stainless steel shell on the criticality of the unit. Consistent with previous results with more dilute solutions, the stainless steel becomes more effective in reducing the worth of the reflector as the concentration is increased (the correction to the critical mass in terms of g Pu/mil of stainless steel thickness is larger). These results also appear to be consistent with limited data from ORNL for the effect of stainless steel on water reflected, highly enriched UO_2F_2 solution cylinders.

<u>Experiments with Plutonium Oxide-Plastic Mixtures</u>

Work continues in preparation for criticality experiments with the remotely operated split table machine. The wiring is now about 95% complete and the necessary control panel modifications have been made. The air motor and centrifugal clutch for automatically separating the table halves in the event of loss of electrical power were installed on the device; initial tests indicate the motor will open the faces at the design speed of 22-in/min.

Two of the control or safety rod drive units were designed for rapid removal of fuel. Some difficulty was encountered in stopping the units at the end of their travel. This problem has now been solved by means of a properly designed air cushion with pressure relief which brings the units uniformly to a stop at the end of their travel in short distance and without bounce (initial acceleration on removal up to ~ 6 g).

Storage cabinets, which also serve as on-site shipping containers, for the special PuO_2-polystyrene fuels were completed. Loaded storage boxes will be positioned in a nuclearly safe array along the wall of the critical assembly room. Nuclear safety specifications covering the handling of these fuels were approved. Transfer of the fuel from Metallurgy Development to the Critical Mass Laboratory is planned during February.

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

In connection with the correlation of critical mass experiments with theory, further effort was made to compute criticality for some of the solutions as measured in the Laboratory. These calculations were carried out on the HFN Multigroup Diffusion code using eleven fast groups and one thermal group.

The calculated results are within ~3% of the measured critical volumes with the exception of one case where a difference of ~10% in critical volume is observed. This particular calculation is presently being examined to ascertain if any error exists in the calculation which would explain the difference.

Criticality Calculations for PuO_2-Polystyrene Mixtures

In support of the proposed experimental program for the PuO_2-polystyrene compacts (HW-66266-SUP 1), multigroup diffusion theory calculations have been made to determine the minimum critical mass (spherical geometry) as a function of plutonium concentration in polystyrene compacts. The calculations were carried out for both unreflected and fully water reflected spheres.

Buckling of Partially Filled Spheres

Since no satisfactory "zeroing function" which reproduced the $\phi = 0$ boundary condition at the top surface of the partially filled sphere could readily be found, another approach was tried. It was found possible to eliminate the lattice point which fell outside the physical system in the 9-point flux fit by solving the equation evaluated at the boundary. Integration of this modified 8-point function (2 of these points being zero) was found impossible without numerical approximation for all except one point. This point could be computed analytically and checked against a numerical integration procedure. It was found that 100 strips with Simpson's rule integration would reproduce the analytic value for this point to ± .05%.

One other modification was instituted during the month to eliminate the singular point at the origin of the spherical coordinate system, the function $\phi\, r^2$ was fit when this point was used, and r^2 dropped in the volume derivative ($r^2 \sin \theta \, dr \, d\theta$). Errors on this assumption are of the order of the fit of $\phi \sin \theta$, i.e., ± .05%.

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The code is ready for use to compute the buckling of nearly full spheres, as these have the least complicated lattice spacings. Later, near-hemispherical cases will be run, perhaps with an approximation for the flux at the origin, a useful number as far as computing derivatives is concerned.

Dissolving Pu Spheres

Transport theory analysis of this series of cases (solid Pu dissolving in PuO_2-H_2O reflected with H_2O) was completed. Criticality conditions for 1, 3, and 4 kg solid Pu spheres with solution concentrations from 10-400 gm/1 Pu were derived.

Subcritical Interactions

HW-75887, entitled, "Eigenvalue Matrix Solution to the N-Component Neutron Interaction Problem," was distributed on plant.

A new code, INTERSET (Interaction Setup), was written and debugged. This code reads the coordinates, height, and symmetry location of equal diameter cylindrical vessels. Output is the interaction matrix on punched cards, based on a simple approximation for large spacing between units. Additional work is necessary to decide when this approximation is good and when the double integral formulation of SOLAN (Solid Angle) is necessary. Likewise, it appears some work is necessary to decide the numerical stability (number of mesh points) of SOLAN for very close cylinders.

INTERSET is being applied to the Critical Mass design problem of 92 (error last month) interacting cylinders. A system including INTERSET, HFN, and EIGEN (a matrix eigenvalue interaction routine) is foreseen. Such a system would read coordinates and nuclear properties of each cylinder in the array and give k_{eff} and component power as output.

One side of the 92 component array, which is situated on one side of the room, was solved and found to have a k_{eff} of about .7. Approximations had to be used as the matrix was set up by hand. INTERSET now has fewer approximations, and is expected to give reliable results for the 92 component system.

Scattering Kernels for Polystyrene

Examination of the Raman and infrared absorption spectra of polystyrene indicates the possibility of adapting the water kernel code KERNEL to the generation of scattering kernels for polystyrene. Such a procedure has been successfully used at KAPL for polyethylene. Polystyrene scattering kernels will be useful in analyzing experiments with the split half machine.

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Instrumentation

Tests are being conducted to determine safety channel trip time of the period measuring instrumentation at the Critical Mass Laboratory. A positive exponential generator was constructed using operational amplifiers. The generator was made into a current source and fed into the detector end of the input cable. Various fixed periods were fed into the period meters and the time-to-trip determined. The tests are not as yet complete. A device is being constructed to "scram" the split-half machine in the event of excessive drive motor speed in the near closed position. Installation and tests are scheduled for the near future.

Wire ducts are being installed in both reactor hoods to provide more convenient routing of instrument signal cables. This should provide more room in the hoods and allow operator actions without interference from the cables.

Consulting Services on Nuclear Safety - Criticality Hazards

Nuclear Safety in HLO

A new mold design was reviewed for the Metallurgy Operation. The mold is for casting 6 Kg of plutonium metal into two bars--0.833-in by 1.5-in by 7.75-in. A geometrically safe 2-in ID pouring crucible will be used in conjunction with the new mold.

Specification C-14 was issued for Critical Mass Physics covering plutonium plastic compacts at an H/Pu of 15.

At the request of Process Equipment and Development, the nuclear safety of the Salt Cycle Process was reviewed. The prototype equipment for this process is currently in the design stage. The process separates PuO_2 from UO_2 by electrolysis in a molten salt bath and will be used to reprocess PRTR moxtyl fuel elements. Preliminary nuclear safety limits were given for each part of the process.

Comments were submitted to Ceramics Research and Development concerning the nuclear safety of several new PuO_2-UO_2 fuel enrichments that will be fabricated into fuel elements soon. The fuels reviewed were:

0.48 w/o PuO_2-UO_2 (normal uranium)
0.9 w/o PuO_2-UO_2 (depleted uranium)
1.3 w/o PuO_2-UO_2 (depleted uranium - 30 w/o Pu^{240})
2-3 w/o PuO_2-UO_2 (depleted uranium - 8 w/o Pu^{240})

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Comments were also submitted concerning a proposed modification to the plutonium storage vault in 308 Building. Several of the 24-in cubicles are to be partitioned into four 12-in cubicles. Specification J-3 permits this change provided the unit plutonium mass is 2.8 Kg or less.

Nuclear Safety in CPD

A proposed increase in the 7.35 w/o Pu-Al fuel element storage array at Z Plant was reviewed for CPD. The array was increased from ten to twenty birdcages containing about 225 fuel elements.

Nuclear Safety in IPD

Comments were submitted to Facilities Engineering concerning the nuclear safety basis used at Hanford for storing 0.95 w/o U^{235} enriched, I&E fuel elements. It was pointed out that nuclear safety specifications for all fuel element storage arrays at Hanford are based on optimum neutron moderation. The source of moderator could be water from a broken water line or the water used to combat a fire. Further, the wooden boxes and pallets used for fuel element storage already afford some moderation.

NEUTRON CROSS SECTION PROGRAM

Scattering Law Measurements for Light Water at Elevated Temperatures

The realignment of the triple-axis spectrometer preliminary to beginning inelastic slow-neutron-scattering measurements from light water at elevated temperatures has been essentially completed. Measurements are now in progress to determine the experimental conditions under which the scattering measurements will be made.

Arrangements are being made to obtain the Fortran programs, LEAP and ADDELT, on magnetic tape from AERE, Harwell. These programs, which compute the scattering law $S(\alpha,\beta)$ from the generalized frequency distribution $p(\beta)$, will be useful in the analysis of inelastic neutron scattering measurements.

Crystal Reflectivity

The Fortran program, DOUBRAGG, to calculate the "double Bragg scattering" from a beryllium crystal monochromator has been put into operation. Calculations of the Bragg angles for double Bragg reflections of half-order neutrons from the [00.2] planes are in good agreement with the available experimental data.

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Rotating-Crystal Spectrometer

The hysteresis synchronous motors to drive rotating crystals and choppers for time-of-flight measurements of slow-neutron scattering have been received. Measurements are in progress to determine the rotor load and speed and phase stability characteristics of the motors.

Fast-Neutron Cross Sections

The fast-neutron total cross section data obtained in December were processed via the data reduction program BIG NED. The analytical subprograms indicated significant shifts in operating parameters of the time-of-flight instrumentation in the course of the measurements. These shifts were directly associated with the failure of the Van de Graaff building air-conditioning system. Efforts are under way to correct for these effects and salvage as much of the data as possible.

Work is in progress to attempt to measure the fast-neutron total cross section of a sample of Pm^{147} during the next series of measurements.

Instrumentation

The logic design for the 6144 channel time-of-flight analyzer has been completed and wiring has been started. The magnet drivers for the paper tape punch have been completed.

Transistorized circuits were designed and built to be used with a 58 AVP photomultiplier tube and a NE-213 liquid scintillator fast neutron detector. These circuits provide four low impedance outputs from the photomultiplier tube: (a) low level, linear output; (b) a fast saturated pulse output for timing; (c) an output from a standard Owens-type pulse shape discriminator; and (d) a logic level pulse derived from the pulse shape discriminator to signify that a fast neutron was detected. All circuits have been built and adjusted in preparation of final check-out prior to full operation.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE

Lattice Parameters for Low Exposure PuAl

Calculations of k_∞ for the PuAl-graphite lattices measured in the PCTR have been continued. The cadmium ratios and reaction rates have been calculated and compared to the experimental results as a measure of the accuracy of the HFN and G-2 program outputs. The copper cadmium ratios

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from HFN were 40% too high and those from G-2 were 20% too low. However, the G-2 input data tape is being revised in several respects. The plutonium cross sections are being revised and the epithermal cross sections are being revised.

Lattice Parameters for Hx PuAl

The measurements in the PCTR on 19-rod clusters of Hx PuAl on a $10\frac{1}{2}$" pitch are three quarters complete. The final value of the quantity of copper required to poison the central cell to a multiplication of unity has been measured for the case in which the flux in the central cell is matched with the flux in the surrounding cells. Measurements were previously made for several unmatched cases. The fine structure of the flux in the central cell was measured using copper foils for cell traverses and PuAl, U-235 Al, Cu, Au and Lu foils to obtain the relative activation rates in each ring of rods, at the cell boundary and in the thermal column. The relative cadmium ratios for PuAl, U-235 Al, Cu, and Au are also being measured in each ring of rods, and on the cell boundary.

A report titled "A Measurement in the PCTR of the Relative Plutonium Content of PuAl Fuel Elements" was written for inclusion in the Physics Research Quarterly Report. Rods for the lattice studies were selected on the basis of these measurements and of the chemical analysis. The rods used for the central cell have a standard deviation of ± 1.5% in plutonium content, while those in the buffers have a standard deviation of ± 2.5%.

Drawings of new graphite plugs for flux traverses have been completed. These plugs are inclined at such an angle that the end of the plug intersects the cell surface at the radius of the equivalent cylindrical cell. Traverses along this plug will very closely correspond to the flux distribution calculated by the computer codes.

PCTR Lattice Measurements for Pu Fuel Elements in a Hydrogeneous Moderator

The infinite medium spectra have been calculated for homogeneous Pu-CH_2 and PuH_2O systems with approximately the same concentration of Pu as the heterogeneous systems which could be studied in the PCTR with existing fuel. The purpose of doing the calculations was to provide information concerning the value of doing experiments with Pu fuel in a polyethylene moderator in lieu of a light water moderator.

The calculated spectra for the two systems show generally good agreement for both poisoned and unpoisoned cases. Since the scattering kernels used for the two moderators are roughly equally valid and are based upon the same model, it appears that experimental data obtained in the PCTR for

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$PuCh_2$ systems would be quite useful in studies on large PuH_2O systems.

Approach to Critical Experiments in H_2O for PuAl Fuel

Experiments similar to the ones performed on the 1.8 w/o PuAl fuel rods are planned for 2.0 w/o PuAl with 16% Pu-240 content. Four hundred and fifty of the 1,000 fuel rods expected have been received.

The Critical Facility of the PRP

Approval has been obtained from the AEC for doing experiments in the PRCF with zoned loadings and with irradiated elements in which the fission products have decayed for approximately 30 days. A draft of a supplement to the final safeguards analysis has been reviewed and comments forwarded to the authors.

The tentative schedule for the operation of the PRCF has been updated and more detail added for the first year of operation. The official schedule will be issued when the PRCF startup experiments are begun and will be revised quarterly.

a) PRCF Startup

Approval has been obtained locally for doing the first ten experiments of the startup program. (HW-71214-Supplement). The void, H_2O and D_2O experiments during startup require thimbles for the fuel assemblies. The construction of these thimbles were contracted to offsite vendors. It was last reported that they were 90% complete and shipment is scheduled for the first week in February.

The vertical and horizontal traversing mechanisms have been installed and are working very well. Reproducibility of the positions of the BF_3 chambers is better than ± 1/16 of an inch. Six BF_3 proportional chambers have been installed to monitor the neutron flux during the loading to critical. Four chambers are to be used during all stages of startup. The two extra chambers are 1/4 inch diameter BF_3 chambers and will be used after the 1/2 inch BF_3 tubes saturate. The initial data for inverse multiplication curves were obtained for moderator levels of 60, 70, and 85 inches with no fuel in the reactor. Measurements at 105 inches are under way.

The neutron counting rate as a function of horizontal position has been measured at a height of approximately 59 inches above the bottom of the tank with the moderator level at 105 inches. The counting rate varies from 4200 to 120 counts per minute in going from 30 inches on

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one side of center of the traverse line to 30 inches on the other side. This asymmetry is attributed to the strong γ-ray source of the irradiated fuel elements stored nearest the side having the high counting rate. The (γ,n) reaction with the deuterium is apparently producing an asymmetric and strong thermal neutron source in the moderator.

The IBM 7090 program SWAP has been used to calculate the characteristics of the PRCF at eighteen stages of the load-to-critical. The results of these calculations are sensitive to the manner in which the control rods are simulated. The calculated k_{eff} for a complete two-zone loading, 25 UO_2 fuel clusters inside of 30 PuAl fuel clusters, is 0.99. This is in fair agreement with earlier calculations which were based on the startup of the PRTR.

b) Irradiated Fuel Measurements

The first measurements will use PuAl fuel clusters irradiated in the PRTR up to 84 MWD exposure. This exposure corresponds to approximately a 50% Pu burnup. Estimates have been made of the expected changes in reactivity of the PRCF as a function of exposure of the clusters. The IBM code MELEAGER was used to compute changes in the total fuel fission and absorption cross sections. These fractional changes were in turn used in perturbation calculations to estimate the reactivities. The three-group diffusion code SWAP was used to compute the perturbation coefficients. The expected reactivity change corresponding to 84 MWD exposure is -11 mk. A similar approach is being used to estimate reactivity changes due to exposure for UO_2 clusters.

<u>Phoenix Fuel - ARMF-MTR Experiments with Plutonium Fuel</u>

The details of an experiment to measure the effect of a PuAl sample on the spectrum in the ARMF has been worked out with the ARMF personnel. The results of this experiment will be used in the analysis of the experiment to study the reactivity lifetime of three PuAl samples which contain concentrations of Pu-240 of 6.25, 16.33, and 27.17 weight percent.

The reactivity of the samples has been measured in the ARMF as a function of irradiation in the MTR. Each of the samples has completed seven MTR cycles of irradiation and the ARMF measurements have been made after each cycle. This completes the irradiations and the reactivity measurements.

The ARMF data have been received from all but the last two irradiation cycles. Counting data from Co-Al wires which were used to monitor the exposure in the MTR have also been received from all irradiations except

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those from the last two samples which were irradiated. Other ARMF measurements which remain in addition to the spectrum measurements are recalibration of the ARMF for various boron and plutonium concentrations.

Neutron Rethermalization in Graphite and Water

The manuscript "Neutron Rethermalization in Graphite and Water" has been revised and resubmitted to Nuclear Science and Engineering for publication.

Changes in Resonance Absorption in Pu-240 with Temperature

The comparison of experimental and calculated values of the difference in the changes in the multiplication factor with fuel temperature for 19-rod clusters of low and high exposure PuAl has now been completed. The rate of change of the multiplication factor is different in the two cases because the concentration of Pu-240 is not the same in the two clusters.

The experimental and calculated values agree to within about fifteen percent for the clusters studied. This work is reported in more detail in a forthcoming Physics Research Quarterly Report.

Neutron Thermometers

Counting data from foils which contain 10 mg/cm^2 of Lu_2O_3 and which were covered with various thickness of cadmium have been analyzed. The results will be included in an article for submission to the journal of Nuclear Science and Engineering. The article is written except for the section which will present these results.

Theoretical Scattering Kernels for Water

Program EWGUS (Egelstaff with Gaussian) is in the final stages of debugging. This program will calculate scattering kernels based on Nelkin's model, but with first order corrections for finite energy level widths.

Numerical Integration of Ideal Gas Kernels

A report on the studies of methods of numerically integrating the Egelstaff S-function for an ideal gas has been prepared for the Physics Research Quarterly Report, October-December, 1962.

PRTR Fuel Cycle Analysis

A coupled two-group, multiregion diffusion theory version of MELEAGER has

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been applied to PRTR startup experiments. Failure to get good agreement with gold foil flux traverses has led to a reappraisal of the method, cross sections, etc. A portion of the code dealing with the definition of self-shielding coefficients and the appropriate fluxes to be used therewith is being rewritten.

MELEAGER B (which prepares the cross section library) is being rewritten so that it will use the individual resonance integrals for the various isotopes to facilitate the calculation of the self-shielding factors as a function of the time varying isotopic composition.

PRTR Burnup Experiment

Fuel element 5092 (L_x Pu-Al) has been disassembled and samples cut from three selected rods. This is the last element of the first series of L_x Pu-Al being subjected to destructive analysis. The remaining three elements will be tested in the PRCF prior to destructive analysis.

PRTR Operating Data Analysis

A program, SOURCE, has been written which calculates the slowing down source at various thermal energies from a known slowing down spectrum. The slowing down spectrum is obtained by the GAM-I code applied to an L_x Pu-Al fuel element, suitably homogenized. Cross sections for the thermalization calculation are obtained from the RBU data tape via the BARNS code. These data will be used in calculations using the thermal neutron spectrum code Thermos.

Advanced Reactor Concepts

Analysis of Two Fast Critical Experiments

The fast, clean, critical experiments GODIVA (oralloy), and JEZEBEL (Pu-239) were utilized in a study whereby various cross section data and methods of analysis could be compared.

S-N calculations (S-4 and S-8) were made on both systems utilizing the Los Alamos cross section set[2], the GAM-I nuclear data tape, and the RBU updated version of the GAM-I tape.

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(2) Hansen, G. E. and W. H. Roach, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543, December 1961.

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Diffusion theory calculations were made utilizing the YOM set[3], the ANP-G-2[4] nuclear data tape, and the RBU version of the GAM-I tape, all with and without smallness corrections applied. Since both a diffusion and transport analysis were based on the same cross section data a direct comparison of calculational method can be inferred.

An informal document covering this work is to be issued.

Plutonium Utilization Studies

In preparation for a more rigorous reactor physics study of PuN, PuO_2, UN, and UO_2 as fuels in a compact fast system, a set of 13-group cross sections has been prepared for use in transport calculations on the S-X code. The GAM-I code was used to obtain cross sections for the 10 groups down to 1 ev. For the three bottom groups, hand averaging of data was used.

Phoenix Fuels for Compact, Water Moderated Reactors

Beginning of life calculations for the plutonium-fueled Zr-H_2O assembly (Mark I) now include 5, 10, 15, and 20% Pu-240 initial fuel charges. The initial reactivity behavior over this entire range of Pu-240 contents is quite similar.

As a check on the treatment of the Pu-240 resonance capture, flux depression factors for input into the group self-shielding code GROUSS are being prepared. In the initial burnout calculations--now in progress--the simple [redacted]

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Code Development

RBU

Modifications have been made to the boundary reflection calculation and the Russian Roulette in the Monte Carlo. The boundary reflection change provided a 50% time reduction and decreased the memory requirements by about 15 words. Additional changes in the Monte Carlo beam-processing logic have guaranteed significantly shorter running time. Finally, modifications were made causing the neutron reaction-balance tallies printed on-line during the Monte Carlo execution to include only equilibrated neutrons. This should provide a clearer description of Monte Carlo progress, and lend more significance to some of the expectation values now computed.

The final RBU documentation is under way with the rough draft of the second of three planned system documents, II RBU - System Structure and Program Description, now 15% complete.

CALX

Three cases have been successfully run using CALX, a constant flux burnup, a constant power burnup, and a constant flux recycle case all with time limit and no initialization. The problems of long running times and sudden large changes in concentration appear to have been rectified.

Subroutine SIGMA which used an erroneous power value for equilibrium xenon calculations was corrected.

Subroutine SETZ which does the initialization was coded and is currently in debug.

Table for Resonance Integral Calculation

The extension by numerical interpolation of the $J(k,\zeta)$ tables for calculation of Doppler broadened resonance integrals was completed by the procedure described last month. Augmented tables of $J(k,\zeta)$ for k = 5.0 (.5) 20.0 and ζ = 1.1(.1) 1.5(.25) 2.0(1.0) 5.0 were computed and distributed to prospective users.

Instrumentation and Systems Studies

A simulation of the PRTR was programmed on the EASE analog computer for the purpose of conducting further nuclear excursion studies as part of a PRTR hazards analysis. For this study the driving function represented an

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accident involving the incorrect addition of moderator water starting at source level. The required information was obtained and forwarded to the customer.

Hazard analysis of a postulated nuclear excursion of the Plutonium Recycle Critical Facility involving light water moderation was performed on the EASE analog computer. The purpose of the study was to obtain response characteristics for comparison with calculated data. The study also indicated ways in which more detailed studies might be conducted. The computer study was successfully performed and the data were provided Nuclear Safety Studies Operation for further analysis.

Spectrum and cross-spectrum measurements were made at the PRTR for further evaluation of noise analysis methods of determining reactor parameters. Difficulty is being experienced in obtaining reproducible data, the cause of which is presently receiving further study.

Calculations were made of the radiation intensity expected at the surface of 350 gallon spherical shipping casks containing particular radionuclides. Estimated requirements and costs were established for the high level in-cell monitors, low level in-cell monitors, general radiation area instruments, gross gamma filter counters, and airborne contamination monitors.

A general review was made with new PRTR personnel regarding the PRTR Fuel Element Rupture Monitor to acquaint them with the system. In addition, a study was started for the possible employment of gas chromatography instrumentation system for PRTR fuel failure monitoring.

All specific design and layout work was completed regarding three scintillation solid-state liquid effluent monitors for gamma-emitting radionuclides at PRTR. A cost estimate was nearly completed for on-site fabrication of the three instruments. The prototype instrument continued to perform correctly at PRTR.

A six channel eddy current instrument capable of monitoring PRTR fuel element vibrations is being developed. The completed system will be composed of six separate channels of electronics with remote readout on a strip chart recorder, and a direct visual presentation on a cathode ray oscilloscope. An eddy current sensing coil will be built into each of the twelve outer fuel rods of a specially designed, nineteen rod, PRTR fuel element assembly. This system will provide data from three different elevations along the length of the fuel assembly. Fabrication of a prototype channel has begun, and the development of a breadboarded channel is nearly complete.

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HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

A new cost estimate of $2,350,000 was received for the High Temperature Lattice Test Reactor, HTLTR. Since this is within the $2,500,000 requested by the AEC, the design analysis is proceeding without drastic reduction in scope. A draft of a project proposal for funds for detailed design and advance procurement has been prepared and is being revised for submission in February.

A number of items have been changed or revised. The size of the graphite stack has been increased from a 9-foot cube to a 10-foot cube and reductions have been made in the size of the reactor room and other items. Safety rods will be pushed from below to provide graphite followers and reduce building height. The cadmium skin will be replaced by boral to reduce shielding requirements. If cadmium were used, the capture gammas would dominate the dose rate outside the shield. With boral the shielding requirement is determined by the fast neutrons from the core. The neutron generator will be in the basement with its beam tube going up through the concrete floor, and the neutron detecting instruments will be on top of the reactor. Containment and monitoring of radioactivity will be determined by the possibility of plutonium leakage from test fuels. Fission product production will not be serious for the low design power level. The reactor room will not have to be gas-tight. Absolute filters will be provided in the circulating nitrogen line and part of the building exhaust. The exhaust stack will be 40'-50' high.

NEUTRON FLUX MONITORS

Plutonium and uranium samples planned for reactor irradiation experiments are not yet completed. However, a new set of cobalt samples is being prepared for determination of sample weights needed for the evaluation of the spectral parameter (r). Twenty weighings were made of the aluminum backing plates prior to cobalt application. Similar weighings will follow the cobalt addition to obtain sample accuracy. A general background report, which summarizes the theoretical bases for the experiment, was prepared and issued. Tests to evaluate the parameters r, T, and ø have been scheduled for February.

Consideration of alternate methods of obtaining in-reactor neutron flux measurements has led to a novel concept employing the Boron-11 isotope arranged in a simple configuration which permits measurement of the electron current produced by beta decay of Boron-12, the latter isotope being produced from a B-11 (n,γ) reaction. Significantly, the new method appears to circumvent burnout problems of other arrangements, and has the further advantage of a 1000 degree C operating temperature. An invention disclosure

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covering these concepts in detail has been submitted.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

The fabrication of the prototype multiparameter eddy current testing device has been completed and the device is ready for evaluation tests. In the study of the diffusion of eddy currents, it has been shown that currents in relatively simple multimesh inductively coupled circuit models exhibit the same general behavior as eddy currents observed in solid and liquid metal. Basic circuit development work has been initiated to give further improved performance in both single frequency and multiparameter eddy current test equipment.

The prototype multiparameter eddy current tester is completely wired and ready for evaluating tests. Most of the two stage plug-in amplifiers had to be rewired to reduce stray capacitance and coupling. This rewiring has reduced the tendency of the amplifier to oscillate which became apparent when the reactive bandpass filters were connected to the amplifier inputs. The interconnecting cables have been completed and all units are mounted in one equipment rack. Preliminary tests indicate the signal generation section and the frequency separation section (except for the 3 Mc channel which required some modification) are operating properly.

Work continues on the eddy current diffusion tests. Additional theoretical work has been undertaken to determine the proper magnetic field produced by the driving coils. In addition, an analog computer is being used to evaluate the validity of some proposed simplified models which can be used to simulate the conducting medium that contains the eddy currents.

The transient solutions of mesh currents in a five mesh inductively coupled circuit have been shown to be similar to solutions obtained empirically for eddy currents in a solid metal. The general natures of these solutions are similar, and the apparent "time delay" effects noticed in flow of eddy currents in the solid or liquid metal case are also observed in the multimesh inductively coupled circuit. The nature and cause of this "time delay" are being explored by a study of energy and time relationships in inductively coupled circuits having 2, 3, 4 and 5 meshes.

Basic circuit development work is under way to devise improved circuits for use in both single frequency and multiparameter eddy current tests. Objectives of this work are to simplify some recently developed circuits, retaining their intrinsic advantages and to improve stability.

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Heat Transfer Testing

The end closures of three N-fuel elements were heat transfer tested. These tests were made in an attempt to gain an understanding of variations in ultrasonic reflections obtained from the end closures. Apparent variations in the heat transfer properties were detected. Further tests are being made in an attempt to minimize any possible errors in the heat transfer results due to surface irregularities.

Zirconium Hydride Detection

Prototype eddy current equipment suitable for laboratory and field tests on hydrided Zircaloy-2 samples is being developed. Eddy current indications are observed on laboratory test samples containing as low as 100 ppm hydrogen. A degree of ambiguity exists, however, in that similar indications are obtained on selected samples which contain no hydrogen. Presumably, the anomalous indications are from sample variations such as surface pitting, metal cold working, grain structure variations, etc.

Special emphasis is being devoted to determining the cause of the undesired signal responses such that they may be circumvented by suitable electronic techniques.

With this objective, steps have been taken to obtain extensive test samples so more detailed and representative studies can be made.

Fundamental Ultrasonic Studies

The effects of diffraction, attenuation, and boundary waves are being studied to more accurately explain basic ultrasonic propagation models. In order to approach the problem systematically, the most simple attenuation and boundary wave behaviors associated with plane waves incident on a liquid-liquid boundary were studied analytically. By means of these studies the physical description of plane boundary waves in the presence of attenuating and non-attenuating liquids was made more evident. The general wave propagation as influenced by the attenuation in the liquid was also examined. It was found that the ordinary Snell's law of refraction must be modified as a result of the attenuation. The modified expression was found to be frequency dependent and also a function of the angle of incidence. Similar wave behavior is expected for liquid-solid boundaries. The simple liquid-liquid boundary studies are expected to facilitate interpretations of the more complicated liquid-solid cases.

Ordinary refraction occurs when plane sound waves, continuous in time and space, are incident on a liquid-liquid boundary and when neither the

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incident nor the refraction liquid is attenuating. If the refraction liquid has a higher compression wave velocity than the incident liquid, a critical incident angle is possible. As the angle of incidence is increased, initially ordinary plane wave refraction results and the pressure amplitude along the refracted wave front is constant. Since the refraction liquid has the higher wave velocity, the angle of refraction is greater than the angle of incidence. This means that there becomes a critical angle of incidence at which the refraction angle is ninety degrees. It is well known that total internal reflection occurs when the incident angle approaches or exceeds the critical angle. However, the solution to the wave equation coupled with the boundary conditions show that an inhomogeneous boundary wave exists together with the reflected wave. This inhomogeneous compression wave propagates along the boundary and has plane wave fronts. However, the amplitude along the wave front is no longer constant, but decreases exponentially with depth into the refraction liquid, this being the reason for the inhomogeneous classification. There is also an attendant phase shift in the reflected wave as the result of the inhomogeneous wave.

The effect of attenuation, which adds much complexity to the analysis, is now being studied.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Early objectives of the program to develop improved methods of ultrasonically inspecting fuel element sheath tubing were: (a) define optimum test parameters such that maximum beneficial use could be obtained of existing inspection equipment, and (b) extend basic understanding of the behavior of waves propagating in tubular geometries with a view toward putting the experimental test on a firm technical basis. Present emphasis is on documenting results which generally meet these objectives. Terminal report, "Recommended Practice for Immersed Ultrasonic Testing of Zircaloy Fuel Element Sheath Tubing Using Lamb Waves," is in final preparation. This report establishes tentative test specifications outlining optimum instrument parameters--such as test frequency, pulse repetition rate, transducer type, beam entry angle, etc. It also defines proper calibration and operational procedures for the inspection of tubing in the range of .030 inch wall thickness. A recently issued document, "Ultrasonic Lamb Wave Models for Nondestructive Testing," as well as an earlier report, "Ultrasonic Testing of Zircaloy Sheath Tubing for Fuel Elements," are devoted to the second objective. The theoretical effort has provided an increased understanding of the fundamental nature of ultrasonic wave propagation in thin walled, curved test specimens. Important new concepts

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have evolved from this effort which are of general interest to the ultrasonic testing field.

"Ultrasonic Amplitude Response as a Function of Notch Depth and Other Effects in Zircaloy Sheath Tubing" and "Masking as a Method for Controlling Beam Configurations of Ultrasonic Transducers" are other documents nearing completion.

Continuing efforts are aimed at extending the work done to date to tubing with wall thickness of 0.045 inches and less. Empirical and analytical studies will give emphasis to the effects of the thinner sections on ultrasonic wave behavior. Increased emphasis will be given the development of specialized instrumentation for application to the inspection of thin walled tubular components.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Progress was made in the analysis of the large quantity of atmospheric dispersion data collected at Hanford, Cape Canaveral, and Vandenberg Air Force Base. Comparisons between the average dispersion conditions at the three sites, after accounting for seasonal and diuranl effects and grouping the experiments according to similarity in meteorological conditions, showed significant similarity. This was particularly evident in the Cape Canaveral and Vandenberg data where the sea-land wind regimes were similar. The narrowest plumes observed were during persistent wind directions associated with wintertime large scale pressure gradients. These periods represented typical frontal-type weather conditions as contrasted to the sea-breeze circulation that dominated the summertime wind pattern.

During the daytime winter experiments, a weak sea-breeze superimposed on the gradient flow resulted in greater wind direction variability, causing greater horizontal dispersion. During the summertime sea-breeze regime, small scale local conditions produced nonpersistent winds, which resulted in greater dispersion. Both daytime and nighttime dispersion characteristics on either coast were similar for areas of level terrain. However, over rough Vandenberg terrain, the plume width for nighttime releases was larger at the greater distances as the flow around terrain features became more pronounced with increased atmospheric stability.

The summer nighttime releases at Hanford resulted in comparable average plume widths to those observed on the coasts during the same season. At large distances, the Hanford results approach the extreme rough terrain

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ghttime values at Vandenberg, indicating a possible similarity in ...ility-terrain effect. The daytime experiments at Hanford reflect the ..hanced dispersion during unstable atmospheric conditions.

..e prototype model of the real-time air sampler for use with fluorescent tracer studies was operated successfully at a distance of one mile from a ground source. Previous trials had suggested that the sensitivity of this sampler would be adequate at this distance, but this test marked the first actual field use of the device at this distance. This success extends the range over which the sampler promises to be useful in investigations of atmospheric dispersion processes.

Dosimetry

In a continuation of the study of the difference in potassium results between the Shadow Shield and the regular whole body counter an extensive calibration and intercomparison of the two counters was carried out. The results indicate that the calibration of the Shadow Shield had been off 4.5%, or slightly more. We were looking for a slightly greater discrepancy. Incidental to this study the fractional excretion rates of potassium and cesium from the body were measured; that of cesium was slightly greater than that of potassium.

Radiation Protection Operation was assisted further in the preparation of their mobile shadow shield. They were also assisted in the preparation of new plutonium wound counters using the low noise photomultiplier tubes reported last month. The wound counter in use for the past several years employs a tube that was an exceptional one of its type. The new tubes give as good or better performance.

A study is under way to develop a method of phosphorus-32 counting that can be employed in the RPO mobile facility. Simple shadow shielding techniques are not applicable because the energy degradation on scattering is not enough to remove background radiation from the energy range of interest.

The study of the background of X-ray scintillation counters continued. The background rate appears to be independent of crystal thickness below 1 mm. Most of the background inside a good shield is due to radioactivity or radiation absorbed in the glass on which the crystal is mounted. Some of this background is due to Cerenkov radiation in the glass.

The Van de Graaff was still limited to voltages below 1.7 Mev but otherwise operated satisfactorily during the month.

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The use of nitrogen in our gas target was investigated, because N(d,n) gives monoenergetic neutrons above 5 Mev, a region not generally accessible to us now. The yield was too low to be useful, but higher Van de Graaff voltages may result in useful yields.

Further adjustment of the pulse shape discrimination circuit made it possible to reject 98% of the pulses from cobalt-60 while still detecting proton pulses down to 0.6 Mev. Several spectra collected earlier were remeasured with targets free of carbon contamination.

The shop is engaged in making a new set of tissue-equivalent chambers for neutron studies.

The gamma-ray calorimeter was prepared to make a measurement of the specific power of the radiation emitted by promethium-147.

The investigation of the heat amplifier was resumed. Earlier results were confirmed. Determination of the optimum parameters for room temperature operation is in progress.

Instrumentation

Improved methods of charging the center rod of pencil-type ionization chambers have been favorably evaluated experimentally. The technique provides constant starting potential and thus assures proper alarm signalling. Additional prototypes are being fabricated for further testing.

Further experiments were carried out for the analysis of smear samples and air filters for entrapped uranium and thorium. The relative health hazards of the two are such that proper determination is important. Alpha energy analysis tests, using surface barrier solid state diode detectors and a multichannel analyzer, demonstrated the feasibility of the method for a series of smear samples obtained from the 306 Building.

The drawings for both the Mark II and Mark III scintillation monitors were completed and approved. The Mark II instrument provides both logarithmic and linear four-decade response from less than 1 mr/hr to 10 r/hr. The Mark III instrument, which has now completed four months of continuous operation without malfunction or need of adjustment, provides a logarithmic six decade response from 1 mr/hr to 1000 r/hr.

A modified luminescent particle detector, for use in field studies of particle movement in the atmosphere, was used in a series of field tests with excellent results at a distance of one mile from the Atmospheric Physics tower. Calculations show that the continuous monitor should be

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effective to two miles distance. A method was devised for direct in-field testing of the phototube detector using a Promethium-147 activator source and a coating of fluorescing pigment material. Three final models of the instrument will be fabricated on-site for field experimental use.

Development of the experimental portable mast system is progressing with main emphasis on detailed electronic circuit development. Circuit designs were completed and favorably evaluated for various electronic functions during the past month. Engineering drawings are being prepared to consolidate work accomplished to date.

Check-out and wiring continued on the 400 channel analyzer during the past month. Most of the control portion of the logic and all of the address and data registers are completed and appear to be working properly.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

The mass spectrometer for this program provided isotopic analyses of program samples received during the month in accordance with current goals.

Development work was resumed on a scintillation-type ion detector. In this detector the ions which are momentum analyzed in the mass spectrometer are post-accelerated to about 60 kev onto an aluminum target. The secondary electrons produced at the target are accelerated in the same electric field into a plastic scintillator and counted with a photomultiplier. The characteristics of the response of this system to single electrons incident on the scintillator are under study on the ion test bench.

TEST REACTOR OPERATIONS

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

The experiment to determine k_∞ data for Pu-Al (20% 240), 19 one-half inch fuel element clusters in $10\frac{1}{2}$-inch graphite lattice continued throughout the month. Null reactivity measurements and foil activation traverses were made.

A key switch was installed in the leveling ring power circuit as an additional safety measure to prevent unauthorized moving of the leveling rings.

Operation of the TTR was on an intermittent basis during the month. There was one unscheduled shutdown caused by electronic failure. The TTR was made available to the University of Washington Graduate Center six nights during the month.

CUSTOMER WORK

Weather Forecasting and Meteorological Service

Consultation service was rendered on meteorological and climatological aspects of 1) oxides of nitrogen release in 300 Area to NRD, 2) low altitude sampling of fallout to CR&D, 3) environmental consequences of reactor accidents, 4) shipment of large quantities of fission products, and 5) temperature and humidity restrictions for punch card data storage to AEC. Annual climatological summaries were prepared for 1962 and new compilations prepared of daily normal temperatures for the period 1931-1960 and extremes for the period 1912-1962.

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

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	93	84.5
24-Hour General	62	85.2
Special	198	89.4

Temperatures during January averaged below normal. There was no measurable precipitation for the first 29 days. However, light snow on the 30th and heavy snow on the 31st raised the monthly total to 0.95 inch, which was only 0.13 inch short of normal.

Instrumentation and Systems Studies

The automatic conveyor-type protective clothing contamination monitor operated another month without malfunction. A detailed maintenance and instruction manual was issued.

The field-model coincidence-count alpha particulate air monitor, which has a detection sensitivity of 15 MPC-HRS for Pu-239, continued to perform correctly in application at the 308 Building.

To improve the detection of plutonium in chemical process line solutions, a special solid state circuit was developed for use with glass scintillator

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alpha detectors. In addition, a special light-level detector with a solid state circuit was developed to provide an alarm signal for buildup of plutonium oxalate powder in a chemical process hood. A study was also carried out regarding methods of measuring plutonium content of sealed [illegible] containers. It was concluded that the 384 kev photon from Pu-239 provides a suitable measurement source, and accurate determinations of content should be possible if the contained mixture is homogeneous. Technical reports were issued regarding all three projects.

A design cost estimate was prepared for an instrumentation system for use in measuring uranium metal enrichment (U 235). The work was for Fuel Fabrication Development, HL.

A general design and cost estimate was made regarding a serial number instrument to prepare paper tape leaders or instructions for Redox laboratory multichannel analyzer data. The instrument is to provide instructive information for the IBM-7090.

A hazards analysis of a maximum conceivable incident at the Hanford Test Reactor was programmed for study on the EASE analog computer. The problem assumed inadvertent reactor charging of a stringer of enriched uranium fuel. Although simulation tests were favorable, results predicted a reactor period three times longer than mathematically predicted values based on extrapolated experimental data. Further work on this program is being deferred until computer time becomes available.

A paper tape punch readout was completed which works in conjunction with a tensile machine to automatically record stress-strain data. The equipment transforms recorder shaft position into digital form, which is then punched on paper tape.

Preliminary system design and cost estimates were developed for a proposed Redox counting room installation. The system is required to generate a punched paper tape containing about 200 characters of information needed to instruct the IBM 7090 in the analysis of pulse height analyser data.

Optics

Evaluation of components for a high temperature microscope being assembled for Ceramic Fuels Operation has shown that a locally fabricated objective mirror has greater resolving power and is otherwise superior to a commercial unit. Specimen manipulation modifications are being incorporated into the system which was originally developed by Los Alamos Scientific Laboratory.

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A bore camera for internal inspection of fuel elements by Testing Methods Engineering, NRD, was completed with the installation of a Polaroid camera and shutter. Preliminary evaluation indicates the polaroid film is not as satisfactory as the Royal Pan film previously used.

A computer program has been completed which was developed to aid the calibration of traverse mechanisms with which reactor process tube distortions are measured. Computer results indicate that the traverse mechanisms data can be made to fit known tube displacements with a standard deviation error of less than 0.040 inches over 34 feet of tube length.

Laser experiments utilizing a ruby crystal and a xenon flash tube have been performed in support of Ceramic Fuels Development Operation. Initial tests were successful in that flashing of the xenon tube caused the crystal to emit brilliant, red light beams. The beam was not strictly coherent, however, and failed to produce the desired heating effects. Improved results are anticipated as xenon tube power input is increased.

A Bausch and Lomb Model L camera was modified to permit use of five inch wide roll film in photographing fuel element cross sections at moderate magnifications. The large number of pictures required to make large composite pictures warranted the use of automatic film change equipment. The modifications incorporate an electrically driven film change mechanism in the standard Model L macro camera system.

<u>Physical Measurements</u>

Work on the evaluation of micro-displacement readout systems to be used by Reactor Metals Research for in-reactor creep measurements has been confined to data analysis during the month of January.

Analysis of the data obtained during the calibration testing of a Physical Science Corporation Model TI-A-03 transducer used with a Model 801-D excitation and readout system was completed, and the results submitted to the customer in an evaluation report, PMO Memo 63-2. The "Zero Shift" data for this system have been processed through the 7090 computer and are presently being correlated for submission as a supplement to the basic evaluation report.

Results indicate that this system is capable of 100 micro-inch, or better, reproducibility at temperatures to 400°C, provided proper data compensation methods are employed.

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Physical Testing

Inspection of the first shipment of K reactor process tubes is continuing at an accelerated rate. A total of 7,031 separate tests of 2,737 tubes (representing 71,526 feet of material) have been performed to date. In addition, a wide variety of testing services were provided twenty-six HAPO customers and a number of AEC contractors. Advice was given on seventy different occasions on general testing theory and applications.

In the course of testing the Zircaloy process tubes for K-reactor retubing program, cracks were found by the fluorescent penetrant test. The cracks are apparently a result of the VanStone flange fabrication. Sample specimens of K-reactor tubing were flanged and subjected to initial tensile testing to determine the effect of the cracking on the mechanical strength. Results are pending completion of the tests.

A 400 KV X-ray unit is being procured to extend the Physical Testing Operation's ability to provide increased service to present and potential customers.

Development of an eddy current test to locate radial cracks in reactor Parker fittings continued. Emphasis has shifted from detection of cracks in the threaded area to detection of the cracks in the thread relief area. Improved results have been achieved by using improved differential coils at frequencies above 400 KC. Instrument sensitivity capable of resolving 0.125 inch long by .010 inch deep thread relief defects are believed possible with minor improvements to existing developments.

An unusual service involving the application of microscopic analysis techniques for the measurement of particulate contamination of hydraulic fluid was performed. The test provides assurance that fluids employed in reactor control rod and elevator mechanisms are free of harmful contaminants.

ANALOG COMPUTER FACILITY OPERATION

The analog computer problems considered during the month include:

1. D-reactor automatic control.
2. NPR pressurizer injection system.
3. NPR graphite coolant gas composition.
4. PRTR nuclear excursion study.
5. PRTR critical facility.

The specifications for a new analog computer were completed. These specifications describe a 156 amplifier computer with iterative capabilities and

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an automatic digital input-output system. The purchase requisition was also written.

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

GEDA	EASE	
35	175	Hours Up
25	25	Hours Scheduled Downtime
0	0	Hours Unscheduled Downtime
140	0	Hours Idle
200	200	Hours Total

INSTRUMENT EVALUATION

Evaluation tests were completed on a commercial solid-state operational amplifier and an evaluation report was written. The amplifier did not meet the performance claims of the manufacturer, and it finally failed completely during the testing.

Evaluation testing was completed on one production prototype, of seven being fabricated off-site, combined alpha-beta-gamma scintillation solid-state hand and shoe counter. A large number of fabrication errors were found; however, some of these were caused by errors on the blueprints. A number of circuit and detector modifications were devised and incorporated to improve the performance of the instrument. All errors will be corrected and all circuit improvements will be incorporated in the remaining six instruments still at the off-site fabricating plant.

Blueprint and purchase specification information was provided to Radiation Protection, HL, for the off-site purchase of a number of solid-state circuit loudspeaker units for use in place of headphones with portable radiation survey instruments.

RS Paul

Acting Manager
PHYSICS AND INSTRUMENTS LABORATORY

RS Paul:mcs

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CHEMICAL LABORATORY

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Ground-Water Temperature Studies

The temperature of the ground water beneath all reactor areas at Hanford has increased significantly since the start of operations. This has resulted mainly from the relatively large volume of thermally warm reactor effluent cooling water percolating downward into the ground water from the retention basins and local disposal sites located within each area. During 1962 a study was made to determine the distribution of thermally warm water beneath each of the reactor areas. Early this month the last series of temperature measurements was made in the wells located in the region of the 100 Areas, and the field work phase of this study was brought to a close. Temperature profiles in all wells located in this region, which were measured nine times during 1962, are now being evaluated. In addition to delineating the extent of movement of the thermally warm ground water beneath each reactor area, it is expected that these data will provide information from which an estimate of the rate and direction of ground-water movement beneath the areas can be made. Interpretation of this information should also reveal seasonal trends which occur in the ground-water temperature patterns.

N Reactor Crib

A recently developed computer program has provided a better estimate of water flow to be expected from the N Crib. In light of this information estimates of fission product release from the facility to the Columbia River were revised.

The only fission products expected to reach the river in significant concentrations are I-131 and Sr-90. A usual rupture with a loss of 50 g of uranium is calculated to release 0.159 curies of soluble I-131 and 1.8×10^{-3} curies of soluble Sr-90. The expected arrival time for I-131 is 11 days following a given diversion. A maximum I-131 release of 2×10^{-2} μc/sec and 2 μc/sec is expected for losses of 50 and 5000 g of uranium, respectively, within hours after breakthrough. Between diversions, at the assumed frequency of one per month, the release of I-131 will decrease by a factor of 40.

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Strontium-90 breakthrough is expected to occur about 3.5 years following the initial diversion. Because of its slow rate of radioactive decay, the rate of Sr-90 release to the river will not decrease significantly between subsequent diversions. The maximum release of Sr-90, 3.5 years after the initial diversion and based on a 50 g loss of uranium, is 4×10^{-6} μc/sec. Based on the assumption of continuous use of the N Crib at the estimated frequency and severity of failures, Sr-90 release rates of 5×10^{-4}, 1.5×10^{-3}, 2×10^{-3}, 3×10^{-3}, and 4×10^{-3} μc/sec are anticipated 1, 5, 10, 20 and 40 years, respectively, after the initial breakthrough.

Cation Exchange Capacity of 100-Area Subsoils

The cation exchange capacity of 100-Area subsoils was determined for all areas except 100-K for which no samples were available. The adsorption of ammonium ion from neutral 1 M ammonium acetate was used as a basis for the exchange capacities measured.

In general, the specific cation exchange capacity of the sediments (meq/100 g) increased with the distance inland from the Columbia River. Some irregularities were evident at 100-F Area, and differences were slight at 100-H Area. The increase in effective cation exchange capacity (meq/ft^3 x depth to ground water) was even more pronounced with the distance inland from the river because of the increasing depth to ground water.

Based on the cation exchange capacity of the local subsoils, the 100 Areas fall into two distinct groups, i.e., those with a high and those with a relatively low cation exchange capacity. Soils underlying the western-most 100 Areas, Areas B and D, have a specific cation exchange capacity of about 4 meq/100 g while those areas to the east, Areas H and F, have cation exchange capacities of about 2 meq/100 g. The exchange capacities of soils beneath 100-K and 100-N Areas are expected to be similar to that beneath 100-B and D Areas.

In the selection of sites for future crib construction in the 100 Areas, the distance inland from the river is an important consideration. Inland sites offer the advantages of longer effective soil columns and additional time for decay of poorly adsorbed radiocontaminants before release to the river.

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Fission Product Release

A special hazards study on irradiated "B Metal" shipments was initiated on the request of IPD. Experiments are being planned and equipment designed and fabricated for studying releases at high temperatures. This work will result in some delay of the study on fission product release from production-scale fuels.

Effluent Water Radioisotope Studies

The corrosion inhibition of aluminum fuel element jackets which were numbered with marking inks prior to reactor irradiation prompted laboratory studies of these inks and the adsorption properties of ink-coated aluminum surfaces. Neutron activation analysis of the inks indicated no inorganic constituents, thus no difficulties from neutron activation products could result from the inks if tested in the flux zone of a reactor. A two-day reactor test was made by exposing four ink-coated super-pure aluminum test pieces together with two uncoated pieces in the central flux zone of D Reactor. Since these pieces were in a process tube containing only dummy elements, the water temperature was low, the adsorbed radioisotopes were very low, and no significant difference was observed between the treated and untreated surfaces. Laboratory studies do indicate a decrease in adsorption of arsenate ion, so these inks and their individual components will be evaluated further in reactor tests.

Some aluminum coupons were anodized using chromic, sulfuric, or oxalic acids as electrolyte. These coupons were tested for adsorption of arsenate together with anodized and sealed coupons, autoclaved coupons, and silicate-treated coupons. Anodizing, with or without sealing, was found to increase adsorption. The autoclaved coupon adsorbed the least. The anodized coupons did resist corrosion well and will be tested for long-term effect on adsorption of radioisotope parent materials.

Downstream perforated spacers were obtained from the two reactor tubes using deionized water and from the two pilot plant tubes using standard process water. The film from measured areas of each of these spacers was removed, dried, weighed and analyzed. The films were about 80 percent aluminum oxide and 18 percent iron oxide in all cases. The film from the deionized water tube spacers was less than one-half the amount from the pilot plant tube spacers although the corrosion rate is greater in deionized water. The iron noted on the deionized water tube spacers apparently derives from the steel water lines between the deionizer and the reactor.

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The deionized water tube film is the intense orange-red iron oxide color and readily redisperses in water to form a colloidal solution with particles having a positive 20 mv zeta potential and a size range from 200 - 1200 millimicrons.

SEPARATIONS PROCESSES

Disposal to Ground

During recent months the gross beta-emitter concentration in the ground water beneath the Purex A-10 process condensate crib increased markedly from 1.5×10^{-5} to 2×10^{-3} μc β/cc. The beta activity is essentially all radioruthenium; the Sr-90 concentration in a recent sample was $< 7 \times 10^{-8}$ μc/cc. This increase is directly attributable to a comparable increase in the radionuclide content of the process condensate arising from a change in the Purex Plant (on a test basis) from double-stage to single-stage distillation in the acid recovery process.

Vertical wells in the Purex 241-A tank farm were monitored for the first time with the scintillation probe to detect possible waste tank leaks; previous monitoring was done with the less sensitive GM probe. The results indicate possible line or tank leaks on the north side of the 101 tank and on the south side of the 106 tank. Further evidence of leaks can best be obtained by reprobing the wells at relatively short-time intervals to detect any progressive increases in counting rates. Well drilling samples from zones showing high count rates were counted with the laboratory 3-inch well crystal; none of the samples showed significant amounts of radioactive material. Thus, if leaks occurred; (1) they occurred after the wells were drilled; (2) the contamination has not reached the wells but may be up to five feet from the wells; or (3) contaminated soil encountered in the well drilling was not sampled in getting the soil grab samples at five-foot drilling intervals.

Iodine Removal Processes

Iodine-131 deposition on silver-plated tube walls was shown to follow the diffusion equations of Browning and Ackerly[1] quite well at low Reynolds Numbers (up to about Re = 300). For more turbulent conditions deposition is greater than predicted by diffusional equations, yet not as great as predicted by mass transfer to a wetted wall tower equation derived by Sherwood and Pigford[2]. The

1. Browning and Ackerly, ORNL-3319, p 44. 6/30/62.
2. Sherwood and Pigford, "Absorption and Extraction," McGraw-Hill, p 77. (1952)

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deposition may be lower because turbulence may not have been fully developed in the experiments performed locally. In fully turbulent flow, the silver-plated wall may not, in fact, act as a perfect sink, thereby also contributing to lower deposition than predicted. More data at even higher Reynold's Numbers should be obtained; however, no extensive work is planned at this time.

Release of I-131 from the walls of a stainless steel tube to a dry air stream was found to range from an initial rate of 0.8 percent per hour to about 0.1 percent per hour after 42 hours. These results contrast with a previous observation that I-131 would diffuse at a fairly high rate from the wall to stagnant air in a tube. No further work is planned in this area for the immediate future.

The low I-131 retention efficiency of charcoal taken from a halogen trap in a Hanford Laboratories building exhaust was further investigated. The charcoal which had been in service for two and one-half years was found to be inefficient due largely to surface deterioration or blocking of adsorption sites. Retention efficiency of 94 percent was restored through mild abrasion by shaking, then washing with water. This efficiency is still significantly lower, however, than that of the replacement charcoal, which was found to be 99.99 percent efficient.

Solid State Electromigration

A specimen of 99.9+ percent aluminum traced with Fe-59 and fabricated into a strip 3/7 inch x 0.030 inch x 5-3/4 inch was electrolyzed with a current of 140 amperes at 0.4 volt for 194 hours. This current produced a temperature in the center of the specimen of 530 ± 20 C. No migration of Fe-59 tracer was observed. This negative result is interesting if one compares the behavior of Fe-59 in cerium where about 10 percent of the Fe-59 migrated in four hours at less than 1/3 the current density employed here. In both cases the Al and Ce specimens were electrolyzed at temperatures which were approximately equal fractions of the melting points of the pure metals.

The reason for the differing behavior of Fe in these two matrixes appears to be a function of the manner in which the Fe is present in the matrix metal. In aluminum, Fe is soluble in the solid state up to a limit of almost one atom percent and occurs as a substitutional impurity. In cerium, on the other hand, the solid solubility of Fe is extremely low and it is mainly present as an intergranular impurity. Intergranular diffusion, in general, is

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more rapid than lattice diffusion by factors of 10^3 or much greater. Further, from the equation for the contribution to conductivity, σ, of the migrating ions,

$$\sigma = Ne^2/_{kT} \cdot D$$

it is apparent that the magnitude of the diffusion coefficient, i.e., the nature of the diffusion process, can have a determining effect on the effectiveness of solid state electromigration.

Analytical data for Fe and Cu by wet chemical analysis are now complete for the 140-hour run on cerium at 600 C (70 amps at 0.85 V). Data for Fe at a concentration of 250 ppm confirm the Fe-59 tracer results. Copper at a concentration of 170 ppm does not appear to migrate.

This result confirms spectrographic evidence from the 70-hour run that copper does not migrate. In view of Henrie's report (on cerium containing an initial 120 ppm Cu) that "migration of Cu is readily observed at 50 to 100 hours", a further check is being made. A specimen of Ce has been marked by imbedding a small fleck of Cu in its surface near the center of the specimen. Using tantalum protected brass clamps, this specimen was electrolyzed for 146 hours at 600 C (60 amps - 0.7 volts). The copper rapidly diffused into the cerium specimen leaving a hole to mark its original position. It is hoped that chemical analysis will show whether the diffusing copper was affected by the electrical current.

Demonstration of the Use of U(IV) in the Purex 1BX Column

The 1BX Column demonstration runs in a miniature pulse column have been discontinued because of a lack of sensitivity to flowsheet changes. Plutonium losses to the organic 1BU stream were nearly constant (between 8 and 11 percent) over a wide range of operating conditions which included column diameter, stream flow rates, and reducing agent [iron(II) and uranium(IV)]. It is tentatively concluded that the factor controlling plutonium loss in these runs was the rate of plutonium transfer from the organic phase. Residence time of the organic phase in these small columns is quite short (15 - 20 seconds) and is almost independent of operating conditions. Current studies are seeking to establish the rate of oxidation of Pu(III) to Pu(IV) in the 1BS column at various nitric acid and hydrazine concentrations in the aqueous phase.

Recuplex Air Pulser

The Recuplex air-driven pulse system generator has been installed on a column currently being used for waste extraction studies. The

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air pressure (0 - 15 psig) above the liquid in the pulse leg of the U-tube column is adjusted over a frequency range of from 0 to 70 cycles per minute by two cam operated poppet valves. A variable speed DC motor drives the cam shaft. A pressure control valve holds the pressure in the accumulator at a near constant adjustable value.

Shakedown runs of the pulser equipment were completed. The pulser will be evaluated for use in waste extraction processes as well as for use in the New Reclamation Facility.

WASTE TREATMENT AND FISSION PRODUCT EXTRACTION

Cesium Precipitation Studies

Several laboratory experiments were performed to determine whether phosphotungstic acid (PTA) employed in precipitating cesium from Purex FTW could be re-used (after caustic dissolution of the precipitate). Such re-use of all or a part of the PTA would substantially reduce the rather high chemical costs of the process. At least under the conditions of the experiments, no precipitate was obtained with recycled PTA, suggesting (as claimed by the British) that PTA is irreversibly decomposed in alkaline solution.

Solvent Extraction of Cesium

Laboratory Studies - Laboratory studies have continued on both the CSREX process and on the simpler BAMBP-only process (the latter for cesium recovery from tank supernate). CSREX studies have concerned both the equilibria and kinetics of cesium and sodium stripping. It was found that 1 M formic acid is a satisfactory scrub for the selective removal of sodium. Cesium stripping was found to be quite temperature dependent and slightly slow, with hold-up times of up to about 10 minutes required to achieve equilibrium. The temperature effect may prove useful in achieving optimum separation of cesium and strontium from cerium and in improving decontamination of cesium-strontium from calcium. In addition, the CSREX phenomenon (synergistic effect of D2EHPA on BAMBP to yield cesium extraction at low pH) was shown to occur with several other mono- and di-alkyl phosphates, but not with tri-substituted phosphates.

Work on the BAMBP-Soltrol system was aimed at generating data needed to support Cold Semiworks studies and included determination of the effect of excess caustic, BAMBP concentration, and temperature on cesium equilibria. Boric acid was shown to be an excellent scrub for sodium removal.

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Additional tracer-level miniature mixer-settler runs were made to test CSREX Process Flowsheet 2 conditions. In runs simulating the extraction column, feed solutions were simulated 1965 composition FTW made 0.19 M in citrate and adjusted to pH 4.5. Extractant was 0.3 M D2EHPA - 0.5 M BAMBP - Soltrol. At steady state conditions, Sr, Cs and Ce losses were 0.46, 0.09 and 7.4 percent, respectively. Organic product solutions were used in mixer-settler partition column runs. Cesium and strontium losses to the organic under partition column conditions were 2 and 0.13 percent, respectively.

Engineering Studies - The first pilot plant tests of the CSREX process was made during the month with favorable results. This process employs a mixture of 0.5 M BAMBP and 0.3 M D2EHPA in Soltrol-170 to co-extract cesium, strontium and rare earths. The cesium and strontium are stripped from the solvent with dilute nitric acid (0.05 M) followed by rare earth stripping with 0.5 - 2 M nitric acid. The process is quite similar to the D2EHPA process and would use the same equipment.

These first CSREX runs were performed in the same nine-foot-tall column used for the cesium-BAMBP process and demonstrated over 99 percent extraction of strontium and cesium from the citrate-complexed feed used in the D2EHPA process. Only 50 to 65 percent of the cerium was extracted at 25 C. However, raising the temperature to ca. 40 C increased the cerium recovery to as high as 97 percent (top interface) but increased the cesium loss to 21 percent. This result was expected from the known effect of increased temperature on cesium distribution ratios.

Partitioning (IB) column results showed that strontium and cesium losses of one percent or less were readily obtainable using 0.05 M HNO_3 at an aqueous-to-organic flow ratio of 1.0. This proved to be about the minimum amount of acid that would achieve these results, evidently because of an excessive amount of sodium in the organic feed. Adequate sodium scrubbing in the 1A column should permit up to two-fold reduction in both the acid concentration and the aqueous flow rate. The cerium decontamination factor appeared to be greater than 100 (no cerium detected in the 1BP stream).

Tentative 1C column results indicate that cerium losses of three percent or less are obtainable with a 0.5 M HNO_3 stripping solution. A higher acid concentration would probably be required to strip other rare earths quantitatively.

The column capacities, using the same 23 percent free area nozzle plate cartridge throughout, were comparable to those obtained in the D2EHPA process (volume velocities up to 600 gal/(hr)(ft^2), sum of flows, were readily achieved).

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Denitration of Purex 1WW

A laboratory test was completed during the month to evaluate a proposed scheme for sugar treatment of 1WW in the Purex Plant, e.g., simultaneous controlled addition of both sugar and 1WW to a heated reaction tank over a period of about 50 hours. Major uncertainties were whether residual carbon content would be excessive and whether efficiency of nitrate destruction (moles of nitrate destroyed per mole of sugar consumed) would be adequate. The residual carbon found was only 2 to 3 percent and the efficiency averaged about 20. Even better plant performance would be expected since (1) accurate control of the small flow rates involved in a laboratory-scale demonstration was difficult, and (2) the radiation associated with actual 1WW would destroy residual carbon and some nitric acid, thus increasing the efficiency.

In engineering studies, residual carbon analyses were completed for the flowsheet demonstration run reported last month. A residual carbon content equivalent to 1.9, 1.1 and 0.4 percent of the total carbon fed as sugar was found in the sugar-treated waste after a digestion period of 12, 15 and 18 hours, respectively.

Solvent Extraction of Np and Pu from Purex Waste

Tri-lauryl amine (TLA) extraction of neptunium and plutonium from Purex 1WW was studied in batch contacts with simulated 1WW of estimated current composition. In the present flowsheet, the 1WW is made 0.2 $\underline{M}$ in hydrazine and allowed to stand 30 minutes at room temperature; Np(V) is reduced to Np(IV) but Pu(IV) is not reduced to Pu(III). When this 1WW is contacted with an equal volume of 0.3 $\underline{M}$ TLA-Soltrol, 93-96 percent of the neptunium and 98-99 percent of the plutonium are extracted. Contact of the organic phase with one-half or an equal volume of 0.1 $\underline{M}$ oxalic acid strips 85 to 95 percent of the neptunium and plutonium. Tracer studies indicate good decontamination (decontamination factors $>$ 100) for all probable contaminants except ruthenium. Ruthenium decontamination factors were about five in the extraction and seven in the strip contact.

Cesium Recovery from Alkaline Wastes by Ion Exchange

Duolite C-3 and Bio-Rex 40 phenolic resins were investigated as possible exchangers for cesium removal from high-level alkaline supernate wastes. It was found that the Bio-Rex 40, a derivative of C-3, is not appreciably different in cesium loading characteristics from C-3 and therefore was not studied further.

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A solution of 0.1 N $(NH_4)_2CO_3$ + 0.1 N NH_4OH was used for sodium removal from Duolite C-3 resin following the loading step. Eight column volumes of this solution in addition to a four-column volume water wash were sufficient to remove 99 percent of the sodium in the bed. Cesium losses during this sodium removal step were very low and about the same as for other exchangers.

Elution of cesium from C-3 was accomplished with 2 M $(NH_4)_2CO_3$. It was found that flow rate during elution is much more critical for C-3 than for other exchangers studied. Particle size is likewise important. For 95 percent cesium elution, 6.8 column volumes of 2 M $(NH_4)_2CO_3$ are required for 20 to 50 mesh exchanger and two column volumes per hour flow rate. This is reduced to 5.6 column volumes for 50 to 100 mesh at two column volumes per hour, and to 4.2 column volumes for 50 to 100 mesh at 0.5 column volume per hour. Changes in eluant concentration make little difference at the large mesh size and high flow rate.

A 1-inch diameter, 18-inch long stainless steel column was equipped with solenoid valves and timer and is operating automatically on an alkaline supernate flowsheet. The first exchanger used was Linde AW-500 and 13 cycles of loading, washing and elution were completed. The flowsheet used resulted in a Na/Cs ratio reduction from 5000 in the waste to 10 in the eluate. Cesium loss to effluent and washes was 2 percent, but this could be reduced with a shorter loading period. After 13 cycles, the AW-500 was removed from the column, dried, weighed and counted. The weight loss was less than 5 percent even though some iron, presumably from the binder, was observed in each of the evaporated eluates. Volume loss of the exchanger was $\pm \sim 10$ percent. Cesium capacity loss as measured on the thirteenth cycle was 10 to 15 percent. Residual uneluted cesium on the exchanger was 2 percent of a normal cesium loading. Physical appearance of the particles was not appreciably different after the 13 cycles than before.

Dissolution of Rare-Earth Oxalate Precipitates

The Purex-B-Plant processing scheme for strontium and rare earths involves a rare earth oxalate precipitation step followed by slurry transfer of precipitate from the centrifuge. A dissolution procedure would be preferable and would minimize cross-contamination of equipment and (when cask load-out is involved) exposure of personnel. The dissolution of rare earth oxalates with nitric acid and manganese-catalyzed nitric acid is accordingly being investigated. Very nearly complete dissolution was obtained with 4 M HNO_3, 0.0005 - 0.005 M Mn in one hour at 80 C; however, it is not clear whether the manganese truly played a catalytic role.

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Promethium Purification

Purification was completed during the month of a quantity of promethium-147 which was recovered approximately a year ago and set aside at that time to allow decay of Pm-148. The purification was carried out by EDTA ion-exchange chromatography in ion-exchange columns installed for that purpose in C-Cell of the High Level Radiochemistry Facility. The feed (which contained about equal amounts of neodymium, promethium, samarium and trace europium) was diluted to 50 liters, adjusted to pH 3.8, and loaded onto a 2-inch x 6-foot and a 1-inch x 6-foot Dowex 50W, X-8 column. The resin in these two columns was in the ammonium cycle. The two columns were then eluted in series through two similar 1-inch x 6-foot columns (but in the yttrium cycle) with pH 8.75, 0.015 M EDTA. The columns were operated at about 60 C. Final isolation of the product was accomplished on a 1/2-inch x 6-foot column from which the promethium was eluted in 13 fractions of 900 ml each. Due to radiolysis (which slowly destroyed EDTA), the fractions began to precipitate shortly after collection. The promethium is being recovered from each fraction by oxalate precipitation followed by conversion to the oxide by ignition at 750 C.

Although analytical data on yield and purity is not yet available, preliminary indications are that the quantity obtained equals or exceeds the 10,000 curies desired and that purity of the heart cut is very high, with total impurities less (perhaps much less) than the estimated 2 percent which represents the detection limit by spectrophotometry.

In supporting laboratory work, completed too late for use in the hot-cell run, it was shown that aluminum is fully as satisfactory as yttrium as a restraining ion for the promethium purification process. For plant-scale application aluminum would be preferred because of its lower cost.

The availability of a macro quantity of highly pure promethium provides the opportunity to measure a number of its pertinent chemical and physical qualities - which either have not been previously determined or not determined with high precision on purified material. These include: emission spectrum, solution absorption spectra, X-ray spectra, and magnetic susceptibility of Pm_2O_3 (for which a value of $16.0 \pm 1 \times 10^{-6}$ c.g.s. units was obtained, almost exactly what a "simple" interpolation would have predicted.)

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Fission Product and Waste Packaging

Dehydration studies on Linde AW-400 Molecular Sieves (14 to 30 mesh) were discontinued because of the formation of a very hard cake of zeolite material in the off-gas line. The bed material had been thoroughly washed to remove fines before the run was begun, therefore the cake is thought to have formed from fines produced by thermal fracturing during the course of the run. X-ray diffraction showed no difference between the cake and "as-received" material. It is apparent that considerable improvement in the zeolite binder will be required before this material can be employed in present concepts. Further studies are planned with type 4A Molecular Sieve (14 to 30 mesh) which is reputed to be more stable than AW-400 at high temperatures.

Strontium and cesium loading on Linde 4A and AW-400, respectively, was determined in the laboratory using tracer solution from pilot plant scale fission product recovery experiments. The solutions were used to study possible difficulties from organic or other substances not present in simulated laboratory solutions.

Breakthrough data indicate more than 3 meq of strontium per gram of exchanger were loaded from a 1BP D2EHPA solvent extraction process solution. Prior to loading the 1BP was evaporated to half its original volume to concentrate the strontium and remove volatile organic material. The solution became slightly turbid when neutralized to pH 6. No deposition of organic or other substance on the bed of 4A was observed.

Forty-six liters of ammonium carbonate cesium eluate were obtained from a pilot scale experiment with Duolite C-3 resin and simulated Purex 103-A tank waste. The solution was evaporated to about 3 liters, centrifuged and pumped through a column of Linde AW-400. Column data indicate a cesium loading of 1.4 meq of cesium per gram of exchanger without detectable cesium breakthrough. Lack of feed stock prevented further loading, but based on the sodium to cesium mole ratio in the feed, a loading of 1.6 meq could be expected.

Volatilization and Elution of Cesium and Strontium from Loaded Exchangers

The following table summarizes the results of the volatilization and elution of cesium and strontium from exchangers which were heated to 1000 C after loading. Elution was normally performed with 1 $\underline{M}$ HNO_3; in a few instances where 2 $\underline{M}$ NH_4NO_3 were used as the eluant, the data are shown in parentheses.

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			Elution with 1 M HNO_3 (2 M NH_4NO_3)		
Exchanger Used	Isotope Loaded	Percent Vaporized	% Eluted at 20 Col. Vol.	Total % Eluted	Total Col. Vol.
Clinopt.	Cs-137	0.045	34 (33)	-	-
13X	Cs-137	0.064	86	98	80
13X	Sr-85	0.013	62	86	75
13X	Sr-85	0.021	62	92	95
4A	Sr-85	0.008	56	96	90
4A	Sr-85	0.014	58 (18)	94 (34)	85 (85)
AW-400	Cs-137	0.052	63 (51)	-	-
AW-500	Cs-137	0.028	20 (28)	-	-
Decalso	Cs-137	0.019	14	18	75
Decalso	Cs-137	0.031	26	31	75
Decalso	Sr-85	0.036	6	10	100
Zeolon	Cs-137	0.084	- (66)	-	-

Work now in progress shows that elution is more complete when the loaded zeolite has been heated only to 700 C.

EQUIPMENT AND MATERIALS

Low Flow Metering Pumps

Two metering pumps have been devised for use in plant process streams, with initial application in 234-5 Building. The pumps employ a "dipper" principle for removing precise quantities of fluid from a reservoir. Control of speed, tube size, and reservoir depth provides a flow rangeability of more than 60 to 1. The pumps are designed for precision and reproducibility of delivery in the flow range around two liters per hour with an accuracy of one percent or better.

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Non-Metallic Materials

Three formulations of ethylene-propylene rubber (EPR) were exposed to several common separations plant solvents at room temperatures and to boiling 10 percent nitric acid. The rubbers showed better resistance to solvents, especially hexone, than other available elastomers having comparable resistance to radiation-induced changes. They show promise for application as gaskets and O-ring seals in separations plants, especially Redox. Recipes are available on request.

Corrosion of Titanium by HNO_3 - H_2SO_4 Solutions

Samples of A-55 titanium have been exposed to boiling 1, 2 and 4 $\underline{M}$ H_2SO_4 solutions containing from 0 to 6 $\underline{M}$ HNO_3 to determine the concentration of nitric acid necessary for satisfactory inhibition of attack by sulfuric acid. A nitric acid concentration of 0.25 $\underline{M}$ appears adequate at any of the sulfuric acid concentrations studied; 0.03 to 0.06 $\underline{M}$ appears adequate at 4 $\underline{M}$ H_2SO_4.

Corrosion of 304-L by Chlorine Scrubber Solution

Samples of 304-L stainless steel were exposed at room temperature to the liquid and vapor phase of a synthetic chlorine gas scrubber solution (1.1 $\underline{M}$ NaOH - 1.75 $\underline{M}$ NaCl - 1.5 $\underline{M}$ NaOCl). Crevice samples were also exposed in the liquid phase. General corrosion rates were about 0.1 mil/mo. However, several samples showed severe pitting attack in the range of 30 to 150 mils/mo. Methods of destroying hypochlorite in the scrubber solution by reducing it to chloride were studied. Ammonia, urea and hydrogen peroxide were all effective reductants. Urea was recommended because of easier application.

Stress Corrosion Cracking of Mild Steel

Following the October inspection of large (3-ft. x 3-ft. x 3/8-in.) welded test specimens, weld metal was removed from a section of the seam weld on four of the weldments and the area rewelded (repaired) using the same weld rod type as used in making the original weld. Three of the large weldments, including two with repaired areas, were exposed to 50 percent $NaNO_3$ - 1 $\underline{M}$ $NaNO_2$ solution. After 82 days' exposure, the two repaired weldments were severely cracked. Cracking appeared to originate at the repair areas. No cracks were found in the weldment which was not repaired.

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Five of the large weldments, including two repaired specimens, were exposed to synthetic Purex alkaline waste (6 M NO_3^-). The two with repaired areas were severely cracked after 82 days' exposure. Two of the three without repaired areas were not cracked while the third developed a crack. The base metal in all of the repaired weldments and all but one of the non-repaired weldments corresponds to ASTM 283, Grade C (0.12 to 0.16 percent C). Base metal in the one non-repaired weldment corresponds to ASTM A-201-57T (0.20 percent C). This specimen has not cracked after 140 days' exposure to synthetic alkaline Purex waste.

Welding rods of types 6010, 6012, 6016 and Inconel 182 were used in fabricating the weldments. No correlation of cracking with weld rod type has been shown.

PROCESS CONTROL

Control System Component Testing

A recently available signal converter was evaluated to determine its suitability for process control applications. The converter can accept 0 to 100 millivolt signals as well as the conventional 1 to 5, 4 to 20, and 10 to 50 milliamp signals and converts to outputs of 0 to 10 millivolts, 0 to 100 millivolts, 0 to 1 volt, and 0 to 10 volts as well as the conventional current levels. Stability over a two-week period with normal line voltage variations and temperature changes appears to be within a one percent of span. The unit does not appear to be affected by supply voltage variation between 100 and 130 volts AC. However, resolution of the span and zero adjustment potentiometers is not good and results in some difficulty during calibration; also ripple in the output may be excessive for some applications. The input impedance when used with a 0 to 100 millivolts input is on the order of 20 K ohms. The device can be recommended for use in the forward loop of a control system such as adapting a controller with a 1 to 5 milliamp output to a valve requiring 10 to 50 milliamps. Accuracy and ripple should be further evaluated before use in feedback loops.

Plutonium Monitor in Purex Canyon

Design of the pre-amplifier location and instrument cabling for a recycle plutonium tank (E-6) neutron monitor has been completed. The charge-sensitive pre-amp will be jumper-mounted external to the E-6 tank containing the BF_3 detector. Fabrication of the electrical connector jumper is underway.

Calorimeter Studies

Preliminary laboratory investigations were made to develop rapid methods of measuring heat generation rates in fission product streams. Using thermistors to measure temperature rise in an insulated system, a heat generation rate of 0.1 watt per liter can be determined. Additional studies are underway to determine the precision of the measurement and to obtain direct read-out of the heat generation per unit volume.

Filter Blowback Cycle Programmer

A solid state time delay relay (TDR) was developed for use in accurately programming the time that reverse air flow is applied to the 234-5 Building glove box air filters. Precise adjustment of the blowback time is necessary to prevent loss of, or gross changes to, the normal glove box vacuum. The time delay relay circuit developed for this application has a time cycle manually variable from 0.2 to 5.5 seconds, repeatable to about one percent. The time delay circuit has potential application to numerous other plant and laboratory control problems. The time cycle may easily be made shorter or longer with an upper limit of about two minutes and a lower limit of a few milli-seconds.

C-Column Studies

The second block of 16 runs in the C-column was initiated, following a shutdown for make-up of new organic feed and for preventive maintenance of the column control instrumentation. A paper tape punch was incorporated into the absorptiometer to expedite the data reduction step. Overnight processing of all column and absorptiometer data is now proceeding smoothly after some initial difficulties in establishing data handling procedures.

An investigation was made to determine the cause of the unexpected curvature in the Am-241 gamma absorptiometer calibration curves. It was postulated that the presence of high energy peaks (higher than the 60 Kev peak) due to an unknown containment might cause the curvature. Analysis of the spectrum with a scintillating crystal demonstrated the absence of such higher energy peaks. On this basis it was concluded that the curvature is due to scattering around the sample cells, plus background. A mathematical explanation of the phenomenon was developed which fits the calibration data.

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REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Fuel Element Decladding - A remotely operable vibratory decladder for removing UO_2 from fuel rods has been tested (not remotely) successfully on 1/2-inch diameter Zircaloy-clad PRTR fuel rod segments. The decladder consists of a nominal 5000 blow-per-minute air hammer, a vertical stationary anvil which supports the fuel rod full length, and a counter-weighted manually-movable air hammer carriage fastened to the anvil. The faces of the hammer and anvil are shaped to give six points of contact on the fuel rod circumference. The 8-foot fuel rods are bisected with a tubing cutter and placed against the anvil with the open end down into a tapered lower support containing a discharge into the UO_2 receiver. The upper support is then clamped over the fuel rod end cap by means of an air cylinder. As the vibrating hammer moves along the tube the fuel fractures, loosens and drops into the receiver. The UO_2 was removed from vibratory compacted elements by merely vibrating the bottom of the tube.

A cold swaged element required two passes up and down the tube to remove the fuel, while a hot swaged element required approximately five passes. In all cases the entire operation took less than 10 minutes for each rod segment. The uranium losses to the cladding, determined by analysis of hot nitric acid cladding leach solutions were 0.012, 0.059 and 0.013 percent for the vibratory compacted, cold swaged, and hot swaged rods, respectively.

Precipitation Studies with Cerium Stand-In for Plutonium - Equipment for the recovery of plutonium as PuO_2 from molten LiCl-KCl salt solution by treatment with O_2-Cl_2 gas was tested on an engineering scale using cerium as a stand-in for plutonium. Contrary to previous experience with thorium stand-in, CeO_2 was found to form as a dense, filterable, crystalline precipitate from a 2.5 LiCl-KCl mixture. A salt solution containing 10 w/o U and 0.63 w/o Ce was gas-lifted with 60 percent Cl_2 - 40 percent O_2 gas mixture through three cascading quartz-frit filters. Of the cerium precipitated, greater than 80 percent was recovered in an early test. Additional studies will be required to increase precipitation efficiency on an engineering scale.

Pilot Plant Electrolyses - The results of five dissolution-electrolyses in a 20-liter molten salt bath indicate that the operating characteristics and the quality of the UO_2 produced in the KCl-2.5 LiCl system are not significantly different from those in the equimolal KCl-LiCl system. However, the higher melting point of the former system with added uranium did require an increase in electrolysis temperature from 535 to 570 C.

Graphite Electrodes - One of the troublesome engineering problems of the Salt Cycle Process has been overcome by application of new materials. Formerly, mechanical difficulties in removing UO_2 from electrodes and intolerably high (100 to 2000 ppm) overall carbon contents resulted from the use of nuclear grade graphite cathodes (sp gr 1.68). The carbon content in the part of the deposit not immediately adjacent to the electrode is about 20 ppm for UO_2 produced at 530-570 C and about 200 ppm for UO_2 produced at 700 C, where appreciable corrosion of the graphite anodes occurs. The high carbon contents are apparently due to mechanical interlocking of the graphite with the UO_2 and fracturing in the weaker graphite.

Recently, by using 1.5-inch diameter by 3-foot graphite cathodes coated with 1-mil-thick pyrolytic graphite, the UO_2 was removed by splitting the deposit lengthwise with a 53 degree wedge blade mounted on a hydraulic press. For both dry and water-soaked deposits the UO_2 did not adhere to the pyrolytic graphite and the electrode was undamaged. The UO_2 product from this operation contains an overall 25 ppm carbon with only 50 ppm in an 1/8-inch thick slice immediately adjacent to the graphite. (In similar tests with nuclear grade graphite the deposit adhered so strongly that the electrodes split longitudinally with no separation of UO_2 from graphite.) The product was then prepared as feed for a crusher by dicing to 1/2-inch cubes by means of the same wedge used above. The diced UO_2 contained only 0.8 percent -100 mesh fines and the entire operation could probably be conducted remotely without dust containment equipment.

Product Purity - The previously reported (HW-75925 C) value of 500 ppm chloride obtained by 24-hour water washing of -6 +10 mesh UO_2 produced in LiCl-KCl salt baths has been revised to 35 ppm chloride by improved analytical techniques. Analysis of arc-fused UO_2 from the Norton Company by the same technique gave 20 ppm chloride.

Flowsheet Development - Continued studies of the electro-codeposition of PuO_2-UO_2 have resulted in the attainment of plutonium enrichment factors as high as 13.5. This enrichment was achieved by electrolyzing equimolar LiCl-KCl containing 20 w/o U and 0.24 w/o Pu at 625 C with a starting current density of 0.2 amp/cm^2 while sparging with 80 percent O_2-20 percent Cl_2. Five percent of the uranium and

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68 percent of the plutonium were deposited, giving a product containing 14 w/o PuO_2. The promethium decontamination factor on a plutonium basis was six, which is about the same as was obtained in codepositions giving plutonium enrichment factors near unity.

Recovery of PuO_2 from PuO_2-UO_2 codeposits was accomplished by selective dissolution of the UO_2 in melts sparged with an O_2-Cl_2 mixture. On digestion of high enrichment PuO_2-UO_2 deposits (about 10 w/o PuO_2) in fresh melts at 575 C, uranium (and promethium) decontamination factors of about 50 were obtained in LiCl-KCl sparged with 80 percent O_2 - 20 percent Cl_2 and in 2.5 LiCl-KCl sparged with 30 percent O_2 - 70 percent Cl_2. This decontamination plus that achieved across the codeposition step itself results in overall decontamination factors of about 300 for promethium and 500 for uranium. The PuO_2 recovered in these experiments was in the form of an amorphous solid, in contrast to the dense crystals obtained by precipitating PuO_2 from 2.5 LiCl-KCl melts by oxygen-chlorine sparging.

Electrochemistry of Uranium in Molten Chloride Salt Solutions - EMF measurements for the cell reaction

$$UO_2 + Cl_2 = UO_2Cl_2$$

have been made as a function of KCl concentration in the LiCl-KCl melt system at various temperatures with a uranyl(VI) concentration of 0.1 molal. At a given temperature the EMF was found to be a linear function of the KCl concentration in the range studied (20 to 60 mole percent KCl). This is in contrast to the behavior observed in the LiCl-NaCl melt system at the same uranyl(VI) concentration, where an "S"-shaped curve was obtained by plotting EMF at a given temperature vs. NaCl concentration (HW-75127 C). Further work is required before extensive conclusions can be drawn regarding this interesting difference in melt behavior.

RADIOACTIVE RESIDUE FIXATION

Zeolite Properties

Rational thermodynamic equilibrium constants were determined for the binary systems potassium-cesium and potassium-sodium as a means of ascertaining the validity of the previously-reported sodium-cesium equilibrium constants for the zeolites clinoptilolite, Linde AW-500, AW-400, AW-300, 13X and 4AXW, and Norton Zeolon. Thermodynamic relationships require that the standard free energy for the reaction, $K_{zeolite} + Cs^+ \rightleftarrows Cs_{zeolite} + K^+$, minus the standard free energy for the reaction $K_{zeolite} + Na^+ \rightleftharpoons Na_{zeolite} + K^+$, equals the standard free energy for the reaction, $Na_{zeolite} + Cs^+ \rightleftharpoons Cs_{zeolite} + Na^+$. As an example, with the standard free energies of the above reactions expressed to the nearest 100 cal/mole, the computation for clinoptilolite is, (-800) - (+1500) = (-2300). The actual value obtained from the experimental data for the standard free energy for the sodium-cesium reaction was -2340 cal/mole. Good agreement between experimental and computed standard free energy values also was obtained for the other zeolites with the exception of Linde AW-300. Irregular isotherms were obtained for the sodium-cesium and potassium-cesium systems, and a regular isotherm was obtained for the potassium-sodium system. It is probable that a "steric hindrance" is the cause of the anomalous behavior of the AW-300 in cesium-containing systems. Numerous "stacking faults" within the AW-300 crystal reduce the smallest opening through which cationic diffusion must occur to about 4 Å. The resulting ionic diameter of potassium plus sodium is about 4.6 Å, and of cesium plus sodium, about 5.3 Å. The latter value shows the space difficulty encountered when a sodium and cesium cation try to pass each other in the 4 Å openings. Zeolon, which has the same crystal structure as AW-300 except for the absence of stacking faults, shows regular isotherms for systems containing cesium. The smallest dimension through which cationic diffusion must take place in Zeolon is 10 Å. It is concluded, therefore, that equilibrium was not attained during the AW-300 experiments with sodium-cesium and potassium-cesium systems, and that thermodynamic data derived from these pseudo-equilibrium experiments are invalid.

Condensate Treatment

Small scale experiments on condensate treatment in the Micro Pilot Plant were suspended. Effort is being directed toward an engineering scale demonstration of the decontamination of Purex Tank Farm condensate. The steam stripper and ion-exchange columns that are being assembled will have a nominal capacity of about one gallon per minute, a reasonable fraction of a condensate stream from a power reactor fuels reprocessing plant that will require decontamination.

Electrostatic Bubble Scrubber

An equation was derived for calculating the collection time of a charged particle trapped in a grounded bubble to aid in understanding the importance of the variables involved. For particles less than about one μ radius the collection time is proportional to $\frac{a^3}{rK_mT^2}$; for larger particles the collection time is proportional to $\frac{a^3}{r^3}$

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for comparable charging conditions, where

a = bubble radius
r = particle radius
K_m = Cunningham factor
T = temperature

The relative importance of bubble size and particle radius is thus evident.

Eighteen-Inch Radiant-Heat Spray Calciner

During the month, 11 runs were made with feed solutions simulating Purex high-level wastes. Two channel recycle tubes of different geometries were installed and tested. The tubes were 11-1/4 inches in diameter by 4-feet long and 13-3/4 inches in diameter by 7-feet long, respectively. Major findings included:

1. At an 800 C furnace temperature, the capacity of the clean reactor with the larger channel recycle tube was 25 percent higher than that measured with no tube.

2. The larger channel recycle tube was effective in greatly reducing the buildup of calcine on the reactor wall. After 14 hours of running the reactor capacity was reduced by only about 10 percent as compared to some 30 percent after four hours of running without a channel recycle tube.

3. The smaller channel recycle tube was not effective in preventing calcine buildup on the reactor wall. Deposits were similar to those found in the calciner with no tube.

4. The density of the calcine produced appears affected by the calciner feed rate. Runs at high rates produce lower density calcines. A comparison of the data obtained on the 18-inch calciner and on the 8-inch calciner appears to indicate that a maximum density product is produced at rates of about one-third to two-thirds the maximum capacity of the unit.

Induction Heater Control System

Methods of controlling induction heaters are under development for use in fused salt or waste calcination systems. Effective on-off [redacted]

The gating circuitry, consisting of a single resistor and a diode for each SCR, obtains its power from the generator line terminals, eliminating the need for a DC power supply, a phase shift circuit, and a pulse generating circuit.

By varying the resistance in series with each SCR gate a small amount of proportional control was obtained over the positive-going portion of each half cycle of the 10 KC sine wave but effective control was obtained only at reduced voltages of 90 V or less. For voltages above 90 V and at mid-position gate circuit potentiometer settings, a resonant interchange of energy seems to occur between the two work stations causing the auxiliary work station current to suddenly increase to a high value and then decrease by about 30 percent at the full "on" setting. The cause of this apparent resonant energy interchange is unknown but attempts will be made to determine its cause in future tests.

Continuous Melt Furnace

A conceptual design of a continuous melt furnace was developed. Effort was also made to designate materials and methods of construction. As envisioned, the unit would continuously receive dried powder from the calciner and semi-continuously discharge melt to the final storage containers.

Rotary-Kiln Calciner Study

At the request of the Atomic Energy Commission, an evaluation of the Brookhaven rotary-kiln calciner is in progress. A conceptual design of the reactor has been outlined along with a chemical flowsheet and engineering flow diagram. The principal points of comparison between the rotary-kiln and the radiant-heat spray continuous calciners are:

1. The ability of the calciner to handle the needs of the reprocessing complex as well as to cope with the perturbations in waste composition.

2. Anticipated operating problems, off-standard operation and recovery, ease of startup and shutdown.

3. Equipment reliability, on-line efficiency, ease of maintenance, corrosion problems, and general life expectancy of equipment.

4. Decontamination, disposal or re-work of output streams other than waste solids, i.e., off-gases and condensates.

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5. Solid product removal, handling, packaging, container decontamination and storage.

6. Development status of the calcination concept, engineering development (cold) still required, major areas of unknown technology, etc.

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BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

Computer programming of the RE-PERM study was begun during the month to obtain needed information for the ground-water analog. This study will convert soil permeabilities to equivalent electrical resistances, define the boundaries of the individual flow systems, and assign a sequential scanner number to each point in the system.

Depth to ground water was found to play a significant role on the flow of fluid from a trench into unsaturated media. Using the Hanford-developed "Steady-State Flow" program and data from a typical Hanford soil, a decrease in seepage rate of about 2.5 was calculated when the water table was lowered from 10 feet to 40 feet beneath a water-filled trench. Also, when the seepage loss was related to the ratios of permeabilities between the trench lining material and the base soil, it was apparent that a large reduction in the permeability of the lining material is necessary to result in a moderate decrease in seepage loss.

Several significant changes were noted when constructing the new ground-water contour map. New basalt data obtained from well drilling and magnetometer interpretations indicate that a portion of the eastward extension of the Yakima Ridge lies above the water table, and that the basalt rises above the water table from the south side of Dry Creek Valley to the Yakima Ridge extension south of 200 West Area. This interpretation means that the ground water from Dry Creek Valley flows northward toward the 200 West Area ground-water mound instead of southeastward toward the Yakima River. Ground-water contour data obtained from new wells in the vicinity of the 100 Areas indicate very permeable sediments between 100-B and 100-K Areas and between 100-D Areas. This new interpretation is consistent with the ground-water temperature data in explaining waste water movements in this region.

Detection of low concentrations of fission product tritium and thermal heat in the bottom of well 699-20-E12, together with piezometric head measurements, indicates that ground-water contaminants are flowing in a confined aquifer along the top of the basalt series to a point 12 miles east-southeast of the 200 East Area. This confirms a situation recognized in 1959 which resulted in the drilling of that and other wells. The possibility of contaminants flowing beneath the Columbia River is being investigated. The

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contaminated ground water at Well 699-20-E12 lies about 200 feet below Columbia River level. Stratigraphic data indicate that ground water will probably flow south or south-southeast to ultimately rise and flow into the Columbia River north of Wallula Gap.

Scintillation logging of several uncontaminated wells revealed that the scintillation probe has use in distinguishing between geologic units. Fine sediments (silt and clay size material) have markedly higher radioactive material content than coarse sands and gravel; basalt exhibits the lowest radioactivity of all the materials tested. There are indications that the radioactivity is a function of the organic content of the material rather than of the mineral content. Further investigations are being made to determine the utility of the probe for defining more exactly the stratigraphic interfaces of project subsoils.

RADIOLOGICAL AND HEALTH PHYSICS

Environmental Studies

The new 100 cfm air sampling pump on the 329 Building roof was calibrated with a precision Roots Meter to permit flow measurement from a water manometer downstream from the pump. Studies of the filter efficiencies for removal of fallout radionuclides on various filter papers are in progress. The filter efficiency is measured by placing the filter to be tested upstream from an AM-3 filter which is essentially 100 percent efficient.

The accompanying table gives the filter efficiencies of IPC filter paper as a function of air velocity:

Air Velocity	Radionuclide Held (%)					
(linear ft/min)	BaLa-140	Ru-103	ZrNb-95	Pb-212	Ce-144	I-131
770	60	65	85	-	-	55
1500	73	70	90	-	77	-
2000	97	95	98	77	-	91
4400	97	97	99	83	96	81

At high velocities the IPC filter paper is about as efficient as the AM-3.

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Radiation Chemistry

Investigation of the radical-scavenging ability of histidylhistidine provided new evidence on several factors which determine the cause of radical reactions in irradiated, aqueous solutions. Although histidylhistidine contains twice the number of atoms arranged in an almost identical manner it does not have twice the radical scavenging ability. Its protective index is 0.41 compared to 0.33 for histidine. Histidylhistidine does have a larger fraction of positively charged sites than histidine and positively charged sites have been shown to inhibit radical reactions. The protective index decreased with increased temperature similarly to the three amino acids investigated previously and the variation corresponded to a difference in activation energy of 1 kilocalorie per mole for all four compounds. This similarity lends weight to the supposition that oxidation of the amino acid due to radiation occurs by hydrogen abstraction through realignment of the hydrogen bonds in the water structure around the acid.

A simple weighted least squares program was written and verified to analyze spectrophotometer absorbance data for the calculation of protection indices in radiation protection studies. The program was written sufficiently broad so that other least squares curve fitting computations can be made.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition in Conduits

Many deposition measurements were completed for 3-inch diameter vertical tubing, 58 feet long, for air flow rates giving Reynold's Numbers from 1600 to 14,000. Measured deposition ranged up to 15 percent for the ZnS particles used, whereas deposition predicted from correlation of data on smaller diameter tubes was less than one percent. Even though the precision in these measurements was not as good as for experiments with smaller diameter tubes, the estimated error in the measurement is not great enough to account for the difference between observed and predicted deposition.

ISOTOPES DEVELOPMENT - 08 PROGRAM

Packaging Plant Study

Study flowsheets were prepared (based where possible on existing technology) for the large-scale purification and packaging of fission products in a proposed new Hanford plant. Feed material

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was assumed to be the output of B-Plant, using a modified flowsheet to permit recovery of promethium along with cerium in the CSREX process. Strontium is separated from cesium by carbonate precipitation, and purified by solvent extraction (or alternately by ion-exchange). Cesium is purified by solvent extraction with either BAMBP or dipicrylamine. The final strontium product is the titanate, and the final cesium product is a glass (either borate or cadmium) made from cesium oxalate. Both cerium and promethium are recovered and packaged as the respective oxides.

Separation of cerium from the other rare earths appears to be the most difficult and least satisfactory part of the process. Alternate flowsheets using solvent extraction (Oak Ridge batch D2EHPA process) and pyrochemical processing were proposed for evaluation. Exploratory laboratory work on the pyrochemical process showed promise; however, more work will be required before a choice can be made. In the pyrochemical process, the rare earth nitrate solution is added to a magnesium nitrate solution and evaporated. Transition to a molten salt occurs without passing through a solid state, and cerium is oxidized to CeO_2 and precipitates when a sufficiently high temperature is reached, leaving the trivalent rare earths in the melt. Potential advantages are equipment compactness and relative process insensitivity to the high heat generation and high radiation flux associated with fission product cerium.

Direct formation of encapsulated $SrTiO_3$ from strontium carbonate by the Dynapak process was further scouted during the month, using heavy walled cans to avoid internal fracturing due to springback. In the most successful test, a mixture of $SrCO_3$ (equivalent to 12,000 curies or about 100 thermal watts) and TiO_2 (10 percent mole excess) was loaded into a Dynapak can (1/8-inch wall), a lid welded in place, an off-gas line connected, and the assembly heated under vacuum for three hours at 1210 C and 45 minutes at 1300 C, followed by compaction with a tool steel punch at a pressure of approximately 200,000 psi. Complete conversion to $SrTiO_3$ was obtained and the product density was 4.46 gms/cc, the best to date. The completed package, including cladding, was only 2-1/2 inches in diameter and 1-1/2 inches thick. Higher densities could doubtless be obtained with carbide dies, which would permit pressures of over 400,000 psi.

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Manager
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BIOLOGY LABORATORY

A. ORGANIZATION AND PERSONNEL

J. J. Davis, Manager, Radioecology, resigned on January 4, 1963, to accept a position elsewhere.

The Plant Nutrition and Microbiology Operation was dissolved, with its personnel transferred to Radioecology, with Dr. Frank P. Hungate as the new Manager.

GENERAL

Remodeling of 144-F Building is completed. Four offices and the lunch room were converted into one large physiology laboratory and one animal colony record office. Crowding in the dog colony continues to be a problem. Most all of the runs now house 3 adult dogs - a hazardous practice due to tendency toward fighting.

Rats are being sacrificed for pathologic examination, immediately upon arrival, to evaluate claims for their pathogen-free condition. Bedding materials are being examined for use in the small animal cages. Air conditioning, humidity, and air flow in the new small animal quarters are being examined to determine their adequacy. It is already apparent that limits must be set on the number of animals housed in each room. It is also apparent that the veterinary requirements of the small animal colony will be about one-half day per week.

Water from melting snows combined with ice pressures to wash out both dams at Rattlesnake Springs. Some recording equipment was lost but most survived the flood.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Effect of Reactor Effluent on Aquatic Organisms

Biological monitoring of reactor effluent with young chinook salmon was initiated on January 16, 1963, at 100-KE. A total of 5000 fish will be utilized to study, in addition to the usual monitoring, the effect of a promising corrosion inhibitor, Quachrom Glucosate, on growth and survival. The experiment will be terminated in late spring.

Toxicity tests of reactor tube cleansers, SULFAM-3 and BISULF-16, continue. Preliminary estimates of LD-50 and LD-10 for young cichlids and chinook salmon are given below:

	SULFAM-3		BISULF-16	
Test Fish	LD-50	LD-10	LD-50	LD-10
Cichlid	205 ppm	180 ppm	280 ppm	230 ppm
Salmon	150 ppm	146 ppm	180 ppm	175 ppm

As expected, young salmon are more sensitive than cichlids, and BISULF-16 is slightly less toxic than SULFAM-3.

Columnaris

Fish from 100 K aquatic biology troughs continue to yield viable columnaris from up to 36% of the gills sampled.

Using minimal amounts of tryptone with addition of individual amino acids, it was possible to categorize these amino acids as stimulatory, indifferent, or inhibitory. A medium composed of the 11 stimulatory amino acids and glucose, which was also stimulatory, was still incapable of sustaining growth in the absence of traces of tryptone.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY AND TRANSFER OF RADIOACTIVE MATERIALS

Waterfowl

Wintering waterfowl populations within the Hanford Reservation increased steadily during the month to a maximum of 330,000 ducks and geese near the month's end. This was the greatest number of waterfowl observed this season and was 60 per cent above peak populations last year.

Zinc

Yearling trout fed 46 μc Zn^{65} and killed 31 days later showed a relatively high concentration in the gill filaments. Concentrations in tissues were 180, 32, 26, 24, 13, and 13 nc/g for gill filaments, bone, eyes, liver, blood, and plasma, respectively. Body burden was 12 μc, or 26 per cent of dose. With three-year-old trout, from 2 to 7 per cent of dose appears in the urine during the first week.

Strontium

Gross autoradiographs of bones from mature female miniature swine, fed Sr^{90} daily from young adulthood, showed increased labeling with longer periods of Sr^{90} feeding. Nine months after initiation of feeding, most of the bones were rather uniformly labeled with Sr^{90} except for the long bones (such as humerus, femur, etc.) in which the cortical bone of the mid-shaft was poorly labeled. This lack of labeling, indicative of very slow turnover of the bone mineral, was still apparent in the long bones of an animal sacrificed two years after initiation of Sr^{90} feeding. These autoradiographic results agree closely with those based on radioanalysis.

A number of Sr^{90} sources to be used as autoradiography standards were prepared by pouring Sr^{90}-spiked mixtures of plaster of Paris on glass plates. Glass plates were used to assure a smooth surface. The sources are of sufficient size to allow measurement of radiation dose rate with an extrapolation chamber. Measured values are approximately the same as those calculated from Sr^{90} concentrations. Use of these sources should provide more accurate autoradiographic dosimetry than hitherto possible.

Iodine

Control values for thyroid I^{131} uptake were determined in twenty sheep. These animals will be utilized in a study to determine the short-term comparative effects of external X-irradiation, and irradition by I^{131}, on the thyroid. A single animal was exposed to 1000 r (measured air dose) to the thyroid (KV-250, MA-30, TSD-50 cm, port size-7.5 cm^2, exp. time-10.5 sec., air dose-95 r/min, HVL ~.5 mm Cu) to determine the immediate effects of a relatively high dose on the esophagus and trachea which lie within the exposure field. No untoward effects were observed during the first three days post-exposure.

The three female miniature goats were radiographed and it was determined that all three are in advanced pregnancy. Therefore, daily low-level I^{131} feeding was initiated and thyroid uptake determinations were made. At parturition, milk, blood and fetal concentrations of I^{131} will be determined.

The I^{131} concentration in milk from cows being fed 5 μc I^{131} has gradually increased to a level of 6, 9, and 18% of the daily dose. The volume of milk secreted by the cow with the greatest I^{131} secretion is about 1.7 times that of either of the other cows, and shows a milk concentration of 0.027 μc/liter, compared with about 0.02 μc/liter in the other two cows. The concentration of I^{131} in the thyroid glands of the cows reached a peak during the week following the birth of the calves, then leveled off and began a gradual rise during January. Thyroid concentrations range from 1.5 to 3 times the daily dose of I^{131}.

Great variations were seen in concentrations of iodine in tissues and excretions of the three calves sacrificed immediately after birth. Tissues and excretions generally showing greater than blood concentrations of I^{131} were thyroid, stomach, and intestinal contents and parotid salivary gland. Lung tissue showed an unusually high I^{131} concentration in one animal.

Cesium

Daily feeding of Cs^{137} to ten male sheep was initiated in order to provide more reliable estimates of the long-term uptake and retention of orally administered Cs^{137} in a ruminant. Both young and old sheep are being used to determine the influence of age on Cs^{137} metabolism.

Neptunium

Male rats were found to be more resistant to acute Np^{237} toxicity than females. A dose of 12 mg of Np^{237}/kg of body weight killed females within about 48 hours. Males receiving the same dose are still alive after 5 weeks. Liver lipid content which increases markedly in females at this dose level is unaffected in males. Although approximately the same fraction of administered Np^{237} is deposited in the liver of both males and females, the sub-cellular distribution of the neptunium is markedly different, the major fraction being found in the nuclear fraction of the females and in the mitochondrial fraction of the males.

Plutonium

Preliminary results are available from toxicity studies on plutonium removal agents. At the level of 6 mM/kg, intraperitoneally, DTPA causes minor distress on the 1st day and transient kidney and liver damage. At the same dose level TTHA and DTPA methyl ester resulted in death of all animals within 5 hours. At the 1.5 mM/kg level DTPA showed no evidence of toxicity. TTHA was similarly well tolerated, but with mild histopathologic changes in kidney and spleen. At the 1.5 mM/kg level DTPA ester caused loss of weight, diarrhea, and pathologic changes in kidney, liver, spleen, and intestine.

Blood Studies

Chromium-51 tagged erythrocytes were used to measure the blood cell volume in three to four year old female miniature swine. The mean value for five animals was 2.1% of the body weight. Total blood volume and plasma volume were calculated from the blood cell volume and the measured hematocrit; they were about 4.3% and 2.2%, respectively, of the body weight. These relatively low values are consistent with our previous measurements made on swine of this age and probably are related to the large amounts of adipose tissue in these animals.

Inhalation Studies

Dogs exposed four months ago to $Ce^{144}O_2$ aerosols are in relatively good health, but those receiving the highest levels of exposure (deposition of 0.7 to 2.4 mc) are beginning to show certain changes. Total blood white cell counts are 50 per cent of the controls. The major decrease is in the lymphocytes, but neutrophil levels have also dropped. This is in contrast to plutonium-exposed dogs in which there is almost always a drop in lymphocytes without changes in other circulating elements of the blood. In the high level Ce^{144} dogs the levels of serum potassium also dropped below control levels, the blood CO_2 levels show a slight increase, and a slight increased respiratory rate was observed. The total radiation dose to the lungs of the $Ce^{144}O_2$-exposed dogs showing these effects ranges from 20,000 to 50,000 rads.

A masonite dog phantom was constructed to estimate the total-body dose from Ce^{144}-Pr^{144} retained in the lung. With Ce^{144}-Pr^{144} dispersed in a sponge in the "lung" cavity, the gamma dose rate through one inch of masonite adjacent to the lung is about 1.4 mr/hr/mc; through 3 inches of masonite, 0.55 mr/hr/mc; and through 10 inches of masonite, 0.04 mr/hr/mc. These data indicate that total-body gamma exposure from 1 to 2 mc Ce^{144}-Pr^{144} in the lung of a dog would be too low to contribute significantly to the toxicologic response from a $Ce^{144}O_2$ inhalation exposure.

A series of rat experiments were performed to test the effect of I^{127} vapor as a diluent of I^{131}, on thyroid uptake of I^{131}. The addition of I^{127}, at a level of about 0.1 μg/cc to air containing 3×10^{-5} μc I^{131}/cc resulted in a maximum thyroid level of about 1 per cent of the total I^{131} deposited. The ratio of I^{127} to I^{131} in the air on a weight basis was about 3×10^{8}.

Inhalation of carrier-free I^{131} resulted in thyroid levels of about 9% of the total I^{131} deposited.

Three years after inhalation of less than 10 μc $Pu^{239}O_2$, 5 of 17 dogs have respiratory rates of 80 - 120/min, compared with 20-30/min for the controls. Twelve of the dogs show significant lymphopenia. During the past year two dogs died as a result of the plutonium exposure, and three were sacrificed. The tissue distribution of Pu^{239} is shown below:

Distribution of Plutonium
(% of body burden)

	Dog number				
	182	184*	79+	104+	158+
Lung	74	76	34	68	33
Bronchial lymph nodes	21	5	57	25	53
Muscle	1	14	0.7	1	2
Liver	3	5	6	2	10
Bone	0.8	0.2	0.7	1	1
All other tissue	1	0.7	2	2	2
Time of death (days after exposure)	855	933	896	831	897
Final body burdens (μc)	2.7	2.5	0.9	0.8	0.14
Particle size of aerosol, μ (CMD)	0.5	0.1	0.1	0.1	0.1

* One of the bronchial lymph nodes was accidentally analyzed with muscle.

+ Sacrificed.

Histopathology was seen in lungs and bronchial lymph nodes, except in dog number 158 in which the lymph nodes showed little damage. During the last month prior to death the daily rate of excretion of plutonium in these dogs [redacted]

Re-evaluation of pathologic data from rats exposed for 60 days to Y^{90} in their drinking water indicates that exposures resulting in a 25 per cent incidence of corneal opacity and ulceration show no evidence of cataract formation. This observation is possibly explained by the lack of penetration of the Y^{90} beta to the lens, and should be of some clinical interest in view of the use of Y^{90} in therapy for human eye lesions.

Radiation Protective Agents

Experiments were performed to study the possibly synergistic effect of glutathione and other protective agents. Glutathione itself was unusually effective. Seventy per cent of the treated animals survived a 950 r total-body X-ray dose. Other agents such as cysteine and the manganous-DTPA chelate, alone or in combination with glutathione, were surprisingly ineffective. These results are not in accord with previously obtained data and further study is obviously required.

Considerable data have been accumulated on various possible methods of altering the secondary disease which accompanies bone marrow transplantation. Certain general conclusions are becoming apparent, although results differ markedly with the strain of host and donor animal employed. Under certain conditions, at least, the pre-treatment of adult donors with host lymphoid tissue results in a greater number of secondary disease mortalities. Increased cell dose also enhances secondary disease. Pre-treatment of adult host with donor lymphoid tissue results in rejection of donor marrow cells.

Indications were previously obtained that bone marrow from Sprague-Dawley rats, neonatally treated with mouse cells, was more effective in protecting irradiated mice than was bone marrow from rats not so treated. Current experiments seem to indicate that pre-treatment of adult Sprague-Dawley donor rats with mouse lymphoid tissue is equally effective in enhancing the protective effect of the rat bone marrow when implanted in irradiated mice.

Plant Studies

The observation that phosphate accumulation was not inhibited by chloramphenicol was confirmed and extended by using low and high salt levels, low and high relative humidity, and double labeling with Rb^{86} and P^{32}. Other anions will be tested to determine whether this is a common difference between cations and anions or may be a difference specific for certain elements.

Carrier iodide was observed to increase uptake of KI^{131} from nutrient solutions, but toxicity at 1 μg iodide/ml prevented examination of an extensive range of specific activities. Increases in I^{131} uptake were noted up to the point of toxicity.

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Columbia River Limnology

One year's collection of data at the three sampling locations was completed. Analysis of plankton from the Vantage station was completed and counting of samples collected below 100-F was started. Preliminary results show a similar qualitative composition between the two stations; however, Asterionella, the dominant diatom, appears to be present in slightly higher numbers at 100-F.

Rattlesnake Springs Limnology

Eight species of cladocerans have been identified from the impoundment by Dr. R. W. Kiser. Mounted slides prepared by him have been compared with qualitative samples in preparation for zooplankton studies. One species of Ostracod identified from Rattlesnake Springs had previously been reported only from Yucatan and Trinidad. Dr. Edward Ferguson, who identified the animal, is preparing a manuscript reporting this range extension.

Plant Ecology

Chemical assays of greasewood and sagebrush leaves showed greasewood leaves to contain more calcium, potassium, and magnesium than sagebrush leaves and about 200 times more sodium. The soils supporting greasewood also contained more sodium than soils supporting sagebrush.

Moisture losses from soil filled cans in the field during the recent winter drought (Dec. 19 to Jan. 24) showed sagebrush soil more susceptible to winter moisture loss than greasewood soil. Greasewood soil lost 58 g of water while sagebrush soil lost 86 g.

Fallout

Radioiodine concentrations in North American deer and elk thyroids decreased during the month. Samples of Montana elk contained concentrations of I^{131} (avg. 1.6 nc/g) comparable to those of Maryland deer (avg. 1.1 nc/g). California deer thyroids were at the lower limit for I^{131} counting (0.05 nc/g).

Radiation Effects on Insects

Flour beetles, Tribolium castaneum, were given 940 rads of 3 to 4.8 Mev neutrons. After irradiation, the month old virgin beetles were established as single pairs and daily production of F_1 adults was recorded for two weeks. Irradiated groups produced significantly less progeny than the control group. Irradiation of males produced a greater effect on reproductive capacity than did irradiation of females. Least productive were cultures in which both sexes were irradiated.

R. Thompson

BIOLOGY LABORATORY

RC Thompson:es

January 1963

TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORY

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposiums

Eberhardt, L. L. Dynamics of ungulates - interpretation and needs. Western Association of Fish and Game Commissioners - Elk Committee Workshop, Meeker, Colorado. January 30, 1963.

Hanson, W. C. Project Chariot. Health Physics Society, Richland, Washington. January 25, 1963.

b. Seminars (Off-Site and Local)

Hungate, F. P. Implication of radiation on genetic systems. Pasco Educational Group, Pasco, Washington. January 16, 1963.

Mahlum, D. D. Some approaches to biological research. Science Fair Workshop, Columbia High School, Richland, Washington. January 28, 1963.

c. Seminars (Biology)

Watson, D. G. Sr^{90} in plants and animals of arctic Alaska. January 15, 1963.

Wilson, D. O. The cobalt requirement of symbiotically grown alfalfa. January 15, 1963.

Foster, R. F. Radioactive wastes at Hanford and its impact in the environs. January 22, 1963.

Uhler, R. L. Effect of drugs on active transport in plants. January 29, 1963.

Casey, H. W. Some aspects of the environmental monitoring program of the Christmas Island nuclear test series. January 29, 1963.

II. Articles Published

a. HW Documents

None

b. Open Literature

Hanson, W. C. 1962. Seasonal patterns of stable and radioactive iodine in thyroids of native jack rabbits. Transactions of the Twenty-Seventh North American Wildlife and Natural Resources Conference, March 12, 13, and 14, 1962: 225-232.

Rickard, W. H. 1962. Comparison of annual harvest yields in an arctic and a semi-desert community. Ecology 43: 770-771.

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II. Articles Published (continued)

b. Open Literature (continued)

Erdman, H. E. 1962. Effects of irradiation on the Mediterranean Meal Moth, *Ephestia kuehniella* Zeller, cultured on ^{89}Sr-spiked food. International J. Radiation Biology 5: 331-38.

McClellan, R. O., L. K. Bustad, W. J. Clarke, N. L. Dockum, J. R. McKenney, and H. A. Kornberg. 1962. Bone-seeking radionuclides in miniature swine, p. 341-348. In Some Aspects of Internal Irradiation. Pergamon Press.

Bair, W. J., A. D. Wiggins, and L. A. Temple. 1962. The effect of inhaled $Pu^{239}O_2$ on the life span of mice. Health Physics 8: 659-665.

Bair, W. J., J. P. Herring, and L. A. George. 1962. Retention, translocation, and excretion of inhaled $Pu^{239}O_2$. Health Physics 8: 639-650.

Park, J. F., D. H. Willard, S. Marks, J. E. West, G. S. Vogt, and W. J. Bair. 1962. Acute and chronic toxicity of inhaled plutonium in dogs. Health Physics 8: 651-658.

Ballou, J. E., L. A. George, II, and R. C. Thompson. 1962. The combined toxic effects of plutonium plus X-ray in rats. Health Physics 8: 581-587.

Bustad, L. K., W. J. Clarke, L. A. George II, V. G. Horstman, R. O. McClellan, R. L. Persing, L. J. Seigneur, and J. L. Terry. 1962. Preliminary observations on metabolism and toxicity of plutonium in miniature swine. Health Physics 8: 615-620.

Clarke, W. J. 1962. Comparative histopathology of Pu^{239}, Ra^{226}, and Sr^{90} in pig bone. Health Physics 8: 621-627.

Cable, J. W., V. G. Horstman, W. J. Clarke, and L. K. Bustad. 1962. Effects of intradermal injections of plutonium in swine. Health Physics 8: 629-634.

Erdman, H. E. 1962. Effects of ingested Pu^{239} on fecundity, fertility, and life span of Habrobracon (Hymenoptera: Braconidae). Health Physics 8: 635-638.

Ballou, J. E., W. J. Bair, A. C. Case and R. C. Thompson. 1962. Studies with neptunium in the rat. Health Physics 8: 685-688.

McClellan, R. O., H. W. Casey, and L. K. Bustad. 1962. Transfer of some transuranic elements to milk. Health Physics 8: 689-694.

Ballou, J. E. 1962. Preliminary evaluation of several chelating agents for plutonium removal. Health Physics 8: 731-737.

III. Visits and Visitors

a. Visits to Hanford

1-15-63 D. Luepje, AEC, Cincinnati, Ohio. Toured Biology facilities.

1-15-63 L. Bingham, AEC, Idaho Falls, Idaho. Toured Biology facilities.

1-15-63 J. Silhanek, USPHS, San Francisco. Discussed the impact of introduction of industrial wastes into aquatic environment and techniques of bioassay with fish, with RE Nakatani.

1-17-63 H. Rosenbaum and R. D. Phemister, Armed Forces Institute of Pathology, Washington, D. C. Consulted with Dr. W. J. Clarke on the availability of pathological material from radiation-damaged animal tissues to be included in the new Registry of Radiation Pathology to be established at the Armed Forces Institute of Pathology.

1-21 to 23-63 Dr. Lars Ekman, Royal Veterinary College, Stockholm, Sweden, now a Kellogg Fellow at University of Washington. Discussed research on cesium and iodine in large animals with L. K. Bustad, R. O. McClellan and other members of the Biology staff during tour of facilities.

1-23-63 W. E. Dodge and M. A. Radwan, U.S. Fish and Wildlife Service, and U.S. Forestry Service, Olympia, Washington. Discuss with L. K. Bustad and R. O. McClellan the use of labeled compounds in the study of animal repellents for use in forests. Techniques of study in laboratory and large animals were discussed.

1-29-63 N. Korniloff, C. M. Barnes, and Geo. Williams, U.S.A.F., Washington, D. C. Discuss research conducted on mibratory birds and their possible influence on spread of viruses with W. C. Hanson and F. P. Hungate. Col. Barnes also conferred with L. K. Bustad on collection techniques for studies in geographic pathology.

1-30-63 Drs. E. Weiss and Palotay, Washington State University, Pullman, Washington. Dr. Weiss is a visiting professor from Munich, Germany. They consulted with Dr. W. J. Clarke on comparative pathology of large animal and small animal lungs.

1-30-63 E. Montgomery and R. McCullough, General Electric, Syracuse, N.Y. Discussed a detection system satisfactory for identifying biological warfare agents with Dr. Hungate and other members of Biology staff.

1-31-63 S. Cohen, Washington State University, Pullman, Washington. Discuss problems regarding electron microscope application to biological materials with Dr. Hungate and R. F. Palmer.

III. Visits and Visitors (Continued)

b. Visits Off-Site

1-6-63 R. T. O'Brien visited Washington State University, Pullman, to consult with Dr. Higginbotham concerning arrangements for teaching a class.

1-11-63 L. K. Bustad discussed Pu release experiences and techniques used in X-irradiation of thyroid with Drs. Palumbo, Barker, and Figley at the University of Washington, Seattle, Wash.

1-14 to 17-63 J. K. Lund conferred on sample preparation for electron microscope examination with staff of Dr. R. L. Bacon, University of Oregon Medical School, Portland, Oregon.

1-27 to 2-1-63 L. L. Eberhardt discussed research at Department of Statistics, Colorado State University, Fort Collins, Colorado, and served on panel for discussing large animal populations at State of Colorado Department of Game and Fish, Meeker, Colorado.

1-28-63 W. C. Hanson discussed bird migration and its influence upon the spread of viruses with Dr. Donaldson, University of Washington Laboratory of Radiation Biology, Seattle, Washington.

1-28-63 L. K. Bustad discussed sample collection techniques in meeting with Dr. Donaldson, University of Washington Laboratory of Radiation Biology, Seattle, Washington.

IV. Achievements

No degrees were earned, nor did any professional licensing or certification occur.

V. Honors and Recognitions

None

VI. Professional Group or Organization Assignments

None

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APPLIED MATHEMATICS OPERATION

MONTHLY REPORT - JANUARY, 1963

ORGANIZATION AND PERSONNEL

Mr. Peter Riggle, who is on the Technical Graduate Program, started a rotation in Applied Mathematics, January 2, 1963.

OPERATIONS RESEARCH ACTIVITIES

The final revision of the report on "learning" patterns in HAPO production experience was completed except for the inclusion of certain 1962 data. These data will be added shortly. The revised report will be issued during February.

Arrangements were made during January to secure accounting data for the proposed flow-of-funds study of HAPO. The project will be under way in February.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

Irradiation Processing Department

Test Reactor reactivity data for enriched cores having different dimensions were used to revise the limits for the acceptance plan in use. In this plan, three cores are measured as a composite, with individual measurements being made only if the composite reading is outside specified limits.

A document, written jointly with IPD personnel, was issued updating post-irradiation results from the Quality Certification Program. In addition to "fitting" the data in assessing the significance of apparent trends and cycles over time, comparisons were made between reactors. Bumper columns were not included in the analyses.

The analysis of data from PT-423, a production test comparing oil quench with water quench fuels, was completed. A simple alternate charge design was utilized, permitting the comparisons to be made using differences between adjacent fuel elements as the statistics of interest.

The existing mathematical model expressing diameter change as a function of reactor variables was altered slightly to permit a reasonable extrapolation to zero residence time. This model is used to estimate changes in

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temperature imbalance occurring within a process tube as a function of time since the metal was charged.

A fair number of columns irradiated under the Quality Certification Program are bumper columns. Analyses are being made of data from these columns to compare their behavior with nonbumpers.

Assistance was given in evaluating deflection vs. load data to determine if up-sized rails perform the same as rails not requiring up-sizing.

Additional data are being analyzed from the experimentation performed to locate an optimum set of process conditions for the hot-die-sizing process.

In order to maintain effective control over measurement biases which may occur in the C-Basin Examination Facility, a control program is being set up. In addition to maintaining control through routine remeasurement of irradiated production fuels, a set of standards is being fabricated having a wide range of diameters, lengths, and warp. These standards will also be useful in establishing and controlling biases between pre- and post-irradiation facilities. Consulting assistance is being given in connection with this program.

The warp which might be expected to occur in a 10-inch fuel element was estimated assuming a pessimistic situation, viz., that the bending would continue in the same direction. This was in connection with preliminary evaluation of the consequences of using 10-inch fuel elements.

As a follow-up to the study performed on ledge corrosion data in December, a similar study is being made of groove corrosion occuring on the fuel elements.

Consulting assistance was provided in connection with proposed in-reactor testing of anodized process tubes. Sample sizes required to detect differences in corrosion attack of practical importance were computed.

The document presenting the probabilistic functions which describe unscheduled reactor outages was issued. These functions were derived for use in generating outages in the Reactor Simulation Study. Also in connection with this study, all machine language source files have been maintained, and are now complete through 1962. The analytical program to delineate outage activity duration and frequency is now in the debug phase.

Extensive analyses were made of dosimeter badge and pencil data for September through December. Estimates were made of the biases between badge and pencil readings, and of the precisions of both measurements. Also, comparisons were made of areas and of crafts.

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N-Reactor Department

A program to estimate crack distribution and frequency for NPR primary piping from ten-sample data was modified to accept three-sample data. Negative values computed in the main program invalidated results. A NELLY Program (nonlinear least squares) failed to converge for several cases. New assumptions in the model were made and parameter estimates from three-sample data were found by solution of a cubic. An iterative method is now being applied to the ten-sample data.

During the past month, work was begun on the N-Reactor reliability study which is to be made through use of four-state reliability algebra. Progress to date has been in the area of scoping the analysis, investigation of the physical and logical relationships between the reactor subsystems, and logical representation of the criteria used to determine the reactor's mode of operation.

Chemical Processing Department

A document was issued giving comprehensive rail accident statistics. Various cause, damage, and speed relationships were discussed in this document. These will be of use in reaching decisions regarding procedures to follow in shipping radioactive materials.

A report was issued regarding the effects of gauging measurement errors on the producer's and consumer's losses resulting from the final inspection and certification of parts.

A brief examination was made of impurity data from reject buttons to determine the effects of possible nonhomogeneity of button impurities in correctly estimating the impurity content.

The relationship between type of ingot feed and the radiographic reject rate of castings was investigated.

Work resumed on the refinement and programming of a mathematical model of spare parts and general inventory control.

Construction Engineering and Utilities Operation

Minor injury frequency data for 1962 were analyzed to determine if the consistently lower-than-HAPO frequencies reported are explained by differences in the nature of the work within the Department. Expected rates were estimated for each Section based on observed rates for similar components outside the Operation, and the results were combined to give an over-all expected rate to which the observed rate could be compared.

Relations Operation

The study of security infractions at HAPO was completed and an oral presentation of results was made. The study was expanded to include a review of HAPO incidents involving lost or forgotten badges during 1962. Study results were put into an informal written report which included suggestions for modification of current data collection efforts.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HL

2000 Program

NPR Reactor

A mathematical model of the NPR stack gas chemical reactions has been completed as part of the general study of the zirconium-graphite compatibility problem. The model consists of a simultaneous set of first order nonlinear differential equations whose solutions yield the stack gas composition as a function of position in the reactor. An attempt is being made to study these solutions as a function of the controllable parameters on analogue computing equipment. A digital program is also being written so that the findings of the analogue studies can be investigated in greater detail.

Pulse Column Facility

Further analysis was done of data from five equilibrium runs on the pulse column using the BIMD power spectrum estimation program. Interpretation of the results of the analysis is now in progress.

A new model was fitted to two sets of gamma absorptiometer calibration data. Each set of data consisted of calibrations on four sample cells each run four times at nine uranium concentrations. The new model is a modification of Beer's law to include a second exponential term representing a scattering of gamma photons at higher uranium concentrations.

Other

A solution to the problem of determining the heat transfer from a buried pipe was obtained and furnished the requesting customer.

A formula used to relate NO_3 to Pu and HNO_3 was studied by comparison with sample data to determine if it is unbiased, and to determine the precision of the sample points.

3000 Program

The results of a series of experimental runs of shear-spinning certain metallic shapes has led the customer to choose a preferable method of designing the original blanks. The mathematics for this mode of design has been formalized and programmed for the digital computer. This program computes the details of the blank contours and the requisite tool center coordinates for either machining blanks directly or forming appropriate molds in which blanks may be cast.

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4000 Program

Nondestructive Testing

The EDPM program which produces tables of the Lamb wave frequency equation for arbitrary elastic material constants is virtually complete. These tables permit a more accurate interpretation of experimental data obtained by ultrasonic nondestructive testing devices which employ Lamb waves. A detailed study is now under way to investigate the reflective, refractive, and absorptive behavior of sound waves at a liquid-solid interface.

Ceramic Fuels Research

A series of particle size-proportionate mix designs were concocted so that alternative methods might be available for producing high density vibrationally compacted fuel elements. The possession of a spectrum of almost equally desirable designs should allow a more efficient utilization of crushed materials with a consequent minimization of reworking. Statistical experiments are in process to determine the relative concentrations of different size fractions which produce maximum density.

5000 Program

Actinide Element Research

The program to index hexagonal crystals was made operational during January. A section was added to the program which computes the standard deviations of the lattice constant estimates. Approximately ten sample cases have been successfully indexed. These were designed to test the program's ability to index crystals where the first few lines were dependent. With all major changes in logic complete, the program advanced to its second phase. This entails the development of criteria to evaluate the validity of solutions and to determine an allowable error in these solutions. A set of test data is now being developed which contains a known amount of random fluctuation in the 2θ diffraction readings.

While major attention was focused on the hexagonal-tetragonal case, some work was done on the orthorhombic problem. Several test decks ran successfully which verified the program's ability to cope with dependent lines.

Computation and Statistical Analysis

The revision of the Gatlinburg paper, "Quantitative Analysis of Sets of Multicomponent Time Dependent Spectra From Decay of Radionuclides" was submitted to Technical Publications as formal Hanford report HW-75806. The GEM program for performing the quantitative analysis has been placed in the EDP FORTRAN Library for use by HAPO personnel interested in the resolution [redacted]e-height spectra data.

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Several additional tables were completed for inclusion in the formal report "Fixed Time Count Rate Estimation with Background Corrections".

Radiochemical Analysis

Debugging of the IRA II data analysis system is approximately 75 percent complete. All but the following three passes are working satisfactorily: IRA 325 which maintains a file of raw data on samples currently being analyzed; IRA 335, which estimates a time zero decay rate; and IRA 525, which maintains the file of reduced data. The conversion of the IRA I master file to the IRA II master file appears to be working according to specification. Personnel responsible for data acquisition and analysis for the new system spent time familiarizing themselves with the logic of the new system.

General

Personnel Monitoring

Currently, Pu d/m in urine samples are found separately for each individual. A study is being made to see if compositing of samples can be used to good advantage. A distribution of sample values for all of 1961 data has been obtained, and is being used in a program developed to determine the probabilities of failing to detect high readings under various situations. Monte Carlo techniques are being used.

A program to print out a table of calculated density versus dose for film badges is now being written.

Other

Analysis of data from an experiment to determine the precision and accuracy of the azure-C method of estimating trace Boron content as a contaminant in aluminum rods was completed.

Carl A. Bennett

Manager
APPLIED MATHEMATICS

CA Bennett:dgl

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REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

UCOST-QUICK Study

Computations have been made to determine the effects of different uranium price schedules upon total reactor fuel costs. In particular, toll separation operation of the diffusion cascade -- as opposed to operation with an optimum tails composition -- was simulated. Fuel costs were obtained by assuming twelve different uranium price schedules and eighteen economic environments for each fueling mode.

A range of initial enrichments in each of two reactor types were calculated for both batch and graded irradiation. The reactor types were selected from the Combined Cycles Study and are (1) a water moderated, stainless steel clad, uranium enriched type; and (2) a water moderated, Zircaloy clad, uranium enriched type. The moderator index chosen for each corresponds roughly to that of a Boiling Water Reactor.

Table I contains some preliminary results to show the effect of optimized versus toll separation operation of the diffusion cascade. A standard economic environment, a high fabrication cost, and graded operation of the reactors was assumed. The two reactor types included in the study are identified by their respective cladding materials.

The data of Table I show that as the cost of enriched uranium is increased -- either by increasing the cost of feed to the cascade or by operating the cascade in a nonoptimum manner -- the fuel cost is increased, while the optimum initial enrichment decreases. The former effect is more pronounced in the stainless steel clad type which has higher fuel costs, while the latter effect is larger in the Zircaloy clad type which has lower optimum enrichments. However, these variations are less than the differences between reactor types. These data do not include examples of the situation wherein a cascade optimized for $23.50 per kilogram of natural uranium feed is operated on a toll basis with lower cost natural uranium feeds. This may be a practical situation in the near future and is part of the UCOST study. The data shown on Table I were selected to emphasize the importance of operating the cascade at optimum tails.

The UCOST-QUICK study will be extended to include a reactor type that is representative of a Pressurized Water Reactor; namely, a water moderated, Hastelloy clad, uranium enriched reactor. In addition, different toll separations situations will be investigated for selected enrichments in each reactor type.

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TABLE I

THE EFFECT OF DIFFERENT URANIUM PRICE SCHEDULES ON THE TOTAL FUEL COST OF WATER MODERATED, URANIUM ENRICHED REACTORS

FEFJ = \$60 per pound of fuel
Depreciating interest rate = 12.5%
Nondepreciating interest rate = 4.75%
Cost of separative duty = \$30/kg of uranium

Cost of Natural Uranium as UF_6, \$/kg U	Reactor Type: Stainless Clad, Min TFC m/kwh_e	Stainless Clad, Percent Enrichment	Zircaloy Clad, Min TFC m/kwh_e	Zircaloy Clad, Percent Enrichment
Cascade using optimized tails composition				
23.50	1.425	3.22	1.130	2.38
19.09	1.329	3.26	1.058	2.48
14.68	1.226	3.36	0.981	2.63
Cascade using fixed tails composition of 0.5%				
23.50	1.776	2.91	1.361	1.99
19.09	1.593	3.02	1.238	2.13
14.68	1.407	3.18	1.108	2.34

Please note that the base set of uranium price schedule parameters for the UCOST-QUICK study, as reported in the Programming Monthly Report for December 1962, contains an error. The costs of feed as U_3O_8 corresponding to the given costs of UF_6 should be \$8.00, \$6.30, and \$4.60 per pound U_3O_8 instead of the \$8.00, \$6.00, and \$4.00 reported, which were rounded off. The cost of conversion from U_3O_8 to UF_6 should have been \$1.23 per pound uranium.

A Price Schedule for U-233-U-238 Mixtures

In certain circumstances, it may be attractive to enrich depleted uranium with the U-233 bred in a thorium blanket. In order to calculate the burn up cost of such a fuel loading, the cost of separating the U-233 from the U-238 in the irradiated fuel must be determined. That is, a schedule of prices based on a specified price and enrichment and on a specified separative duty cost must be constructed. For simplicity, it will be assumed that the cascade will be operated with optimum tails composition, that the U-234, U-235, and U-236 in the fuel can be ignored, and that the unit cost of separative duty is proportional to the relative difficulty of separating U-233 from U-238 compared with U-235 from U-238.

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One of the basic factors in the AEC Uranium Price Schedule is that the energy required to effect a separation of isotopes in a diffusion cascade is directly proportional to the total flow rate within the cascade. A formula for this quantity of an ideal cascade with single waste, feed, and product streams is:

$$J = \frac{\beta + 1}{(\beta - 1)\ln \beta} \Delta \qquad (1)$$

where

J = total flow rate in an ideal cascade, kg uranium/unit time
Δ = separative duty, kg uranium/unit time
β = heads separation factor.

The separative duty is a function of the waste, feed, and product flow rates and their compositions and does not depend upon the isotopes being separated. Therefore, the ratio of the energies required for two different systems (denoted as 1 and 2) is for heads separation factors very near 1:

$$R = \frac{(\beta_1 + 1)(\beta_2 - 1)\ln \beta_2}{(\beta_1 - 1)(\beta_2 + 1)\ln \beta_1}\left[\frac{\beta_2 - 1}{\beta_1 - 1}\right]^2 \qquad (2)$$

One of the key properties of an ideal cascade is that $\beta = \sqrt{\alpha}$ where α, the stage separations factor, equals the square root of the ratio of the molecular weights of the isotopes being separated. Thus, for U-235-U-238 systems, β = 1.00214 while for U-233-U-238 systems, β = 1.00358. Using these numbers in Equation (2) gives R = 0.357. This means that the separation of U-233 from U-238 requires roughly one-third of the energy required to separate U-235 from U-238 or, based on the present AEC Separative Duty Cost of $30/kg U, $10.71/kg U.

A uranium price schedule is usually established by specifying a feed composition and unit cost (e.g., natural uranium at $23.50/kg), a unit cost of separative duty (e.g., $30/kg), and by specifying that the tails stream will be valueless and will have the optimum composition to minimize the product cost. An identical schedule can be derived by considering a product composition to be the feed composition provided that the "feed" price is the same as would have been obtained in the former case. This principle can be used to derive a schedule for a U-233-U-238 system, given a value for the U-233 bred in a thorium blanket. Table II contains schedules calculated by assigning different values to the U-233 mixed with U-238 to a two percent enrichment. Also shown is the cost of a 50 percent fissile burn up in each schedule.

TABLE II

EXAMPLE PRICE SCHEDULES FOR U-233-U-238 MIXTURES

Unit Cost of Separative Duty = $10.71 per kg Uranium

Percent Enrichment % U-233 in U-238	Unit Value of U-233 at 2% Enrichment, $/gram			
	$7.85	$10.00	$12.63	$15.00
	Optimum Tails Composition, Percent			
	0.108%	0.088%	0.072%	0.062%
	Prices Per Gram of U-233 Contained			
0.2	1.26	2.42	4.07	5.71
0.5	4.50	6.32	8.62	10.75
1.0	6.48	8.52	11.04	13.33
2.0	7.85	10.00	12.63	15.00
99.0	10.00	12.26	15.00	17.45
	Cost of Burning From 2% to 1%/gm, $/gm U-233 Destroyed			
	9.13	11.46	14.18	16.52

Several items are worthy of note from Table II. First, that similarly to the U-235-U-238 system, if U-233 is mixed with U-238 to produce two percent enrichment and then burned to one percent, the burn up costs are greater than the cost of U-233 in two percent enriched fuel. This emphasizes the reduction in U-233 value that must be compensated by some other factor if such fuel cycles are to be advantageous. This points up that the chemical separability of plutonium from U-238 is a significant asset. The second item to note is that extremely low tails compositions are involved as compared to a U-235-U-238 separations because of the low unit cost of separative duty for U-233. This low value may not be achievable in practice because of inventory and clean up problems involved with converting the cascade operation from U-235-U-238 to U-233-U-238 and back again.

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It is planned to incorporate a price schedule of this type into the QUICK program in order to determine the minimum total fuel cost for this fueling mode. The schedule will be constructed such that the U-233 contained in the initial enrichment of the reactor will have an assigned price, the unit cost of separative duty will be 35.7 percent of the specified value, and the tails composition will be optimized. This will require rewriting the SKED subroutine of the QUICK code for this special case because a different price schedule must be used for each initial enrichment. Otherwise, an iterative calculation in which the schedule parameters would be varied must be made.

Plutonium Values and U-235 Burn Up Costs in Specific Cycle Analysis

When analyzing plutonium values, it is logical to compare them to the costs of burning U-235 in equivalent circumstances. An opportunity to do so in a variety of situations is offered by the data in Table III. These data represent a sizable computation of plutonium value for a reasonable spread of reactor types and economic cases.

The computed plutonium values are compared by forming two ratios: (1) the computed plutonium fuel value to the price of 90 percent enriched uranium, and (2) the computed plutonium fuel values to the uranium burn up cost for each base step. By placing the comparable cases in descending order with respect to either ratio, the correspondence of the ratios can be observed. By observing Table III, one notes that there may well be an underlying relationship between computed values and the price of 90 percent enriched uranium. On the other hand, there appears to be little promise of a basic relationship between computed plutonium values and the corresponding uranium burn up costs. The definitions and other particulars upon which these calculations are based are described in HW-72217 and the complete data are listed in HW-76195.

Combined Cycle Fuel Costs

Fuel cost calculation for two simulated reactors using the cross progeny of U-238 and Th-232 bred fuel have been extended to better show differences in performance. Last month batch and graded operation was considered under equivalent physics conditions (specific power = 10 MW, ϵ for U-238 systems 1.03, and standard Westcott cross section) and with no limit on fuel exposure. Some of the more important parameters pertaining in each fueling have been varied as shown in Table IV. The fuel costs in Table IV are computed with U-233 priced at $14/gram and Pu-239 and Pu-241 at $10.24/gram. To have equal fuel costs in the majority of cases, it would be necessary to raise the price of plutonium relative to U-233. An exception to this is U-235 enriched U-238 whose fuel costs would go down further if the plutonium price were increased. Therefore, more equitable adjustment may be to lower U-233 prices relative to plutonium. After more of the physics cases are calculated an over-all analysis will be made to determine plutonium and U-233 prices relative to the price of the uranium from the cascade that will lead to equal fuel costs for each reactor situation.

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TABLE III

COMPARISON OF URANIUM BURN UP COST IN BASE STEP WITH COMPUTED VALUE OF PLUTONIUM DISCHARGED FROM THE FIRST (BASE) STEP AS ENRICHMENT FOR SUCCESSIVE STEPS IN SEVERAL REACTOR SIMULATIONS

Reactor Simulations	Initial U-235 Enrichment, w/o for Base Step	Uranium Burn Up Cost in Base Step $/gram U-235 Burned	Computed Pu Value for First Recycle Step, $/gram Fissile	Ratio of Pu Value to Price of 90% Enriched Uranium*	Ratio of Pu Value to U Burn Up Cost
(A) Self-Produced Plutonium Recycle - Batch**					
GCS	2.28	9.37	12.69	1.06	1.35
OMS**-3	2.74	10.18	12.49	1.04	1.23
OMS	2.38	10.06	12.35	1.03	1.23
BWS**-2	2.72	10.33	12.15	1.01	1.18
APWS	2.62	10.22	12.14	1.01	1.19
OMS**-2	2.55	10.23	12.13	1.01	1.19
BWS	1.66	9.00	11.85	0.99	1.32
HWS	1.31	7.92	11.53	0.96	1.46
BWS**-3	1.53	8.57	11.22	0.93	1.31
(B) Self-Produced Plutonium Recycle - Graded**					
GCS	1.54	5.61	11.77	0.98	2.10
APWS	2.48	9.51	11.52	0.96	1.21
OMS	2.33	9.42	10.45	0.87	1.11
BWS	1.59	7.71	10.29	0.86	1.33
HWS	1.31	5.66	10.29	0.86	1.82
(C) Optimum Plutonium Recycle in Natural Uranium - Batch**					
OMS	2.30	10.01	12.75	1.06	1.27
GCS	1.49	8.49	11.16	0.93	1.31
APWS	2.65	10.23	11.12	0.93	1.09
BWS	1.67	9.01	10.28	0.86	1.14
HWS	1.36	8.01	10.15	0.85	1.27
(D) Optimum Plutonium Recycle in Natural Uranium - Graded**					
GCS	1.47	5.77	11.27	0.94	1.95
BWS	1.56	7.71	9.82	0.82	1.27
HWS	1.44	6.28	7.33	0.61	1.17

* Listed in descending order of this ratio to facilitate comparisons.
** See the next page for a description of each simulation and major approximation of the fuel cycles.

** MELEAGER, the code used to provide the fuel burn up data does not account for details of the reactor design and uses several integral reactor parameters to simulate the different reactors. As elaborated in HW-72217, the simulations are as follows:

APWS - Simulated Pressurized Water Reactor as described in TID-8502.

BWS - Simulated Boiling Water Reactor with Zirconium Tubes - not of specific design.

HWS - Simulated Deuterium Oxide Moderated Reactor with Zirconium Tubes - not of a specific design.

GCS - Simulated Gas-Cooled Graphite Reactor - a low temperature moderator type patterned after "Bradwell".

OMS - Simulated Organic Moderated Reactor as described in TID-8501.

There are several special variations to some of these simulations as follows:

OMS-2 - This simulation is the same as the OMS, except that the moderating power was increased to the value used in the BWS.

OMS-3 - This simulation is the same as the OMS-2, except that the specific power was increased by a factor of 3.

BWS-2 - This simulation is the same as the BWS, except that the parasitic absorption index was increased to that of the APWS.

BWS-3 - This simulation is the same as the BWS, except that the moderator index was increased to that of the APWS.

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TABLE IV

FUEL COST FOR VARIOUS FUELS (Mills/kwh_e)

Parameter Variation	Fissile-Fertile Fuel: U-235 in U-238 Batch	U-235 in U-238 Graded	U-233 in U-238* Batch	U-233 in U-238* Graded	Pu in U-238 Batch	Pu in U-238 Graded	U-235 in Th-232 Batch	U-235 in Th-232 Graded	U-233 in Th-232 Batch	U-233 in Th-232 Graded	Pu in Th-232* Batch	Pu in Th-232* Graded
Simulated Water Moderated Reactor with Stainless Steel Jacketing with Variation of Some Key Fueling Parameters												
Standard	2.05	1.52	2.24	1.71	2.01	1.48	2.48	2.09	2.42	2.02	2.20	1.84
ϵ = 1.06			2.10	1.56	1.84	1.33						
ϵ = 1.10	1.72	1.23	1.88	1.39	1.66	1.16						
Specific Power = 20 MW/T							1.93	1.58	1.74	1.50	1.76	1.30
Specific Power = 40 MW/T							1.49	1.38	1.37	1.37	1.44	0.96
$Alpha_{49}$ = 0.42					1.79	1.27					2.05	1.63
$Alpha_{49}$ = 0.42 and ϵ = 1.10					1.49	1.00						
Simulated Water Moderated Reactor with Zirconium Jacketing with Variation of Some Key Fueling Parameters												
Standard	1.59	0.96	1.78	1.15	1.72	1.03	1.77	1.45	1.62	1.35	2.00	1.39
ϵ = 1.1	1.21	0.76	1.40	0.87								
Specific Power = 20 MW/T									--	0.90		
Specific Power = 40 MW/T							1.78	0.89	--	0.79		
$Alpha_{49}$ = 0.42					1.50	0.84					1.28	1.19

* Fuel cost value of these fuels are preliminary and will be improved at a later date.

TABLE V

FUEL EXPOSURES WITH ZONED REACTORS

(For various allowed average excess reactivities.)

Case Number	Allowed Average Excess Reactivity	Fuel Exposure MWD/T	Number of Zones Used
1	100 mk	13,500	3
2	28 mk	20,500	9
3*	18 mk	21,070	10

* Reactivity increase controlled rather than reactivity decrease as in 2 and 1. Fuel costs will be calculated after final debugging of the physics routine is completed.

Computer Code Development

1. PROTEUS Code

It is planned to incorporate a flux weighted reactivity for graded operation into the PROTEUS code. The average reactivity presently calculated by the MELEAGER code reflects the fact that the reaction rate is constant with respect to time which overemphasizes the contribution of the high k_{∞} fuel. This simplification leads to a small error for the reactors studied to date which have low conversion ratios. A more correct formulation will balance the neutron production and absorption by applying a flux weighting to the reactivity averaging. A practical problem involved in the programming is that only the time averaged flux is available to PROTEUS and various differentiation schemes are being considered.

2. MELEAGER-ALTHAEA Code

A revised MELEAGER-ALTHAEA code has been written and debugging is in process. This code includes correction for xenon and samarium neutron absorption, which eliminates the need of doing this on a k_{∞} basis. The output includes flux time weighted averages of reactivity; absorptions in each isotope of the fuel; absorptions in fertile, fissile, fission product, and structural isotope groups; and the heat produced by each fuel isotope. Instantaneous values of the above averages plus conversion ratios for each fissile isotope are also given.

The fast fission factor, ϵ, was increased for the U-238 system to demonstrate possible fuel cost reduction. The specific power was increased in the Th-232 system, where Pa-233 decay to U-233 is of prime importance. A significant change in the alpha for Pu-239 can be realized by proper shielding techniques. This involves self-shielding in the resonance region where alpha is higher. The effective Pu-239 alpha is normally about 0.54; whereas, the shielded value may be as low as 0.42.

Because some of the cases above lead to breeder operation, it was necessary to eliminate the fuel going to unlimited exposure. Hence, an arbitrary limit was set at 30,000 MWD/T for both batch and graded operation. Thus, all fuel costs quoted are at exposure of 30,000 or less.

As in the December 1962 report, it must be noted that these fuels are used in reactor types none of which may be optimized to burn any particular fuel considered. The results should be interpreted as showing trends for each fuel type in the reactor considered under the economic conditions considered, but not showing which fueling system is generally superior.

Many of the possible variations have not been calculated at this time, but as computer time is available more of them will be completed.

Zoned Spectrum

The reactivity of any fuel element is a function of the neutron spectrum. As the spectrum is changed, the relative fission to absorption rates are altered. By making frequent spectrum adjustments, the excess reactivity can be minimized so that excess neutrons are absorbed in the fuel instead of a control system. Such systems achieve essentially the same results as the graded or continuous discharge schemes, but may be superior from a control and power flattening basis.

Reactors can be designed so that various regions or zones will have different neutron spectrums. A reactor of this type has been simulated by the MELEAGER burn up code and some calculations have been made showing the effect of minimizing the excess reactivity by using a zoned reactor and shifting the fuel from zone to zone with different allowed reactivities.

The reactor type used in these calculations was a near breeder and the fuel reactivity would normally rise for a while as burn up proceeded. This effect tended to amplify the excess reactivity problem. So, in case 3 the spectrum was adjusted to limit this increase also. The result of these calculations are shown in Table V for various allowed average excess activities.

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Miscellaneous

Twelve copies of a detailed uranium price schedule were distributed on the plant. These data, which were obtained from the UCOST computer code, are stored on magnetic tape. Additional copies can be obtained for the cost of binding. The computer time required to solve for the optimum tails composition and to calculate the prices and feed requirements of 688 enrichments was approximately 20 seconds.

Evaluations and Studies

At the request of the Washington AEC Office of DRD, a study of the feasibility of producing heat source radioisotopes in civilian power reactors was begun. The study will emphasize curium-244 with some effort applied to uranium-232 (and thorium-228). Such production would be incidental to the primary purposes of power reactors but may be viewed as a source of beneficial economic effects.

The study would not cover short-lived heat sources such as Cm-242 or Po-210, but may cover valuable long-lived intermediates such as Am-241 and Am-243. Fission products are not to be considered within the scope of this study.

REACTOR TECHNOLOGY DEVELOPMENT - 02 PROGRAM

A study of the significance of the thorium-230 content of commercially available thorium as a factor in the formation of uranium-232 contamination in uranium-233 was initiated. Since thorium is normally derived from the mineral monazite as well as from the waste streams from certain uranium ore milling processes, the thorium-230 content, though small, is expected to show wide variations depending on the source. The Th-230 concentration will depend on the uranium to thorium ratio in the source mineral since the Th-230 originates only via the radioactive decay of uranium.

Although n,2n reactions on thorium-232 and uranium-233 have been commonly considered responsible for the appearance of U-232 in fissionable U-233 produced by neutron irradiation of Th-232, essentially no attention appears to have been given to U-232 production via successive neutron captures on Th-230 and the resulting Pa-231. In view of anticipated Th-230 concentrations in Th-232 ranging from a few tenths to possibly one hundred parts per million, significant differences in resulting U-232 concentration are also expected.

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By analysis of samples of commercial thorium obtained from deposits identified as South African and Australian monazite as well as material obtained as a by-product from Canadian uranium ore milling operations, the Th-230 content of these samples is being obtained. Irradiation of samples will then be conducted to provide material in which U-232 contents of the resulting U-233 may be determined. By such analyses, the relationship between Th-230 concentration and the development of U-232 in fissionable U-233 produced by the large-scale irradiation of thorium will be more firmly established.

W. K. Woods

Manager,
Programming

WK Woods:jm

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF JANUARY 1963

A. ORGANIZATION AND PERSONNEL

Carlos E. Newton joined the Radiation Protection Operation as Manager, Composite Dose Studies and Records. J. Walter Vanderbeek transferred from Environmental Studies and Evaluation to the N Reactor Department. David L. Silver was reactivated from military leave into Internal Dosimetry. William C. Wann transferred from Internal Dosimetry to Reactor and Fuels Laboratory. Transfers within the Section included James R. Bovingdon, Joseph R. Berry and Irving L. Winter transferring from Radiation Monitoring to Environmental Studies and Evaluation. Daniel McConnon, William E. Parker and James A. Deardorff transferred from Environmental Studies and Evaluation to Radiation Monitoring. Leo G. Faust transferred from Radiological Development and Calibrations to Radiation Monitoring; and Herman J. Paas, Jr. transferred from Radiation Monitoring to Radiological Development and Calibrations.

B. ACTIVITIES

Occupational Exposure Experience

A CPD process operator assigned to the 234-5 Building sustained a contaminated puncture wound in the hand in June 1962. Surgical excision of the contaminated area at the time of the injury was successful in removing about 1 μc Pu. Subsequently, .01 μc Pu was excreted as a result of treatment with DTPA. The medical treatment with DTPA delayed the evaluation of the employee's body burden. The current evaluation now shows that the body burden is 0.018 μc Pu or about 45% of the maximum permissible body burden. The original wound site before excision was estimated to have 25-30 times the maximum permissible body burden.

During the month four CPD employees at the 234-5 Building and one HL employee at the 308 Building received plutonium contaminated skin injuries. Tissue excision was performed by an industrial physician to reduce the contamination in the wounds for three of the CPD employees and DTPA was administered in two of these cases. The maximum plutonium contamination detected in the injuries was 1.1 μc, based on wound counter measurements. The maximum quantity remaining in the injuries was 8×10^{-4} μc. Three of the CPD employees were previous plutonium deposition cases with body burdens estimated to be less than ten percent of the MPBB.

The total number of plutonium deposition cases that have occurred at Hanford is 316 of which 230 are currently employed. The number of new deposition cases that occurred in CY-1962 was 32, all resulting from reported contamination incidents.

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Four new plutonium deposition cases were confirmed by bioassay analyses during the month of January. The new deposition cases resulted from three previously reported contamination incidents involving CPD employees at the 234-5 Building in November and December, and one contamination incident that occurred this month at the same facility. In three of the cases, including the one that occurred in January, inhalation was attributed as the mode of intake and the plutonium body burden was estimated to be one percent or less of the permissible body burden (MPBB = 0.04 μc). The fourth case occurred as the result of a plutonium contaminated injury received by a CPD process operator in November. Medical excision of tissue at the time of the accident reduced the plutonium contamination in the puncture wound from 10^{-2} μc to 10^{-3} μc, as measured by the wound counter.

Abnormally high radioactivity in the PRTR primary coolant followed by stagnation water tests in the process tubes, led to the removal of three fuel elements suspected of rupture. The three elements were placed in the storage basin and will be inspected at a later date. Unusual radioactivity in the steam from the PRTR condenser prompted a sampling program to identify the contaminants and their origin as concentrations of 1×10^{-7} μc/cm^3 were encountered with a half-life of 10-14 minutes. Fuel element movement during process tube inspection involved beams above the tubes to 30 rads/hour and average dose rates in the work area of 15 mr/hour. Smearable contamination to 100 mrads/hour and surface dose rates to 2 rads/hour were experienced during primary pump maintenance. The addition of new, low tritium content, heavy water to the heavy water systems has reduced the tritium exposure potential. The approximate tritium concentrations in the primary, moderator, and reflector systems are 50 μc/cc, 400 μc/cc and 400 μc/cc, respectively.

During the purification of previously separated promethium solution in C cell of the 325-A Building, a radiation measurement of 100 rads/hour was detected at the surface of the columns with corresponding momentary personnel dose rates of 500 mrads/hour through the open cell door. The purified promethium was subsequently precipitated, encapsulated and removed from the cell. A radiation measurement of 40 rads/hour was obtained at one inch from the promethium in the capsule.

Environmental Experience

About 17 mc of mixed fission products were emitted from the 327 Building stack during the 24-hour period ending at 8:00 a.m. January 8. Another 10 mc were emitted during the 24-hour period ending at 8:00 a.m. January 18. No environmental contamination was detected on follow-up surveys. Further investigation of operational performance did not reveal the cause of the emissions.

Two routine aerial surveys were made during the month. One flight followed standard pattern 2-E to Ellensburg, Moses Lake, Ritzville and return. The other flight followed the Columbia River to Astoria. Activity levels between Richland and Arlington were significantly higher than observed on previous survey flights. The water level was noticeably low below McNary Dam exposing more of the islands in that portion of the river. Radiation levels over these islands were also noticeably higher than those observed on previous flights. A portion of the estuary area was resurveyed and locations indicating significantly higher radiation levels were identical with those observed on previous surveys.

Average particulate fallout concentrations at various locations in the Pacific Northwest ranged from 2.0 to 10 μμc gross beta/m^3 of air.

The Radiochemical Analysis Operation identified an unaccounted-for-fraction of beta activity present in caustic scrubber samples from the separations area stacks as C^{14}. A plausible source could be an (n-p) reaction with nitrogen impurities in the uranium fuel during irradiation. The preliminary data indicate that releases on the order of one curie of C^{14} may occur on some days. Additional analyses are scheduled in order to characterize the emissions more precisely.

The estimated exposure to the GI tract from drinking Pasco sanitary water was as low in December as in any month during the past five years. This estimated dose (1.2 mrem) was about ten percent of the December 1960 value (11.4 mrem).

Studies and Improvements

All action required by the Hanford Laboratories to achieve the desired level of capability for coping with a serious radiation event, as described in HW-72200-D, is completed.

A new X-ray machine for routine use in identifying personnel neutron dosimeters and to provide back-up equipment for identifying personnel beta-gamma dosimeters was installed in the 3705 Building.

Bids for the fabrication of the Scintillation Portable Poppies were received from four manufacturers. They ranged from a low bid of about $270 each to a high bid of $446 each. The contract for the fabrication of 30 Scintillation Portable Poppies was awarded to the low bidder, K. F. Products, Inc., of Denver, Colorado. All supporting items including a prototype instrument for copy were shipped to the fabricator on January 15.

The plutonium-beryllium neutron source (a right circular cylinder with a diameter of 1.32 inches and a height of 2.69 inches) previously returned to Mound Laboratories for inspection and repair was received from the U.S. National Bureau of Standards following calibration. The certified neutron flux value was 9.53×10^6 n/sec. A small aluminum can container with a bail is being designed to facilitate handling of the source.

An r-meter check (r-meter calibrated by the U.S. National Bureau of Standards) of the portable instrument calibration positions for both calibration wells was completed. The calibration wells were accurate to within ± 3% of the stated dose rate at all calibration positions.

Two calibration checks were performed during the last quarter to relate the response of personnel dosimeter pencils, personnel beta-gamma film badge dosimeters and portable dose rate monitoring instruments when exposed to a common gamma radiation source on a common calibration jig. For this experiment, all of the dosimeters were exposed both on the gamma film calibration jig and on the instrument calibration wells. For radiation exposures above 20 mr, all of these dosimeters indicated within ± 10% of the applied dose with the majority of them falling within ± 5% of the applied dose.

On January 24, 16 Hanford Criticality Dosimeters, 16 personnel beta-gamma film badge dosimeters and 16 personnel neutron film badge dosimeters were exposed at the Sandia Pulsed Reactor facility at Albuquerque, New Mexico. Neutron doses up to about 5000 rads and gamma doses up to about 250 r were delivered during two reactor bursts. The evaluation of the data attained during this experiment is not nearly completed; however, the following preliminary results were noted:

1) All personnel beta-gamma film badge dosimeters gave GM readings above 10,000 c/m following their exposure. The lowest estimated neutron dose received was about 60 rads which resulted in a GM reading of 1.7×10^4 c/m. This would indicate that "quick-sort" procedures using the induced radioactivity in the beta-gamma personnel dosimeter could be performed down to exposure doses of a few rads of fission spectrum neutrons.

2) The total rad-neutron dose measured with the Hanford Criticality Dosimeter was within the ± 15% accuracy indicated by previous Hanford calibration experiments.

3) The gamma doses observed were about 5% of the neutron dose for these bare metal criticality exposures.

Two groups of the new diodes from Battelle Memorial Institute were exposed to two radiation bursts at the Sandia Pulsed Reactor facility. The first burst totaled 1.9×10^{16} fissions, the second 2.4×10^{15} fissions. The calculated neutron doses ranged from 271 rads to 2544 rads for the high-level burst and from 34 rads to 315 rads for the low-level burst. The diodes were read four days after the burst tests. Based on a preliminary measurement of the sensitivity of one diode made with the positive ion accelerator, the measured dose was determined. The results obtained for the high-level burst are shown in Table I.

TABLE I

DIODE DOSE MEASUREMENT
READING CURRENT - 25 ma

Dose Expected (rads)	Diode Dose Measured (rads)	Ratio
2544	1865	1.36
2544	1926	1.32
1257	1360	0.92
1257	1317	0.95
525	611	0.86
525	611	0.86
271	279	0.97
271	265	1.02

Only the less sensitive 0.04" base thickness diodes were used in the high-level burst test. This permitted a better determination of the dose response of the diodes. The ratio of the expected-to-measured dose is essentially constant except for the highest dose. The response is approximately linear up to a certain point. These results are preliminary. More measurements of the sensitivity of the individual diodes will be made before accurate results are quoted. The fact that measured doses are low indicates possible problems with dose rate and annealing effects. The results of the low-level burst are not yet analyzed. The more sensitive 0.070" base thickness diodes were used for the low-level burst.

C. RELATIONS

No suggestions were submitted by personnel of the Radiation Protection Operation during the month. One suggestion was re-opened and one was adopted. Two suggestions are pending evaluation.

Safety meetings were held throughout the Section during the month. Topics included job hazard breakdowns and the health and safety manual.

Radiation protection orientation lectures were presented to personnel of Metallurgy Development and Chemical Effluents Technology. Nine talks on plant emergency procedures and civil defense from fallout were presented at plant information meetings. R. F. Foster presented a Hanford Biology Seminar on January 22 entitled "Radioactive Wastes at Hanford and Its Impact in the Environs".

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D. SIGNIFICANT REPORTS

HW-74307-12 - "Radiological Status of the Hanford Environs for December 1962" by R. F. Foster.

HW-76428 - - "Monthly Report for January 1963 - Radiation Monitoring Operation" by A. J. Stevens.

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PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

	January	1963
External Exposure Above Permissible Limits		
Whole Body Penetrating	0	0
Whole Body Skin	0	0
Extremity	0	0
Hanford Pocket Dosimeters		
Dosimeters Processed	6,335	6,335
Lost Results	0	0
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	9,510	9,510
Results - 100-300 mrads	105	105
- 300-500 mrads	15	15
- Over 500 mrads	4	4
Lost Results	24	24
Average Dose Per Film Packet - mrad (ow)	3.65	3.65
- mr (s)	31.11	31.11
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	1,494	1,494
Results - 50-100 mrem	1	1
- 100-300 mrem	0	0
- Over 300 mrem	0	0
Lost Results	6	6
Fast Neutron		
Film Read	362	362
Results - 50-100 mrem	14	14
- 100-300 mrem	173	173
- Over 300 mrem	2	2
Lost Results	0	0
Hand Checks		
Checks Taken - Alpha	38,290	38,290
- Beta-Gamma	60,751	60,751
Skin Contamination		
Plutonium	35	35
Fission Products	42	42
Uranium	0	0
Tritium	0	0
Thorium	0	0

1235697

Whole Body Counter

Subject	Number of Examinations 747-A WBC	1963	Mobile WBC	1963
GE Employees				
Regular	65	65		
Incident Cases	9	9		
Terminations	6	6		
New Hires	0	0		
Special Studies	11	11		
Non-Employees				
Children	1	1		
Visitors	1	1		
Environmental Studies	4	4		
	97	97		

Bioassay

Analysis	Current Reporting Limit	Results Above Reporting Limit January	1963	Samples Assayed January	1963
Plutonium	2.2×10^{-8} μc/sample	95	95	595	595
Fission Product	3.1×10^{-5} μc/sample	9	9	638	638
Strontium	3.1×10^{-5} μc/sample	0	0	0	0
Tritium	5.0 μc/l	172	172	239	239
Uranium	0.14 μgm/l	0	0	131	131
Special Studies		0	0	47	47

Calibrations

	Number of Units Calibrated January	1963
Portable Instruments		
CP Meter	1,083	1,083
Juno	281	281
GM	582	582
Other	219	219
Audits	107	107
	2,272	2,272
Personnel Meters		
Badge Film	1,028	1,028
Pencils	105	105
Other	354	354
	1,487	1,487

1235698

	Number of Units Calibrated January	1963
Miscellaneous Special Services	626	626
Total Number of Calibrations	4,385	4,385

AR Keene
Manager
RADIATION PROTECTION

AR Keene:AJS:ljw

1235699

FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

FY 1963 allocations to Hanford Laboratories from the Chemical Processing Department for sponsored programs were increased during the month as follows:

	Amount of Increase	New Annual Authorization
02 Program Research & Development	$27 000	$1 052 000
03 Program Research & Development	25 000	369 000
02 Program Process Technology	31 000	431 000
03 Program Process Technology	50 000	150 000

During the month Hanford Laboratories allocated $22,500 to the N-Reactor Department to perform additional research and development work on the 04 Program, Advance Studies; the revised FY 1963 authorization for this work is $40,500. Hanford Laboratories also authorized the Chemical Processing Department $45,000 for research and development work during FY 1963 on the 08 Program, Fission Products Production Study.

Budget assumptions for FY 1965 and related cost and manpower projections through FY 1968 on Research and Development Programs sponsored by the AEC Divisions of Reactor Development, Physical Research, Biology and Medicine, and Isotopic Development were prepared and submitted to RLOO-AEC early in January.

An authorization amounting to $45,000 was relayed by RLOO-AEC from the U.S. Air Force Cambridge Research Laboratories to fund continuation of Atmospheric Physics Research Studies during the period April 1, 1963 through March 31, 1964.

In cooperation with the Radiation Protection Operation, a report on film badge processing costs for FY 1962, as requested by RLOO-AEC, was prepared for transmittal to Washington-AEC.

Arrangements were completed in January for billing costs of recovering plutonium scrap generated in the Project Whitney program sponsored by UCLRL. CPD will classify scrap material received from Hanford Laboratories, and using predetermined standard recovery rates, will bill UCLRL for both fund and non-fund costs. Recovery costs applicable to plutonium scrap transferred to CPD during the six month period ending December 31, 1962 are:

	Conversion (Fund)	Depreciation (Non-Fund)
SANL 720 Research & Development	$ 6 015	$1 305
SANL 727 Fabrication	13 475	2 987
Total	$19 490	$4 292

A system of accumulating the costs of the Waste Calcination Demonstration Prototype has been developed. The prototype is currently estimated to cost $2.2 million and completion is scheduled for early in FY 1965.

General Accounting

Letters under Agreement AT-6 were processed as shown below:

Date	Subject	Date Accepted
1-7-63	Request for Air Sampling Equipment Supplies	1-10-63
1-21-63	Training of Euratom Physicist at Hanford	1-22-63

Comparison of travel activity with previous years is given below:

Number of Trips Started - FY to Date	
FY 1960	700
FY 1961	727
FY 1962	626
FY 1963	724

OPGs issued during January included:

OPG No.	Title
22.1.1	Hanford Laboratories (organization)
22.1.2	Chemical Laboratory (organization)
22.1.3	Reactor and Fuels Laboratory (organization)
22.1.4	Physics and Instrument Laboratory (organization)
1.13	Participation in Hazardous Business
9.2	Procurement of Equipment, etc. (pages 1 & 2)
55.1.4	Payment to Nonexempt Employees for Time Spent on Off-Site Travel

OPG 7.3, Sensitive Positions, was cancelled.

Hanford Laboratories' completed Plant and Equipment Research Facilities at January 1, 1963, totaled $70,797,306. This represents 86.3% of total HAPO Research Facilities' investment of $82,053,149. Since September 1956, Hanford Laboratories' Plant and Equipment investment has increased $39.5 million. Investment by Area is as follows:

Area	Investment
100-B	$ 91 476
100-D	2 144 875
100-F	3 511 747
100-H	23 505
100-K	849 666
200-E	1 319 510
200-W	5 203 097
300	54 360 310 -1)
600	993 537
700	450 014
1100	2 510
White Bluffs	604 035
Off-site	1 243 024
	$70 797 306

(1- Includes such major facilities in excess of one million as:

306 Bldg.	$ 5 354 945
308 Bldg.	4 904 355
309 Bldg.	14 358 192
314 Bldg.	1 184 719
321 Bldg.	2 235 340
325 Bldg.	5 911 820
326 Bldg.	4 147 606
327 Bldg.	3 114 610
328 Bldg.	1 677 593
329 Bldg.	2 160 442

During the month, $115,471 were transferred to classified plant accounts from Work In Progress accounts.

The number of manually prepared input data sheets covering additions and the revisions to Hanford Laboratories records in the EDP system totaled 10,874 for the first six months of FY 1963 compared to 15,039 for 1961 and 10,949 for 1962. Further mechanization of Hanford Laboratories records should increase this number considerably. A study is currently being made to determine the feasibility of using a Flexowriter in this work and its effects HAPO-wide.

Hanford Laboratories' material investment at January 1, 1963 totaled $28.1 million, distributed as follows:

	(In thousands)
SS Material	$26 309
Reactor and Other Special Materials	1 368
Spare Parts	388 -1)
Total	$28 065

(1- Includes a reserve of $79,592 established at January 1, 1963.

The value of nuclear materials consumed in research this fiscal year is $3.0 million of which Hanford Laboratories' portion was:

	(In thousands)
2000 Program	$ 932
3000 Program	605
4000 Program	1 366
Total	$2 903

The inventory of heavy water at January 31, 1963 indicated a net loss for January of 30,241 pounds valued at $44,860. Scrap generated during the month amounted to 3,408 pounds and a charge to cost estimated to be $2,897. Costs of $47,757 accrued in January were based on December's average unit prices. Adjustments will be made during February when actual prices of scrap material are determined.

As required by HAPO OPG 8.6, a Semi-Annual Major Machine Tool Summary Report was completed for Hanford Laboratories and forwarded to Property Management, C&AO. This report covered tool utilization and maintenance costs for the period July 1 through December 31, 1962.

A total of 15,366 grams of contaminated platinum materials and scrap valued at $35,649 and having no further value at HAPO was made available to the AEC for shipment to the NYOO for recovery.

Notification was received from Contract and Accounting Operation that spare parts made from zirconium should be carried in the Reactor and Other Special Materials Inventory account and not in the Spare Parts Inventory account. This is in compliance with a Washington-AEC directive. The transfer of $143,731 worth of zirconium spare tubes for the PRTR to the Hanford Laboratories Reactor and Other Special Materials account is involved. A 25% reserve established for the spare parts zirconium tubes, as spare parts, will remain in the reserve account until June 1963, when it will be factored into the reserve adjustment required at year end.

Hanford Laboratories, as a service to Contract and Accounting Operation, agreed to record the C&AO Spare Parts Inventory for Analex Printer spares which will be in the $3,000 - $5,000 range. The 25% reserve and any future inventory adjustments will be charged to C&AO.

A report of results for the quarterly inventory of Other Special Materials in custody of 95 HL material holders as of December 31, 1962, revealed a net shortage of 5 grams consisting of platinum and silver. These minor differences, attributed to rounding, will not be adjusted unless they are again substantiated by the March 31, 1963 quarterly inventory.

Laboratory Storage Pool activity for the Month of January 1963 is summarized below:

Equipment	Current Month Qty	Current Month Value	FY to Date Qty	FY to Date Value
Beginning Balance	1 354	$785 979	1 081	$562 200
Items Received	211	25 480	836	470 098
Items Reclaimed by Custodians	(6)	(37 260)	(80)	(98 834)
Equipment Transfers	(43)	(8 035)	(131)	(46 966)
Items Disposed of by PDR	(82)	(4 778)	(116)	(15 809)
Items Disposed of by Excess	(146)	(4 187)	(302)	(72 225)
Price Adjustment				(41 265)
Total	1 288	$757 199-1)	1 288	$757 199

(1- Includes 148 items valued at $92,840 on loan at January 31, 1963.

During the month, 50 items valued at $12,585 were loaned and/or transferred in lieu of purchases. A total of 203 items valued at $108,360 has been redirected to useful purposes this fiscal year in lieu of purchases.

The following is a summary of materials activity in the Laboratory Storage Pool during January and fiscal year 1963 to date:

Type of Material	Balance 6-30-62	Current Month Receipts	Current Month Disburs.	Fiscal Year Receipts	Fiscal Year Disburs.	Balance 1-31-63
Reactor & Other Special						
Beryllium	$ 592	$	$	$ 39	$ 203	$ 428
Hafnium	1 499				1 499	--
Gold	2 924		1	270	11	3 183
Silver	463		22	15	23	455
Platinum	5 601		1 311	18 964	6 399	18 166
Clean Scrap	636	162		2 280	648	2 268
Contam. Scrap	19 529	1 001		20 115	4 022	35 622
Palladium	2 535		244	200	326	2 409
Zirconium	122 320		286	21 426	45 134	98 612
Sub-total	156 099	1 163	1 864	63 309	58 265	161 143

	Balance 6-30-62	Current Month Receipts	Current Month Disburs.	Fiscal Year Receipts	Fiscal Year Disburs.	Balance 1-31-63
Other (Memo)						
Hastelloy	82 231	5 871	86	5 871	12 385	75 717
UO_2	111 622				88 996	22 626
Aluminum			2	15 288	2	15 286
Zirconium-R&D	12 791			1 769	12 900	1 660
Zirconium-Scrap	17 226		39	4 025		5 156
Graphite		16 719		27 719	16 095	27 719
Steel		1 708		1 708		1 708
Miscellaneous	12 696	15	15	2 192	5 458	9 430
Sub-total	236 566	24 313	142	58 572	135 836	159 302
Total Material	$392 665	$25 476	$2 006	$121 881	$194 101	$320 445

The first shipment of steel (5,216 lbs.) for the Irradiation Damage to Reactor Metals program was received this month. New pallet storage racks were erected in the 3718-B Building to provide bin storage for the steel. A record card and individual piece number system was established which can be easily converted to a mechanized system when C&AO establishes a program for material mechanization.

A total of 37,998 pounds of graphite valued at $16,719 was received at the Laboratory Storage Pool during the month for use by the Materials Research and Service Operation. The control of this material by type of graphite, number of bars and by weight, can also be easily adapted to mechanization.

Project unitization reports were issued during January on the following projects:

AEC-167	PRTR	$14 315 479
CAH-927	Addition to 271-CR Bldg. - Waste Treatment Demonstration Facility	87 406

Action on projects during the month is indicated below:

New Money Authorized HL

CAH-916	Fuels Recycle Pilot Plant	$ 10 000
CAH-922	Burst Test Facility	108 900

Physical Completion Notice Issued

CGH-951 A-C Column Facility

Construction Completion and Cost Closing Statement Issued

CGH-785 In-Reactor Studies Equipment - 100-K

A complete revision of the Authorization and Performance of Work Manual was published early in January to reflect HAPO and RLOO organizational changes and OPG-induced procedural changes.

Contracts processed were:

DDR-156 Supplement 2	Armour Research Foundation
MRO- 55	Fisher Scientific Company
SA-262	Betz Laboratories, Inc.
SA-237 Supplement 1	Kadlec Methodist Hospital
CA-369	R. T. De Hoff
CA-370	R. Wells Moulton
CA-371	Washington State University
CA-373	Clyde E. McNeilly

Personnel Accounting

The following employees received invention report awards of $125 during the month:

Name	HWIR No.	Title
D. W. Shannon	1302	A Quantitative Test for Adhesion of Coatings to a Substrate
R. L. Moore	1448)	
R. T. Allemann	1448)	An Electrostatic Bubble Scrubber
U. L. Upson	1448)	

Number of Hanford Laboratories Employees

Changes During Month	Total	Exempt	Nonexempt
Employees on payroll at beginning of month	1 637	702	935
Additions and transfers in	21	7	14
Removals and transfers out	21	14	7
Employees on payroll at end of month	1 637	695	942

Overtime Payments During Month	January	December
Exempt	$ 3 678	$ 3 863
Nonexempt	21 607	21 843
Total	$ 25 285	$ 25 706

Gross Payroll Paid During Month		
Exempt	$ 658 236	$ 660 459
Nonexempt	512 525	634 916
Total	$1 170 761	$1 295 375

Participation in Employee Benefit Plans at Month End	January Number	January Percent	December Number	December Percent
Pension	1 471	99.4	1 475	99.4
Insurance Plan - Personal	399		401	
- Dependent	1 236	99.8	1 235	99.8
U. S. Savings Bonds				
Stock Bonus Plan	158	43.4	158	43.6
Savings Plan	69	4.2	70	4.3
Savings and Security Plan	1 131	88.4	1 132	88.4
Good Neighbor Fund	1 184	72.0	1 177	71.6

Insurance Claims	January Number	January Amount	December Number	December Amount
Employee Benefits				
Life Insurance	0	$ 0	0	$ 0
Weekly Sickness and Accident	8	633	12	938
Comprehensive Medical	93	4 983	60	4 311
Dependent Benefits				
Comprehensive Medical	159	14 621	109	9 312
Total	260	$20 237	181	$14 561

TECHNICAL ADMINISTRATION

Employee Relations

Twelve non-exempt employment requisitions were filled during January; twenty-four remain to be filled.

Professional Placement

Advanced Degree - Fifty-seven Ph.D. applicants were invited to visit HAPO for employment interviews. Five visited; two offers were extended; two acceptances and one rejection were received. Three offers are currently open.

BS/MS - Nineteen Program offers and eleven direct placement offers were extended. Offers accepted: six Program and five direct placement. Offers rejected: ten Program and one direct placement. Current open offers: seventy-one Program and eight direct placement.

Technical Graduate Program - Five technical graduates were placed on permanent assignment. Three new members were added to the roll. Current Program strength is fifty.

1235707

Technical Information

Hanford Laboratories' contribution to the AEC's annual research report covering 04, 05, and 06 research programs was compiled.

The annual inventory of secret research and development reports and scientific and technical atomic weapon data reports was begun this month and is 90 percent complete.

FACILITIES ENGINEERING

At month's end Facilities Engineering Operation was responsible for seven active projects having total authorized funds in the amount of $1,229,500. The total estimated cost of these projects is $7,524,000. Expenditures on them through December 31, 1962 were $613,000.

The following summarizes the project activity in January:

Number of authorized projects at month's end -------------------- 7

Number of new projects authorized ------------------------------- 1
CAH-982, Addition to the Radionuclide Facilities

Projects completed -- 0

New projects submitted to the AEC ------------------------------- 1
CGH-992, Additional Fuel Loading Equipment - 308 Building

New projects awaiting AEC authorization ------------------------- 4
CGH-974, Analog Simulation Facility
CAH-985, Addition to the 222-U Building
CAH-986, 300 Area Retention Waste Expansion System
CGH-992, Additional Fuel Loading Equipment - 308 Building

Project proposals complete or nearing completion ---------------- 7
Neutron Calibration Facility - 3745-A Building
Irradiated Structural Materials Testing Facility
Atmospheric Physics Building
PRTR Storage Basin and Experimental Facilities Modifications
Heat Transfer Apparatus for Model Studies
Ventilation Improvements - 309 Building
High Temperature Lattice Testing Reactor

Pages appended to this report provide detailed project status information.

1235708

Services

During the month engineering service was provided on the following jobs:

108-F Revision of power feed for counting room
Preparation of proposal for intercom system
327 Bid recommendation on intercom system components
329 DC Power requirements from Laboratory 11B
231-Z Provide design information and prepare purchase specification for intercom system addition
309 Completion of pre-installation review and approval of plans for evacuation alarm modifications
105-C Preparation of equipment specifications for Irradiation Studies Loop

Pressure system problems on which assistance was given included:

PRTR Rupture Loop pump vibration
PRTR Gas Loop heater safeguards
FRPP Waste Calcination equipment designs
Irradiation Studies Loop safeguards
314 Building EDEL II Loop safety audit

Plant engineering effort was expended on:

3702 Lighting modifications
Planning for continuous waste monitoring instrumentation for Buildings 308, 325, 326, 327 and 329
Lighting and receptacle layout for room 14A, 326 Building
306 Building criticality alarm modifications
Electrical layout for proposed manipulator maintenance shop, 327 Building
Emergency power load check in Buildings 325, 326, 327 and 329
325 electrical load increase planning
325 ventilation heat automatic shut-off
325 mezzanine ventilation plans
3702 Building renovations
Design of machine shop exhaust system, 327 Building
326 monorail relocation design
209-E ventilation modification design
700 Area heating system study
328 ventilation temperature control
325-A cell filter installation design
M & E lists for Projects CGH-857, 951, and Waste Calcination Disposal Prototype were issued
Requisitions for equipment estimated to cost $1000 were issued

1235709

Facilities Operation

Landlord costs for December were $149,217 which were about 80% of the forecast for the month. The total cost to date is $842,525, which is 84% of the predicted. Improvement maintenance was $4,255. All maintenance was about $20,000 below the anticipated. Steam with its adjustments exceeded the forecast, but the steam for the year was only 81% of predicted. Preliminary cost figures for January show steam at about 88%, and the total landlord budget was 88% of forecast.

Waste disposal operations during December, compared to the previous month, are summarized below:

	November	December
Concrete barrels disposed	16	12
Loadluggers of dry waste	40	27
Crib waste (gallons)	305,000	325,000

Building Operations

The filter plant operated satisfactorily. On January 11, however, the influent orifice lines froze and heating arrangements were improvised. Later the lines were relocated and a better cover was installed on the pit.

Both new laboratory vacuum pumps are installed in 325 Building. One is in service, and the other will be ready by February 2, 1963.

Drafting

The equivalent of 135 drawings was produced during the month for an average of 28 man-hours per drawing.

Major jobs in progress are - PRTR "As-Builts", Shim Rod Control, PRTR Cladding Cutter Assembly, Hi-temp Vacuum Furnaces, Process Tube and Fuel Handling Carriage, Inhalation Studies Hood, Fuel Element Identification System, Salt Cycle for "C" Cell, High Temperature Test Apparatus, Capillary Loading Glove Box, Hi-Pressure Furnace.

Work performed by CE&UC drafting and Bovay engineers during the month was 168 and 173 man-hours, respectively. Work assigned was 168 and 50 man-hours.

Construction Supervision

Activity during the month on construction work (J. A. Jones Company) being performed for Hanford Laboratories components is given below:

	Unexpended Balance	Waste Calcination
Orders outstanding beginning of month	$172 933	$
Issued during the month (inc. suppl. and adj.)	414 021	350 000
J. A. Jones expenditures during month (incl. C.O. costs)	132 630	5 250
Balance at month's end	454 324	
Orders closed during month	160 200	

In addition, work on twelve maintenance work orders having a total face value of $28,206 issued to Plant forces was supervised.

Construction and maintenance activities completed during January included:

108-F Gas bottle storage facility construction
108-F Gas piping replacement
141-H Muffle furnace installation
189-D Equipment and piping installation
321 Tank farm sprinkler servicing
327 Decontamination cell framework installation
3718-B Rack and pallet installation

W Sale

Manager
Finance and Administration

W Sale:whm

1235711

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- [illegible]

DATE January 31, 1963

PROJ. NO.	TITLE	FUNDING
CAH-822	Pressurized Gas Cooled Facility	4141 Operating

AUTHORIZED FUNDS $ 1,170,000	DESIGN $ 43,000 / CONST. $ 1,127,000	AEC $ 15,000 / GE $ 1,155,000
COST & COMM. TO 1-13-63	$ 1,147,527 (GE)	
ESTIMATED TOTAL COST	$ 1,170,000	
STARTING DATES	DESIGN 12-15-58	CONST. 10-17-60
DATE AUTHORIZED 12-26-58*	DIR. COMP. DATE 6-30-63	
EST'D. COMPL. DATES	DESIGN 4-29-60	CONST. 6-30-63
ENGINEER	FRAC-MEEO - DP Shively	

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100	94	94
PF	1.4	0	0
CPFF	23	99	99
FP	6.5	100	100
Gov. Eq.	69.1	93	93

SCOPE, PURPOSE, STATUS & PROGRESS

Insulation of loop piping is approximately 99% complete.

Blower No. 3 was run at 400 psig and 10,000 rpm but limited to approximately 500F inlet temperature. New starter is being installed with testing schedule to be resumed February 1, 1963.

* Initial authorization was December 12, 1958.

PROJ. NO.	TITLE	FUNDING
CAH-916	Fuels Recycle Pilot Plant	4-62-a-3

AUTHORIZED FUNDS $ 500,000	DESIGN $ 500,000 / CONST. $ --	AEC $ -- / GE $ 500,000
COST & COMM. TO 1-20-63	$ 499,900	
ESTIMATED TOTAL COST	$ 5,450,000***	
STARTING DATES	DESIGN 3-15-61	CONST. 8-15-63*
DATE AUTHORIZED 10-19-61**	DIR. COMP. DATE --	
EST'D. COMPL. DATES	DESIGN 11-15-62	CONST. 2-1-65
ENGINEER	FEO-RW Dascenzo	

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		2300
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I	11	100	100
GE-TIT. II	89	100	100
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project is to provide a facility to perform a full scope of engineering tests and pilot plant studies associated with fuel reprocessing concepts.

All drawings have been approved for construction. The specifications are 100% complete. The bid package was completed November 30, 1962.

The AEC has reviewed the specifications and the comments have been incorporated.

To date the revision of the project proposal requesting total project funds has not been approved by the Commission.

* Estimated construction starting date for removal of burial ground fill.
** Original authorization for design was February 9, 1961.
*** Including transferred capital property valued at $100,000.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 76315

DATE January 31, 1963

PROJ. NO.	TITLE	FUNDING
CAH-922	Burst Test Facility for Irradiated Zirconium Tubes	62-k

AUTHORIZED FUNDS			
$ 289,000	DESIGN $ 32,000	AEC $ 141,500	COST & COMM. TO 1-20-63 $ 30,006 (GE)
	CONST. $ 257,000	GE $ 138,500	ESTIMATED TOTAL COST $ 289,000**

STARTING DATES			EST'D. COMPL. DATES
DESIGN 11-7-61	DATE AUTHORIZED 12-26-62*		DESIGN 5-31-62
CONST. 3-15-63	DIR. COMP. DATE 1-31-64		CONST. 1-31-64

ENGINEER

FEO - DL Ballard

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER - Bovay Engineers	1	270
DESIGN ENGINEERING OPERATION	1	285
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	57	100	100
AE-TIT. II	43	100	100
CONST.	100	0	0
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide facilities to permit deliberate destructive testing of irradiated zirconium tubing. It will provide operating and tube life data not available because of the limited operating history of Zircaloy-2 pressure tubing in reactors.

The AEC Work Authority allocating funds to the General Electric Company in the amount of $138,500 was received on January 15, 1963. The design portion of this amount has been allocated to CE&UO.

Bid package information for Fixed Priced Construction and equipment specifications for purchased items are being prepared.

* Original authorization for design was October 23, 1961.

** Includes transferred capital property valued at $9,000.

PROJ. NO.	TITLE	FUNDING
CAH-936	Coolant Systems Development Laboratory - 1706-KE Building Addition	62-k

AUTHORIZED FUNDS			
$ 130,000	DESIGN $ 9,000	AEC $115,000	COST & COMM. TO 1-20-63 $ 15,407 (GE)
	CONST. $ 121,000	GE $ 16,000	ESTIMATED TOTAL COST $ 130,000

STARTING DATES			EST'D. COMPL. DATES
DESIGN 9-8-61	DATE AUTHORIZED 4-5-62*		DESIGN 1-1-62
CONST. 5-1-62	DIR. COMP. DATE 3-1-63		CONST. 3-15-63

ENGINEER

FEO - DL Ballard

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE	3	790
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		166
GE FIELD ENGINEERING		47

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	100	100
TITLE I			
GE-TIT. II	100	100	100
AE-TIT. II			
CONST.	100	100	80
PF			
CPFF			
FP		100	80

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides facilities for conduct of corrosion and decontamination studies for nuclear reactor coolant systems, by the addition of a laboratory facility on the west side of the 1706-KE Building.

Installation of heating, ventilating and exhaust equipment underway.

The contractor has indicated he now expects to receive the laboratory equipment from Metalab during early part of February.

* Original authorization for design was August 9, 1961.

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 76315

DATE January 31, 1963

PROJ. NO.	TITLE	FUNDING
CAH-958	Plutonium Fuels Testing & Evaluation Laboratories-308 Bldg.	62-k

AUTHORIZED FUNDS $ 150,000 — DESIGN $ 15,500; CONST. $ 134,500 — AEC $ 134,500; GE $ 15,500 — COST & COMM. TO 1-20-63 $ 15,500 (GE) — ESTIMATED TOTAL COST $ 150,000

STARTING DATES: DESIGN 11-16-62; CONST 3-1-63 — DATE AUTHORIZED 6-22-62 — DIR. COMP. DATE 5-15-63 — EST'D. COMPL. DATES: DESIGN 2-28-63; CONST. 7-30-63

ENGINEER: FEO - OM Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100	60	70
TITLE I			
GE-TIT. II	100	60	70
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides for the extension of plutonium research laboratories on the second floor of the 308 Building by erection of plastered ceilings and walls to provide contamination control barriers. It also includes laboratory service extension and fabrication of a metallography hood.

Metallography hood comment prints have been issued.

PROJ. NO.	TITLE	FUNDING
CAH-962	Low Level Radiochemistry Building	05-1-63-4-001-23

AUTHORIZED FUNDS $ 113,000 — DESIGN $ 113,000; CONST. $ - - — AEC $ 82,000; GE $ 31,000 — COST & COMM. TO 1-20-63 $ 16,500 (GE) — ESTIMATED TOTAL COST $ 1,200,000

STARTING DATES: DESIGN 7-23-62; CONST. 8-1-63 — DATE AUTHORIZED 6-28-62 — DIR. COMP. DATE - - — EST'D. COMPL. DATES: DESIGN 7-1-63; CONST. 9-1-64

ENGINEER: FEO - DL Ballard

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		135
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I		100	100
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides a building in which extremely sensitive radioanalysis and methods development can be performed in an atmosphere protected from the environs. It consists of designing and constructing a building housing approximately 22,000 square feet of floor area including the basement.

The design criteria was transmitted to the Commission on December 31, 1962.

The selection of an Architect-Engineer is to be made by the Commission.

1235714

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 76315

DATE January 31, 1963

PROJ. NO.	TITLE	FUNDING
CGH-974	Analog Simulation Facility	62-a-1

AUTHORIZED FUNDS $	DESIGN $ / CONST. $	AEC $ / GE $
COST & COMM. TO	$	
ESTIMATED TOTAL COST	$ 1,600,000*	
STARTING DATES	DESIGN / CONST.	
DATE AUTHORIZED		
DIR. COMP. DATE		
EST'D. COMPL. DATES	DESIGN / CONST.	
ENGINEER		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide an appropriately sized and consolidated analog computer simulation facility for the Hanford complex. Initial application will be associated with the startup program for the NPR.

The preliminary project proposal which was submitted to the Washington AEC has been returned to AEC-RLOO.

* Approximate estimate.

PROJ. NO.	TITLE	FUNDING
CAH-977	Facilities for Radioactive Inhalation Studies	62-k

AUTHORIZED FUNDS $ 13,500	DESIGN $ 13,500 / CONST. $ - -	AEC $ 10,500 / GE $ 3,000
COST & COMM. TO 1-20-63	$ 3,000 (GE)	
ESTIMATED TOTAL COST	$ 140,000	
STARTING DATES	DESIGN 2-1-63 / CONST. 7-15-63	
DATE AUTHORIZED	9-24-62	
DIR. COMP. DATE	- -	
EST'D. COMPL. DATES	DESIGN 6-1-63 / CONST. 7-1-64	
ENGINEER	FEO - J. T. Lloyd	

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			98 (GE
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide additional facilities essential to the conduct of biology research programs involving the effects of inhaled radioactive particles. It will comprise an addition to the 144-F Building, consisting of approximately 2,000 square feet of indoor dog pens and supporting facilities and approximately 2,200 square feet of outside dog pens.

The General Electric approved design criteria masters and scope tracing were presented to AEC-RLOO for approval on December 21, 1962.

To date the Commission has not selected an Architect-Engineer.

1235115

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 76315

DATE January 31, 1963

PROJ. NO.	TITLE	FUNDING
CAH-982	Addition to the Radionuclide Facilities - 141-C Building	63-1

AUTHORIZED FUNDS $ 14,000	DESIGN $ 14,000	AEC $	COST & COMM TO 1-20-63 $ - 0 -
	CONST. $ ---	GE $	ESTIMATED TOTAL COST $ 155,000

STARTING DATES	DESIGN 3-15-63	DATE AUTHORIZED 1-22-63	EST'D. COMPL. DATES: DESIGN 8-1-63
	CONST.	DIR. COMP. DATE	CONST.

ENGINEER: FEO - JT Lloyd

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide an addition to the 141-C Building in 100-F Area to supplement the present radionuclide study facilities. The building addition will comprise approximately 2,500 square feet of floor area for laboratory facilities and controlled feeding pens for swine.

The project proposal requesting $14,000 was authorized by Directive No. AEC-215, dated January 22, 1963. To date the AEC Work Authority, authorizing funds to General Electric Company, has not been issued.

PROJ. NO.	TITLE	FUNDING
CAH-985	Addition to the 222-U Building	63-1

AUTHORIZED FUNDS $	DESIGN $	AEC $	COST & COMM. TO $
	CONST. $	GE $	ESTIMATED TOTAL COST $ 150,000*

STARTING DATES	DESIGN 3-16-63**	DATE AUTHORIZED	EST'D. COMPL. DATES: DESIGN 7-15-63**
	CONST. 9-15-63**	DIR. COMP. DATE	CONST. 5-15-64**

ENGINEER: FEO - D. S. Jackson

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project provides an addition to the 222-U Building in which to perform;
1) geologic and hydrologic studies related to waste disposal practices, and
2) studies on release of fission products from reactor fuels heated to high temperatures.

The project proposal requesting design funds in the amount of $17,000 was submitted to the RLOO-AEC October 8, 1962.

* Approximately
**Based on AEC authorization by March 1, 1963.

1235716

SEMI-MONTHLY PROJECT STATUS REPORT

GENERAL ELECTRIC CO. — Hanford Laboratories

HW- 76325

DATE January 31, 1963

PROJ. NO.	TITLE	FUNDING
CAH-986	300 Area Retention Waste System Expansion	63-1

AUTHORIZED FUNDS	DESIGN $	AEC $	COST & COMM. TO	$
$	CONST. $	GE $	ESTIMATED TOTAL COST	$ 100,000*

STARTING DATES	DESIGN 3-15-63**	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN 7-15-63**
	CONST.	DIR. COMP. DATE		CONST.

ENGINEER

FEO - O. M. Lyso

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT-ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

This project will provide two additional 50,000 gallon retention basins, automation of the basin influent valving and semi-automation of the effluent valving. It will provide required storage basin capacity and obtains maximum use of existing basins.

The project proposal requesting design funds in the amount of $14,000 was submitted to AEC-RLOO October 8, 1962.

* Preliminary estimate.
** Based on AEC approval by January 31, 1963.

PROJ. NO.	TITLE	FUNDING

AUTHORIZED FUNDS	DESIGN $	AEC $	COST & COMM. TO	$
$	CONST. $	GE $	ESTIMATED TOTAL COST	$

STARTING DATES	DESIGN	DATE AUTHORIZED	EST'D. COMPL. DATES	DESIGN
	CONST.	DIR. COMP. DATE		CONST.

ENGINEER

MANPOWER	AVERAGE	ACCUM MANDAYS
FIXED PRICE		
COST PLUS FIXED FEE		
PLANT FORCES		
ARCHITECT - ENGINEER		
DESIGN ENGINEERING OPERATION		
GE FIELD ENGINEERING		

PERCENT COMPLETE	WT'D.	SCHED.	ACTUAL
DESIGN	100		
TITLE I			
GE-TIT. II			
AE-TIT. II			
CONST.	100		
PF			
CPFF			
FP			

SCOPE, PURPOSE, STATUS & PROGRESS

1235717

04 PROGRAM - REACTOR DEVELOPMENT

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output for January was 617 MWD for an experimental time efficiency of 63% and a plant efficiency of 44%.

There were six operating periods during the month, one of which lasted through the month-end. Each of the other five were terminated manually due to operational difficulties as follows: High disolved O_2 content in the primary system, once; excessive D_2O recovery system collection (cap gasket failures), twice; suspected fuel element defects, twice.

The core loading at startup on January 3, was composed of 26 UO_2 elements, 29 Pu-Al elements and 30 UO_2-PuO_2 elements. At month-end the loading consisted of 24 UO_2, 24 Pu-Al, and 37 UO_2-PuO_2 elements. Fuel exposure history at month-end was:

Maximum UO_2 exposure/element	2560 MWD/TU
Average UO_2 exposure/element	1651 MWD/TU
Maximum Pu-Al exposure/element	73.0 MWD
Average Pu-Al exposure/element	52.3 MWD
Maximum Moxtyl exposure/element	27.9 MWD (~558 MWD/TU)
Average Moxtyl exposure/element	12.8 MWD (~257 MWD/TU)

The status of the various test elements on January 31, 1963, is shown in the table below. Those test elements which had reached their assigned goal exposure or had been permanently discharged for other reasons prior to January 1, 1963, have been deleted from this table.

PRTR Test No.	Channel Location	Fuel Element Number	Description	Date Initial Charge	Date Discharged	Accumulated MWD
10	1447	1082	UO_2-Hot Swage	11/3/61	---	63.9
10	1647	1067	UO_2-Vipac	11/3/61	---	67.9
13	1853	5094	Pu-Al Physics	12/3/61	---	70.9
14	1253	5098	Moxtyl-Vipac	5/8/62	--- repad	21.4
14	1544	5099	Moxtyl-Vipac	5/8/62	--- repad	18.4
37	1449	1096	UO_2-Physics	5/12/62	--	29.1
37	1649	1097	UO_2-Physics	5/12/62	--	29.3
37	1552	1098	UO_2-Physics	5/12/62	--	26.8
37	1548	1099	UO_2-Physics	5/12/62	--	28.5

1235718

PRTR Test No.	Channel Location	Fuel Element Number	Description	Date Initial Charge	Date Discharged	Accumulated MWD
37	1651	1100	UO_2-Physics	5/12/62	--	24.9
37	1451	1101	UO_2-Physics	5/12/62	--	28.8
48	1243	5150	Moxtyl ($\frac{1}{2}$"x$\frac{1}{2}$" pads)	8/1/62	--	13.5
54	1948	5116	Moxtyl (clip-on pads)	5/8/62	--	24.9 (9.4 w/clip)
54	1443	5117	Moxtyl (clip-on pads)	5/8/62	--	27.9 (12.0 w/clip)
54	1459	5118	Moxtyl (clip-on pads)	5/8/62	--	27.9 (12.9 w/clip)

Unusual activity indications on the rupture detection system resulted in two unscheduled reactor outages. In both cases, a total stagnant water test was conducted, and in a third instance, five tubes were checked in this manner. Five elements were removed as "leaker suspects" as a result of these tests. One element was later cleared as a suspect and was recharged. Confirmation tests on all suspects were initiated.

Indicated D_2O and Helium losses for the month were 3,240 pounds and 117,500 scf, respectively. Considerable effort was expended on ventilation balance and measuring techniques to provide more accurate stack loss indications. Stack sample indications showed a D_2O loss of 1765 pounds, which is more representative as the inventory loss is believed influenced by an inventory error in December. System balance work on the D_2O recovery system resulted in "best yet" performance of this system.

Equipment Experience

A total of 81 outage hours were charged to repair work. Major contributors were tube-to-nozzle gasket repair and helium compressor problems. In addition a total of 45 hours of outage time was charged to high O_2 in the helium and D_2O systems. O_2 problems were traced to shaft seal inleakage on the core blanket blower.

The mechanical seal on PP #1 was replaced after approximately 550 hours of running time.

Preventive maintenance required 614.7 manhours or 13% of the total maintenance effort.

1235719

Improvement Work Status (Significant Items)

Work Completed:

Gasometer Pressure Relief
Light Water Injection Manual Control Interlock with Gas Balance Compressor
Monitoring Voltmeters for DC Systems
Installation of Improved Hofer Comp. Oil Pumps and Oil Regulating Valves

Work Partially Completed:

Enlarge Chemical Feed System
Reactor Core Level Indicator
Primary Loop Drain and Flush Valve Modifications
Keithley Power Plug Modification
Interlock Between C-D Machine Shroud Seat and Discharge Hoist
Primary Pump Control Circuits
Shim Rod Connector Modifications

Design Work Completed:

Pressurizer Vent Valve Relocation to Minimize D_2O Loss
Recording Ammeter for Primary Pumps
Compressed Air Supply Modifications
384 Emergency Power to Primary Pumps
Control Room Criticality Alarm

Design Work Partially Completed:

Additional Fuel Storage and Examination
Boiler Feed Pump Seals
Install Vibration Snubbers for Earthquake Protection
Actuator with Hydraulic Snubber for Bottom Blowdown Line
Shim Well Rotameter Modification

Process Engineering and Reactor Physics

Work on the transient reactor poisoning computer program has continued. The contributions from the buildup of Samarium-149 and 151 have been included. Agreement between the calculation and the experimental data from Power Test 16 was quite reasonable.

A study of the functional testing of emergency systems was completed and a summary report prepared.

A study of the functional requirements, methods of operation and consequences of actual and spurious trips of the containment system was started.

1235720

During the month of January, there were 90 fuel element movements, in and out of the process tubes. Since the PRTR was first loaded, there have been over 2,000 such movements, some to the storage basin and back, and some from process tube to process tube. The process tubes now in service have averaged 23 movements with one tube having 43 movements to date. One fuel element, a UO_2, has been moved 21 times.

Procedures

Revised Operating Procedures Issued	4
Revised Operating Standards Issued	6
Temporary Deviations to Operating Standards Issued	13
Revised Process Specifications accepted for use	2
Maintenance Manuals and Procedures Issued	0

Drawing As-Built Status

	January	Total
Approved for as-built	17	884
Ready for approval		9
In drafting		40
Voided		78
		1 011
Scheduled for review		400
		1 411

Personnel Training

Qualification Subjects	212 manhours
Specifications, Standards, Procedures	49
Fueling Vehicle	0
Maintenance Procedures	91
FEEF	6
	358 manhours

Status of Qualified Personnel at Month-end

Qualified Reactor Engineers	7
Provisionally Qualified Reactor Engineers	2
Qualified Technicians	6
Qualified Technologists	16
Provisionally Qualified Technologists	1

Plutonium Recycle Critical Facility

Cell leak rate testing required considerable time. Minimum consistent value at 6 inches w.g. was 1200 ft^3/day with inability to reduce below 1000 ft^3/day. A discrepency between weir level and moderator level of 22-23 inches was detected with the full flow moderator addition pump in service. Piping changes relieved hydraulic resistances correcting this problem.

1235721

Base count measurements using the source were taken in preparation to loading the reactor.

Fuel Element Rupture Test Facility

Project Status (CAH-867)

The design features to permit complete loop testing ex-reactor were completed. Work progressed on the structural steel for shielding, revised high and low pressure relief systems, emergency depressurizing valve; and inlet piping shut-off valve. No Design Test items were completed.

Operation

Continued performance of various operating tests to demonstrate performance characteristics and train personnel. Continued charge-discharge practice in the 314 Building. A total of 88 hours was devoted to training matters.

GAS COOLED POWER REACTOR PROGRAM

Gas Cooled Loop

Project Status (CAH-822)

The project completion date was extended to June 30, 1963, by Directive No. AEC-145, Modification 8. The project is 94% complete. Re-insulation of main loop piping following heater installation is essentially complete. The second and third blower units were tested by Bristol-Siddeley at the minimum conditions. Tests at more severe conditions were planned. Completed work on several punch list items.

Operation

Assisted in Project design tests to gain system familiarity. Eighty-three hours were used in training.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 23,884 hours. This includes 18,511 hours performed in the Technical Shops, 4,754 hours assigned to Minor Construction, 150 hours assigned to off-site vendors, and 469 hours to other project shops. Total shop backlog is 23,181 hours, of which 70% is required in the current month with the remainder distributed over a three-month period. Overtime hours worked during the month was 5.6% (1,195.1) of the total available hours.

1235722

Distribution of time was as follows:

	Man-Hours	% of Total
N-Reactor Department	4,948	20.72
Irradiation Processing Department	4,321	18.10
Chemical Processing Department	662	2.77
Hanford Laboratories	13,953	58.41
Construction Engineering and Utilities	0	0

LABORATORY MAINTENANCE OPERATION

Total productive time realized was 16,900 hours of a possible 17,600 hours potentially available. Of the total productive time realized, 91% was expended in support of Hanford Laboratories components with the remaining 9% directed toward providing service for other HAPO organizations. Overtime worked during the month was 3.3% of total available hours.

Manpower utilization for January is summarized as follows:

A. Shop Work		2 500 hours
B. Maintenance		8 200 hours
1. Preventive Maintenance	1 600 hours	
2. Emergency or Unscheduled Maintenance	1 600 hours	
3. Normal Scheduled Maintenance	5 000 hours	
4. Overtime	585 hours	
C. R&D Assistance		6 200 hours

MD Richmond

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
W. L. Bunch, G. F. Garlick	HWIR-1593, A Method of Measuring Neutron Flux Inside a Reactor
L. A. Bray, E. C. Martin	Use of Sugar to Neutralize Nitric Acid Waste Liquors (HW-75565)
H. L. Brandt	A Device for Effecting Plug Flow of Particulate Solids in Liquid Media
O. H. Koski	Automatic Solvent Extraction Control System

H M Parker

Manager, Hanford Laboratories

1235724